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OFFICE FOR

ANALYSIS and EVALUATION of OPERATIONAL DATA

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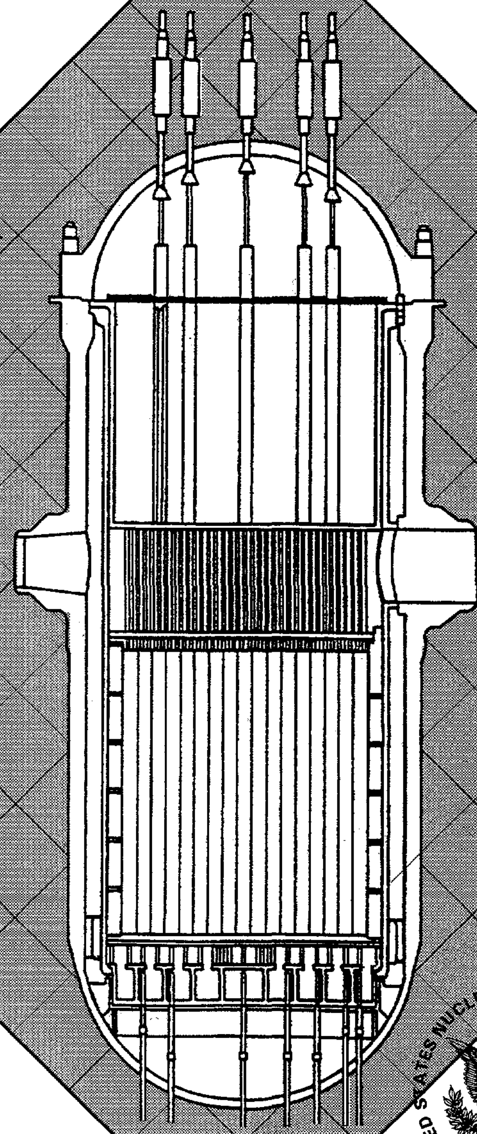

ANNUAL REPORT, 1996

REACTORS

DECEMBER 1997

U.S. NUCLEAR
REGULATORY
COMMISSION

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, NUREG-1272, AEOD/S804
  Vol. 2, No. 1, Power Reactors, October 1988
  Vol. 2, No. 2, Nonreactors, October 1988

- *Office for Analysis and Evaluation of Operational Data 1988 Annual Report*, NUREG-1272
  Vol. 3, No. 1, Power Reactors, June 1989
  Vol. 3, No. 2, Nonreactors, June 1989

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

- *Office for Analysis and Evaluation of Operational Data 1990 Annual Report*, NUREG-1272
  Vol. 5, No. 1, Power Reactors, July 1991
  Vol. 5, No. 2, Nonreactors, July 1991

- *Office for Analysis and Evaluation of Operational Data 1991 Annual Report*, NUREG-1272
  Vol. 6, No. 1, Power Reactors, July 1992,
  Vol. 6, No. 2, Nonreactors, August 1992

- *Office for Analysis and Evaluation of Operational Data 1992 Annual Report*, NUREG-1272
  Vol. 7, No. 1, Power Reactors, July 1993
  Vol. 7, No. 2, Nonreactors, October 1993

- *Office for Analysis and Evaluation of Operational Data 1993 Annual Report*, NUREG-1272
  Vol. 8, No. 1, Power Reactors, November 1994
  Vol. 8, No. 2, Nuclear Materials, May 1995

- *Office for Analysis and Evaluation of Operational Data 1994-FY95 Annual Report*, NUREG-1272
  Vol. 9, No. 1, Power Reactors, July 1996
  Vol. 9, No. 2, Nuclear Materials, September 1996
  Vol. 9, No. 3, Technical Training, September 1996

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ABSTRACT

The United States (U.S.) Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) has published reports of its activities since 1984. The first report covered January through June of 1984, and the second report covered July through December of 1984. After those first two semiannual reports, AEOD published annual reports of its activities from 1985 through 1993. Beginning with the report for 1986, AEOD Annual Reports have been published as NUREG-1272. Beginning with the report for 1987, NUREG-1272 has been published in two parts, No. 1 covering power reactors and No. 2 covering nonreactors (changed to "nuclear materials" with the 1993 report). AEOD changed its annual report from a calendar year (CY) to a fiscal year report, and added part No. 3 covering technical training, beginning with the combined Annual Report for CY 1994 and fiscal year 1995, NUREG-1272, Vol. 9, Nos. 1-3

This report, NUREG-1272, Vol. 10, No. 1, covers power reactors and presents an overview of the fiscal year 1996 operating experience of the nuclear power industry from the NRC perspective. NUREG-1272, Vol. 10, No. 2, covers nuclear materials and presents a review of the events and concerns associated with the use of licensed material in applications other than power reactors. NUREG-1272, Vol. 10, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission. Throughout these reports, whenever information is presented for a calendar year, it is so designated. Fiscal year information is designated by the four digits of the fiscal year.

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ABBREVIATIONS

A

ac	alternating current
ADS	automatic depressurization system
AEOD	Analysis and Evaluation of Operational Data (NRC Office for)
AFW	auxiliary feedwater
AISI	American Iron and Steel Institute
AIT	augmented inspection team
ALARA	as low as reasonably achievable
AMS	Allegations Management System (NRC)
ANSI	American National Standards Institute
AO	abnormal occurrence
ARG	Accident Review Group (NRC)
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
ATWS	anticipated transient without scram

B

B&W	Babcock & Wilcox Company
BWR	boiling-water reactor
BWRO	BWR Owners Group

C

CDP	core damage probability
CCDP	conditional core damage probability
CCF	common-cause failure
CCW	component cooling water
CFR	Code of Federal Regulations
CIV	containment isolation valve
CRD	control rod drive
CRGR	Committee to Review Generic Requirements (NRC)
CS	core spray
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CST	condensate storage tank
cSv	centisievert
CVCS	chemical and volume control system
CW	circulating water
CY	calendar year

D

DBR	design basis reconstitution
dc	direct current
DE	diagnostic evaluation/Division of Engineering (NRR)
DEP	Diagnostic Evaluation Program (NRC)
DER	design electrical rating
DET	diagnostic evaluation team
DG	diesel generator
DRPM	Division of Reactor Program Management (NRR)
DSA	diagnostic self-assessment

E

ECCS	emergency core cooling system
EDG	emergency diesel generator
EDO	Executive Director for Operations (NRC)
EFO	equipment forced outage
EFW	emergency feedwater
EHC	electrohydraulic control
EMEB	Mechanical Engineering Branch (NRR)
EOP	emergency operating procedure
EPIP	emergency plan implementing procedure
EPIX	Equipment Peformance and Information Exchange System (INPO)
EPRI	Electric Power Research Institute
ERDS	Emergency Response Data System (NRC)
ESF	engineered safety feature
ESW	essential service water

F

FOR	forced outage rate
FR	Federal Register
FRERP	Federal Radiological Emergency Response Plan
FRP	Federal Response Plan
FSAR	final safety analysis report
FW	feedwater

G

GE	General Electric Company
GL	generic letter

H

HOO	headquarters operations officer (NRC)
HP	high pressure
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger

I

IAEA	International Atomic Energy Agency
IC	isolation condenser
I&C	instrumentation and control
ICS	integrated control system
IEEE	Institute of Electrical and Electronic Engineers
IIP	Incident Investigation Program (NRC)
IIT	incident investigation team
ILRT	integrated leak rate test
IN	information notice (NRC)
INES	International Nuclear Event Scale (IAEA)
INEX2	Second International Emergency Exercise
INPO	Institute of Nuclear Power Operations
IPE	individual plant examination
IPEEE	individual plant examination for external events
IRM	intermediate range monitor
IRS	Incident Reporting System (NEA)
ISA	independent safety assessment
ISI	inservice inspection

L

LER	licensee event report
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LP	low pressure

M

MCC	motor control center
MDC	maximum dependable capacity
MG	motor-generator
MOOS	maintenance-out-of-service
MOV	motor-operated valve
mrem	millirem
MSB	multi-assembly sealed basket
MSIV	main steam isolation valve
MSL	main steamline
MSR	moisture separator reheater
MTC	multi-assembly transfer cask
MW	megawatt
MWe	megawatts-electric
MYAPS	Maine Yankee Atomic Power Station

N

NEA	Nuclear Energy Agency (OECD)
NMAC	Nuclear Maintenance Assistance Center (EPRI)
NMSS	Nuclear Material Safety and Safeguards (NRC Office of)
NOUE	notification of unusual event
NPRDS	Nuclear Plant Reliability Data System (INPO)
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation (NRC Office of)
NSSS	nuclear steam supply system
NTSB	National Transportation Safety Board
NUDOCS	Nuclear Documents System

O

OECD	Organization for Economic Cooperation and Development
OIG	Office of the Inspector General (NRC)
OP	Office of Personnel (NRC)

P

PDR	Public Document Room (NRC)
PECB	Events Assessment and Generic Communications Branch (NRR)
PI	performance indicator
PIM	plant issues matrix
PORV	power-operated relief valve
PPR	plant performance review
PRA	probabilistic risk assessment
PTM	plant temporary modification
PWG	principal working group (NEA)

PWR pressurized-water reactor

R

RCIC reactor core isolation cooling
RCM Response Coordination Manual (NRC)
RCP reactor coolant pump
RCS reactor coolant system
REIRS Radiation Exposure Information and Reporting System (RES)
rem roentgen equivalent man
RES Nuclear Regulatory Research (NRC Office of)
RG regulatory guide
RHR residual heat removal
RI Region I (NRC) King of Prussia, PA
RII Region II (NRC) Atlanta, GA
RIII Region III (NRC) Lisle, IL
RIV Region IV (NRC) Arlington, TX
RPS reactor protection system
RPV reactor pressure vessel
RTD resistance temperature detector
RTM Response Technical Manual (NRC)
RWCU reactor water cleanup

S

SALP systematic assessment of licensee performance (NRC)
SBGT standby gas treatment
SBLC standby liquid control
SCSS Sequence Coding and Search System (NRC)
SE significant event (NRC)
SER Safety Evaluation Report (NRC)
SET Special Evaluation Team (NRC)
SFP spent fuel pool
SG steam generator
SI safety injection
SOV solenoid-operated valve
SRM staff requirements memorandum/ source range monitor
SRV safety/relief valve
SSA safety system actuation (NRC)
SSES Susquehanna Steam Electric Station
SSF safety system failure (NRC)
SW service water

T

T/C thermocouple
TDP turbine-driven pump
TEDE total effective dose equivalent
TGIS toxic gas isolation system
TMI Three Mile Island
TS technical specifications

U

U.K. United Kingdom
U.S. United States

V

VAT vulnerability assessment team
V&V validation and verification

EXECUTIVE SUMMARY

General

The Office for Analysis and Evaluation of Operational Data (AEOD) was created in 1979 to provide a strong, independent capability to analyze and evaluate operational safety data associated with activities licensed by the United States (U.S.) Nuclear Regulatory Commission (NRC). AEOD is also responsible for the NRC's Incident Response Program, Incident Investigation Program, and Technical Training Program. In addition AEOD provides management direction and oversight of independent safety inspections, as well as administrative and technical support to the NRC's Committee to Review Generic Requirements. AEOD also obtains industry feedback on these activities.

The AEOD programs constitute the essential independent review and assessment of power reactor and nuclear materials safety performance, and complement the regional, the Office of Nuclear Reactor Regulation (NRR), and the Office of Nuclear Material Safety and Safeguards reviews of operating events. They perform a quality verification function that provides assurance of feedback of important operational safety lessons. AEOD findings and recommendations continue to be addressed through generic correspondence, in the resolution of generic issues, and in initiatives taken by industry.

AEOD has published annual reports of its activities since 1985. AEOD changed its annual report from a calendar year (CY) to a fiscal year report beginning with the combined Annual Report for CY 1994 and fiscal year 1995, NUREG-1272, Vol. 9, Nos. 1-3. This report, NUREG-1272, Vol. 10, No. 1, covers power reactors and presents an overview of the fiscal year 1996 operating experience of the nuclear power industry from the NRC perspective. Throughout this report, whenever information is presented for a calendar year, it is so designated. Fiscal year information is designated by the four digits of the fiscal year.

Nuclear Reactor Safety Performance

Through the many activities of AEOD, trends in overall safety performance of power reactors may be inferred. The Performance Indicator (PI) and Accident Sequence Precursor (ASP) Programs of AEOD have been applied to analyze data and information in a consistent manner over a number of years. These programs show a substantial reduction in safety-significant operational events since 1985. The number of initiating events resulting in scrams has declined significantly over the past ten years, and this is reflected in fewer and less complicated plant transients (safety system actuations and significant events). In 1996 the industry average number of scrams, safety system actuations, and significant events continued to decline slightly. However, equipment problems persist, as evidenced by the percentage of scrams caused by equipment failure (the leading cause of all scrams), and the lack of sustained improvement in safety system failures, forced outage rate, and equipment forced outages per 1000 critical hours. In 1996 safety system failures, forced outage rate, equipment forced outage rate, and collective radiation exposure leveled off or worsened. Although average unit availability has improved considerably over the past 10 years, this has been due not to fewer forced outage hours but to greatly reduced scheduled outage hours. This is a consequence of longer fuel cycles, which result in greater intervals between refueling outages, and of shorter refueling outages. Implementation of the maintenance rule, and the collection and use of equipment reliability and availability data associated with it, should provide a means to reduce the number of safety system failures as well as both the number and duration of forced shutdowns.

Operating Experience Feedback

Performance Indicators. The PI program includes eight indicators: automatic scrams while critical, safety system actuations, significant events, safety system failures, forced outage rate, equipment-forced outages per 1000 critical hours, collective radiation exposure, and cause codes. PI reports are issued annually in January of each year with data through the previous fiscal year; they are distributed widely within the NRC and to all operators of commercial nuclear power plants. They are used in various NRC programs such as the Senior Management Meeting process and in plant-specific analyses of safety performance. Industry average PIs have been used for the past nine years to monitor trends in the safety performance of the commercial nuclear power industry.

Abnormal Occurrences. AEOD administers the Commission's program for reporting abnormal occurrences (AOs) to Congress. AOs are incidents or events that the Commission determines are significant from the standpoint of public health and safety. Beginning in 1996, AO reports are issued annually with data through the previous fiscal year. The AO report for 1996 (NUREG-0090, Vol. 19) contains two AOs for events at nuclear power plants. One involved a plant trip with multiple complications at Wolf Creek Nuclear Generating Station, and the other involved containment bypass leakage via disconnected hydrogen monitor lines at Braidwood Units 1 and 2. The number of AOs at nuclear power plants since 1988 has remained low, averaging just over two per year.

Radiation Exposures. The NRC regulates both reactor and nonreactor applications of nuclear materials. All NRC licensees are required to monitor employee exposure to radiation and radioactive materials at levels sufficient to demonstrate compliance with the occupational dose limits specified in Part 20 of Title 10 of the *Code of Federal Regulations* (CFR). Licensees of power reactors are required by 10 CFR 20.2206 to provide to the NRC annual reports of exposure data for individuals for whom personnel monitoring is required. Almost all radiation doses from nuclear power plants are occupational doses, that is, doses to nuclear power plant employees and contractors who work at the plant. The economics of operating a plant creates a strong impetus to reduce exposures and achieve ALARA (as low as reasonably achievable) objectives. As a result, utility violations of NRC limits on personnel exposure are rare, and the vast majority of nuclear power plant personnel have annual exposures far below NRC regulatory limits specified in 10 CFR Part 20. This is believed to result primarily from the licensees' extensive dose-reduction efforts. Some measures that reduce collective exposure are an effective maintenance program, experienced and well-trained personnel, a good water chemistry control program, effective decontamination and cleanup practices, good fuel cladding integrity, effective radiation exposure control programs, good housekeeping, and an alert health physics staff. The average dose per worker has declined from 0.94 centisievert (cSv [rem]) in CY 1973 to 0.31 cSv (rem) in CY 1995 (the latest year for which data are available).

AEOD Reliability and Risk Activities

Accident Sequence Precursor Program. The ASP Program uses probabilistic risk assessment (PRA) techniques to evaluate the conditional core damage probabilities of nuclear power plant events and equipment unavailabilities. It serves as one of several tools to ensure that important operating lessons are not overlooked. The program uses a rigorous method that integrates actual initiating events, plant conditions, and the reliability and availability of standby safety equipment into an overall quantitative assessment, which is expressed as a conditional core damage probability (CCDP) for initiating events and an increase in core damage probability (ΔCDP) for conditions and equipment unavailabilities.

Results of the ASP Program are peer-reviewed by outside consultants, other NRC offices, and the affected licensees. They are used in NRC initiatives such as the Senior Management Meeting process. There were ten events in FY 1995 that met the criterion for an ASP event (CCDP or ΔCDP greater than 10^{-6}), two caused by initiating events and eight due to conditions or equipment unavailabilities.

System Reliability Studies. AEOD uses operational data to determine the reliability of risk significant systems in U.S. commercial reactors. The data are obtained from licensee event reports (LERs), special reports, and monthly operating experience reports. Each of the studies covers the period from CY 1987 through CY 1993. Three have been completed. Reports on the reliability of the high-pressure coolant injection (HPCI) system in the 23 boiling-water reactors (BWRs) with HPCI systems, and the emergency diesel generator (EDG) trains in all plants with EDGs, were completed in prior years. The third study, on the reliability of the isolation condenser (IC) system at the five BWRs with that system, was completed in 1996. The best estimate of IC train unreliability (including recovery), based on operational experience data, is 0.02. The failure to operate failure-mode of the IC train and failure to provide makeup water to the isolation condenser contributed equally to the overall unreliability. The recovered and non-recovered train unreliability estimates differ by a factor of five. The difference is primarily attributable to the spurious isolations of the IC train, as observed in the unplanned demands. All of the IC train failures to operate were caused by spurious isolation of the IC train.

Common-Cause Failure Database. AEOD has compiled common-cause failure events from LERs and data records contained in the Nuclear Plant Reliability Data System. These events are contained in the common-cause failure database. This database represents the most complete collection of common-cause failure events in the world. The initial database was completed in December 1995 with associated technical documentation (6 volumes). Technical review of the database and draft technical reports have been completed.

Results of AEOD Studies

AEOD studies of operational experience are broadly disseminated throughout the nuclear community and to the public. They provide a basis for decision-making based on operational experience. AEOD used a systematic process to nominate, prioritize, and select safety issues to be studied, and continued its efforts to more effectively communicate the lessons of operating experience through a variety of forums.

In 1996 the AEOD staff reviewed a broad spectrum of data and issued seven special studies, four engineering evaluations, and five technical reviews. These reports covered a wide range of subjects that varied from relatively broad evaluations of aging effects and allegation data, to in-depth reviews of subjects such as the Emergency Diesel Generator Power System and steam generator tube failures. Several of the studies resulted in interactions of AEOD staff with industry groups to facilitate resolution of the underlying technical problems. As a result of AEOD's 1992 study entitled *Safety and Safety/Relief Valve Reliability (AEOD/S92-02)*, the staff worked with the Electric Power Research Institute/Nuclear Maintenance Assistance Center to develop a maintenance good practices guide for pressure relief valves for use by nuclear power plant maintenance organizations. The AEOD studies *Evidence of Aging Effects on Certain Safety-Related Components (NUREG/CR-6442)* and *Steam Generator Tube Failures (NUREG/CR-6365)* were prepared for the Nuclear Energy Agency of the Organization for Economic Cooperation and Development as part of an international effort on these two topics. Also in 1996, as a follow-on to AEOD's 1994 *Operating Experience Report - Reliability of Safety-Related Steam Turbine-Driven Standby Pumps (NUREG-1275, Vol. 10)*, AEOD continued to participate in the Terry Turbine Users Group.

Operating Experience Data

The average number of LERs per plant (excluding supplemental, canceled, proprietary, voluntary, and safeguards LERs) has declined from about 26 in 1987 to about 11 in 1996. AEOD uses the Sequence Coding and Search System (SCSS) for storing and retrieving LER information. In 1996 AEOD staff used the SCSS data to support NRC activities such as customized inspection programs and senior management meetings. The SCSS database is also a primary source of operating experience information for NRR, the Office of Nuclear Regulatory Research, and the regions. AEOD also maintains data on LERs, monthly operating reports, and plant outages to generate the NRC's Performance Indicator Reports.

Incident Response

Operations Center. The NRC Operations Center provides the focal point for NRC communications with NRC licensees, State agencies, and other Federal agencies about operating events. The center contains a state-of-the-art information management system which integrates voice, video, and data systems to provide timely and effective information flow.

In 1996 the NRC Operations Center received 1677 immediate notifications, primarily from nuclear power plant licensees (1415). Of the 1415 nuclear power plant events, 65 were classified as "Unusual Events" and 5 as "Alerts." None were classified as a "Site Area Emergency" or a "General Emergency." The NRC entered the Monitoring Phase of the Normal Mode for two of the five Alerts reported. The NRC also entered the Monitoring Phase for four of the Unusual Events, all involving external events.

Emergency Exercises. Emergency exercises are held periodically to ensure that response organizations of the NRC, the licensees, the States, and other Federal agencies are proficient in dealing with each type of emergency. In 1996 the NRC headquarters and regional offices participated in four full scale emergency exercises and six limited participation exercises at nuclear power plants. The NRC initiated a new exercise element in 1996 called ingestion exercises, and conducted three of them at nuclear power plants.

State Outreach. AEOD continued its aggressive State Outreach Program designed to increase and improve interactions with States during events and exercises. This year AEOD expanded the program to include training on the response manuals. Outreach sessions were conducted with 15 states and numerous licensees. The NRC also negotiated a Memorandum of Understanding to make the Emergency Response Data System available to the States of Louisiana, Wisconsin, and Iowa. In addition, the State of Vermont has requested ERDS and negotiations are in progress.

Coordination with Other Federal Agencies. The NRC continued its participation with other Federal agencies in the issuance of the Federal Radiological Emergency Response Plan (FRERP). The NRC also participated in drafting the Radiological Incident Annex to the Federal Response Plan (FRP), which describes how the FRP and the FRERP are integrated when both are used in an emergency. In addition, AEOD participated in an activity to evaluate the adequacy of Federal plans in response to nuclear, biological, and chemical terrorist events.

Incident Investigation Program

In 1996 the NRC conducted seven Augmented Inspection Team inspections at nuclear power reactors. There were no events that were judged to have a level of safety significance high enough to warrant an Incident Investigation team investigation. Examples of problems found and communicated to licensees from these Augmented Inspection Team inspections included the potential for one or more control rod assemblies to fail to fully insert following a reactor trip, and the potential for hydrogen gas ignition during closed welding of VSC-24 Multi-Assembly Sealed Baskets.

Independent Safety Assessments

The NRC performed an independent safety assessment of the Maine Yankee Atomic Power Station (MYAPS) in response to an anonymous allegation and the concerns of the Governor of Maine. The Independent Safety Assessment Team found that, while overall MYAPS performance was adequate for operation, a number of deficiencies existed. These deficiencies included poor problem identification and resolution, weaknesses in the scope, rigor, and evaluation of testing, and declining material condition. The root causes were determined to be economic pressure to reduce costs and the lack of a questioning attitude that resulted in the failure to identify or promptly correct significant problems.

International Exchange of Information

The Incident Reporting System. The Incident Reporting System (IRS) is a cooperative program of the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA) of the United Nations. Member countries submit reports of operational experience that may be applicable to other nuclear power plants. This broadens the operational experience database to include all nuclear power programs of the member states of the NEA and the IAEA. In 1996 AEOD prepared and submitted 67 IRS reports that addressed individual operational events and various generic concerns. AEOD also reviewed approximately 110 IRS reports from other countries and disseminated the applicable information to the NRC staff and to the Institute of Nuclear Power Operations.

International Support Activities. AEOD exchanges information and ideas on a variety of topics of international interest, such as emergency response, control rod insertion problems, undetected safety system failures, and common cause failures. AEOD is also the principal U.S. technical representative on reactor operating experience to the NEA's Committee on the Safety of Nuclear Installations' (CSNI) Principal working Group 1, "Operating Experience and Human Factors." In addition, AEOD is a participant in the Expert Group on Nuclear Emergency matters, established to improve the quality of national and international nuclear emergency arrangements.

Lisbon Initiative Activities. AEOD continued to assist Russia and Ukraine in the development of their own emergency response capabilities. The AEOD staff helped the regulatory authorities in each country to establish reliable communications with each site, to prepare response plans and procedures, and to provide equipment for a basic emergency response center. AEOD also assisted Ukraine in establishing an incident reporting and operating experience feedback system, including data collection, events analysis and evaluation, regulatory response to events, and experience feedback to nuclear plants.

Limited Participation in the International Nuclear Events Scale. The NRC has participated in a limited manner in the International Nuclear Events Scale (INES) since December 1992. INES is a ranking system that, in principle, is used to promptly and consistently communicate to the public the safety significance of reported events at nuclear installations worldwide. After a two year trial period, the NRC decided to continue indefinitely its limited participation in INES. AEOD submitted reports of six events that occurred in 1996 to INES.

1 INTRODUCTION

The Office for Analysis and Evaluation of Operational Data (AEOD) was created in 1979 to provide a strong, independent capability to analyze and evaluate operational safety data associated with activities licensed by the United States (U. S.) Nuclear Regulatory Commission (NRC). The office serves as the focal point for the assessment of operational events through the collection, review, analysis, and evaluation of the safety performance of both reactor and nuclear materials facilities. To accomplish this mission for commercial nuclear power reactors, AEOD (1) collects, analyzes, and disseminates operational data; (2) identifies important events and their associated safety concerns and root causes; (3) assesses the adequacy of corrective actions taken to address safety concerns; (4) determines the generic applicability of events to other nuclear power plants; (5) assesses trends in performance; (6) evaluates operating experience to quantify and to improve the understanding of the risk-significance of events; (7) conducts reliability studies of risk-important systems; (8) analyzes human performance in operating events; and (9) produces periodic Performance Indicator, Abnormal Occurrence, and Accident Sequence Precursor Reports.

AEOD is also responsible for the NRC's Incident Response Program, Incident Investigation Program, and Technical Training Program. The Incident Response Program provides a coordinated NRC emergency response to ongoing events through the NRC Operations Center. The Incident Investigation Program provides a structured NRC investigative response to significant operational events according to their safety significance. The Technical Training Program provides initial and continuing technical training for NRC staff and contractors. In addition, AEOD provides management direction and oversight of independent safety inspections, as well as administrative and technical support to the NRC's Committee to Review Generic Requirements. AEOD also obtains industry feedback on these activities.

AEOD reviews and evaluations include the following specific functions:

- identification of operational safety data needed to support safety analyses, and development of agency-wide reporting of these data and the methods and systems to retrieve them
- analysis of operational safety data and identification of safety issues that require new or additional NRC staff actions
- development of a coordinated system for feedback of operational safety information to NRC offices, licensees, and other organizations, as appropriate
- serving as the focal point for coordinating generic operational safety information and data systems with industry, foreign governments, and other agencies involved with the collection, analysis and feedback of operational data
- development and implementation of the agency program on reactor performance indicators for use by senior managers
- analysis of selected operating events using the Accident Sequence Precursor program to gain insight into events and to improve the understanding of them from a risk perspective
- studies of the impact of human performance
- preparation of the Abnormal Occurrence Report to Congress
- continuous staffing of the NRC Operations Center to screen reactor and nuclear materials events and any other information reported to the center to

ensure appropriate NRC response to reported events

- development, in consultation with other NRC offices, of the NRC policy for response to incidents and emergencies, as well as assessment of the NRC response capabilities and performance

- development of policy, procedures, and program requirements for NRC incident investigations of significant operational events

- tracking of the recommendations and staff actions contained in AEOD studies, incident investigation team reports and independent safety assessments until they are resolved

- 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

The AEOD programs, taken as a whole, constitute the essential independent review and assessment of power reactor and nuclear materials safety performance, and complement the regional, the Office of Nuclear Reactor Regulation, and the Office of Nuclear Material Safety and Safeguards reviews of operating events. They perform a quality verification function that provides assurance of feedback of important operational safety lessons. AEOD findings and recommendations continue to be addressed through generic correspondence, in the resolution of generic issues, and in initiatives taken by industry.

In 1996, as a consequence of the elimination of its responsibility for oversight and administration of the Diagnostic Evaluation Program, AEOD's Incident Response Division was reorganized. AEOD now consists of three divisions organized as follows: the Incident Response Division, which includes the Response Operations Section, the Response Coordination Section, and the Operations Officer Section; the Safety Programs Division, which includes the Reactor Analysis Branch and the Reliability and Risk Assessment Branch; and the Technical Training Division, comprised of the Reactor Technology Training Branch, the Specialized Technical Training Branch, and the Technical Training Support Branch.

AEOD changed its annual report from a calendar year (CY) to a fiscal year report beginning with the combined Annual Report for CY 1994 and fiscal year 1995, NUREG-1272, Vol. 9, Nos. 1-3. This report, NUREG-1272, Vol. 10, No. 1, covers power reactors and presents an overview of the fiscal year 1996 operating experience of the nuclear power industry from the NRC perspective, including trends of some key performance measures. It also incudes the principal findings identified in AEOD studies of power reactor events and issues during the year as well as a summary of information from licensee event reports, independent safety assessments, and the NRC Operations Center. Throughout this report, whenever information is presented for a calendar year, it is so designated. Fiscal year information is designated by the four digits of the fiscal year. The report also includes the following appendices:

- Appendix A contains data from 1996 to support the section on operational experience

- Appendix B lists and summarizes 1996 Abnormal Occurrences

- Appendix C lists AEOD reports issued in 1996

- Appendix D lists AEOD reports issued from CY 1980 through CY 1995

- Appendix E presents all AEOD reports from CY 1980 through 1996 sorted by subject

- Appendix F presents the status of recommendations contained in AEOD studies

- Appendix G presents the status of NRC staff actions resulting from the findings of NRC Incident Investigation Teams

- Appendix H presents the status of NRC staff actions involving potential generic issues resulting from the findings of NRC Diagnostic Evaluation Teams

- Appendix I presents the status of NRC staff actions involving potential generic issues resulting from the NRC/Institute of Nuclear Power Operations (INPO) team review of the effects of Hurricane Andrew on Turkey Point Units 3 and 4

- Appendix J presents the status of NRC staff actions involving potential generic issues resulting from the findings of the NRC Independent Safety Assessment of Maine Yankee Atomic Power Station

The report on nuclear materials, NUREG-1272, Vol. 10, No. 2, presents a review of the events and concerns during 1996 associated with the use of licensed material in applications other than power reactors. NUREG-1272, Vol. 10, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission.

2 OPERATING EXPERIENCE FEEDBACK

2.1 Operating Performance

AEOD collects, analyzes, and disseminates a wide range of operational data, obtained primarily from immediate notifications to the NRC Operations Center in accordance with Section 50.72 of Title 10 of the *Code of Federal Regulations* (CFR), licensee event reports (LERs) submitted in accordance with 10 CFR 50.73, monthly operating reports submitted in accordance with plant Technical Specifications (TS), and the database of component failures in the Nuclear Plant Reliability Data System managed by the Institute of Nuclear Power Operations (INPO). Other operational data include 10 CFR Part 21 reports, NRC regional inspection reports, preliminary notifications of events or unusual occurrences issued by the NRC, quarterly collective radiation exposures from INPO, and allegations of impropriety or inadequacy received by the NRC. A subset of this information is monitored in the NRC Performance Indicator (PI) Program: (1) automatic scrams while critical, (2) safety system actuations, (3) significant events, (4) safety system failures, (5) forced outage rate, (6) equipment forced outages per 1000 commercial critical hours, (7) collective radiation exposure, and (8) cause codes.

Figure 2.1 presents industry-wide annual averages since 1985 for seven of the PIs that AEOD monitors as indicators of plant performance. With the exception of collective radiation exposure, plants in extended shutdown which require Commission approval for either restart or operation above low power are excluded from the calculation of industry average PIs. Radiation exposure can be significant during extended outages, hence these data are not excluded. Additionally, plants are excluded after they are permanently shut down.

This section presents the results of analyses of selected operational experience data for calendar years 1992 through 1995 and fiscal year 1996 (note that quarter 95-4 data are included in both calendar year 1995 and fiscal year 1996). It also presents statistical analyses of trends over the past 9 years in six of the PIs. The operational data collected within the PI Program are presented in Appendix A-1 and other plant operational experience data are shown in Appendix A-2.

2.1.1 Reactor Scrams

AEOD monitors reactor scrams that occur while the affected reactor is critical. Reactor scrams can result from initiating events that range from relatively minor incidents to precursors of accidents. Automatic scrams are included in the PI Program (see Table A-1.1 of Appendix A-1). AEOD also tracks manual scrams and total scrams per 1000 critical hours (see Table A-2.1 of Appendix A-2). Tables A-2.2 through A-2.5 of Appendix A-2 summarize statistical data on combined automatic and manual reactor scrams.

Figure 2.2 shows the industry trend in scram causes for 1992 through 1996. Equipment failures remain the leading cause of scrams. Of the scrams caused by equipment failures during 1996, over half were initiated by problems in four systems: feedwater, main turbine and control, main generator, and electrical. Figure 2.3 shows that, in 1996, over half of all scrams occurred during normal plant operation.

Automatic Reactor Scrams. Almost three-fourths of the scrams that occurred in the last 5 years were automatic scrams. The leading causes of automatic scrams, the dominant initiating systems of those scrams, and the activities in progress are the same as those given for total scrams. The number of automatic scrams has decreased since 1992, with the largest drop occurring from 1992 to 1993; it continued to decrease in 1996.

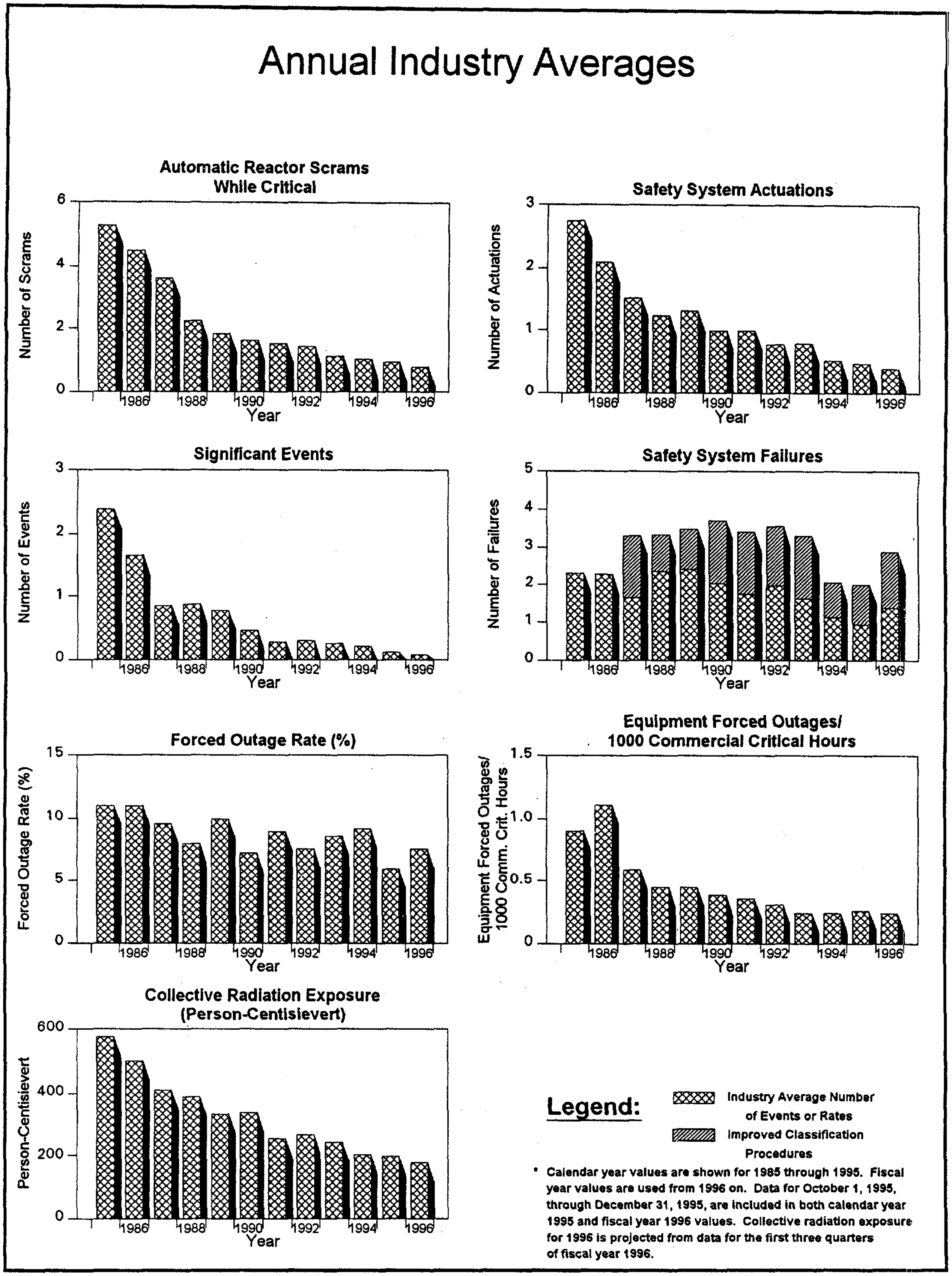


Figure 2.1 Performance Indicators - Annual Industry Averages

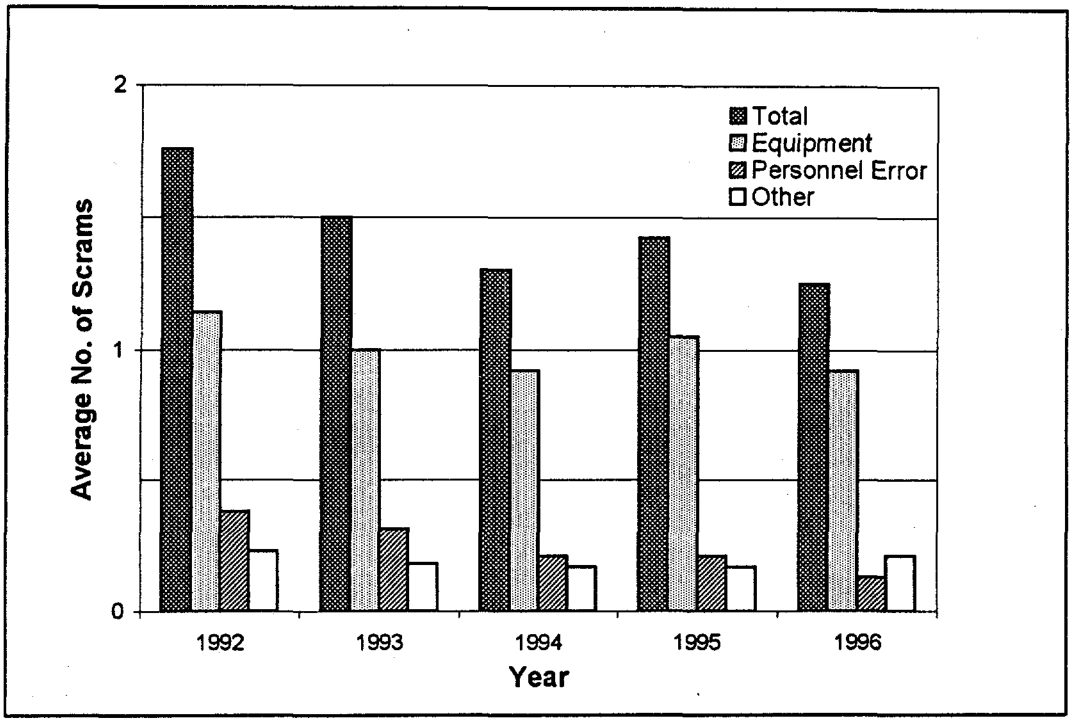


Figure 2.2 Reactor Scrams - Causes

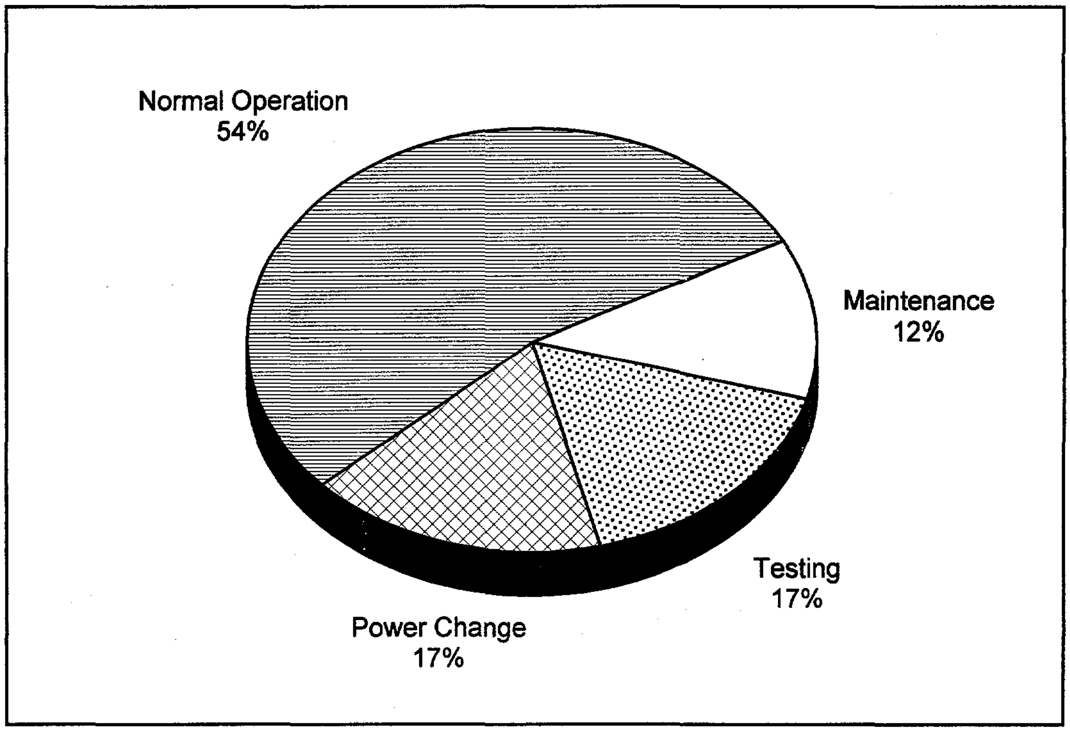


Figure 2.3 Reactor Scrams - Activity in Progress

Manual Reactor Scrams. Slightly more than one-fourth of all scrams during the past 5 years were manual scrams. The number has fluctuated from year to year but has averaged about 44 per year. Since total scrams have declined, the percentage of manual scrams has increased over the past 5 years.

2.1.2 Engineered Safety Features Actuations

AEOD monitors actuations of all engineered safety features (ESFs), a subset of which are included as Safety System Actuations (SSAs) in the PI Program. The SSA PI includes manual or automatic actuations of certain emergency core cooling systems (ECCS) and actuations of the emergency ac power system in response to low voltage on a vital bus. Data for SSAs may be found in Table A-1.2 of Appendix A-1. The number of SSAs has declined steadily since CY 1992.

Figure 2.4 shows the industry trend in total ESF actuations for the past 5 years, including trends in actuations of heating, ventilation, and air conditioning (HVAC) systems, emergency power systems, and ECCS. The number of ESF actuations has decreased each year since CY 1992. The largest decrease occurred between CY 1992 and CY 1993, primarily because the requirement to report actuations of certain HVAC systems ended in October 1992. Tables A-2.6 through A-2.9 of Appendix A-2 present industry data for all ESF actuations.

Boiling-water reactor (BWR) plants have more safety systems that are included in ESF counts than do pressurized-water reactor (PWR) plants. For example, an additional row is provided in Table A-2.7 for the reactor water cleanup (RWCU) system in BWRs. As shown in Table A-2.7, the isolation of this system accounts for a significant percentage of ESF events. Overall, the number of isolations of the RWCU system

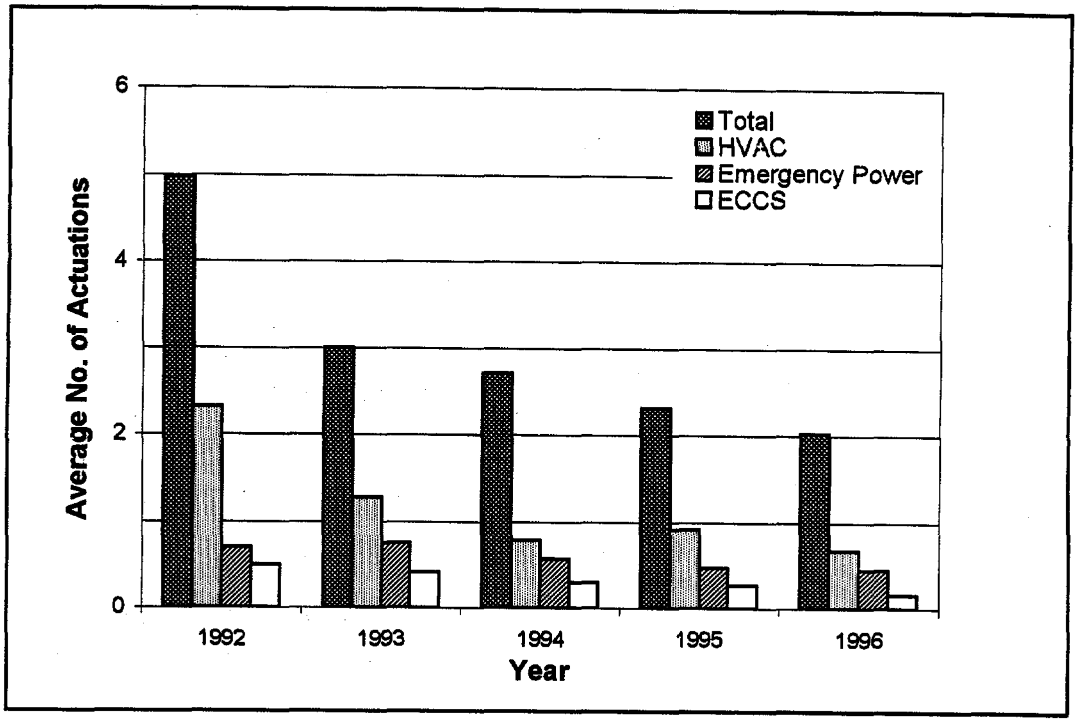


Figure 2.4 ESF Actuations - Selected Systems

declined during the last 5 years. The requirement to report invalid RWCU isolations ended in October 1992, thereby reducing the number of recent reports.

2.1.3 Significant Events

Significant Events (SEs) are those events that the NRC staff identifies for the PI Program as meeting one or more of the following criteria:

- degradation of important safety equipment
- a major transient or an unexpected plant response to a transient
- degradation of fuel integrity, the primary coolant pressure boundary, or important associated structures
- a reactor trip with complications
- an unplanned release of radioactivity exceeding the TS or regulations
- operation outside the TS limits
- other events considered significant

Figure 2.1 shows the industry trend in the average number of SEs since CY 1987. The number of SEs decreased steadily from 32 in CY 1992 to 8 in 1996. Table A-1.3 of Appendix A-1 describes the SEs that occurred during 1996. Table A-1.4 of Appendix A-1 contains SE data for quarters 94-4 through 96-3.

2.1.4 Safety System Failures

The Safety System Failure (SSF) PI includes any actual event or condition that could prevent the fulfillment of the safety function of any of 26 safety systems, subsystems, or components. For a system that consists of multiple redundant subsystems or trains, inoperability of all trains constitutes an SSF. An SSF may be indicative of a plant's readiness to respond to anticipated events and postulated accidents. SSFs include unconditional failures (those events or conditions that render the system incapable of performing its safety function in all situations), and conditional failures (conditions that could, in certain specific situations, e.g., a high energy line break or seismic event, prevent the system from performing its safety function). Table A-1.5 of Appendix A-1 provides quarterly plant-specific SSF data for quarters 94-4 through 96-3. Table A-1.6 of Appendix A-1 contains annual SSF data for each plant for CY 1992 through 1996.

The same four system groups as reported in the last AEOD Annual Report continued to be the predominant contributors to SSFs: the ECCS group, the containment and containment isolation systems group, the emergency power systems group, and the control room emergency ventilation systems group. These four groups accounted for more than 60 percent of the failures.

2.1.5 Forced Outage Rate

The Forced Outage Rate (FOR) PI is calculated by dividing the number of forced outage hours in a period by the sum of the generator on-line hours and the forced outage hours. (This information is contained in monthly operating reports.) Forced outages are defined as those outages required to be initiated by the end of the weekend following the discovery of the off-normal condition. The trend in FOR can provide a perspective on overall plant performance.

Figure 2.1 shows that the FOR is slightly lower the past 2 years after remaining relatively constant the previous 8 years. The decrease in CY 1995 may be an indication that the FOR is beginning to improve, although it could simply be a reflection of variability in the indicator. Table A-1.7 of Appendix A-1 presents plant-specific FOR data for quarters 94-4 through 96-3.

2.1.6 Equipment Forced Outages per 1000 Commercial Critical Hours

The Equipment Forced Outage (EFO) PI is the number of forced outages caused by equipment failures in each 1000 hours of operation with the reactor critical after the plant is placed into commercial operation. (This information is contained in the monthly operating reports.) The EFO rate is the inverse of the mean time between forced outages caused by equipment failures. AEOD monitors the EFO rate as an indicator of the effects of equipment problems on overall plant performance.

Figure 2.1 shows that the industry average EFO rate remained relatively constant over the past 4 years after a slight drop in 1992. Table A-1.8 of Appendix A-1 contains quarterly EFO rates for quarters 94-4 through 96-3.

2.1.7 Collective Radiation Exposure

Licensees of power reactors are required by 10 CFR 20.2206 to provide annual reports to the NRC of exposure data for each individual for whom monitoring is required. The PI Program initially included annual collective radiation exposure for each nuclear plant, derived from the data reported as required by 10 CFR 20.2206. Beginning in 1989, the PI Program included quarterly collective radiation exposure received from INPO, who routinely receives collective radiation exposure from each plant on a quarterly basis. AEOD uses the INPO data to provide more timely information without duplicating INPO's effort.

Figure 2.1 shows that the industry average collective radiation exposure reported by commercial reactors declined during 1993 and 1994 and remained relatively constant in 1995 and 1996. Table A-1.9 of Appendix A-1 shows quarterly collective radiation exposures for quarters 94-4 through 96-3 for each plant.

2.1.8 Cause Codes

Tables A-1.10 through A-1.15 of Appendix A-1 show quarterly cause code data for each plant for quarters 94-4 through 96-3. The cause codes indicator is intended to identify possible programmatic deficiencies. Cause codes are developed from data in the Sequence Coding and Search System database. The indicator captures the trends in administrative control problems (Table A-1.10); licensed operator errors (Table A-1.11); other personnel errors (Table A-1.12); maintenance problems (Table A-1.13); design, construction, installation, or fabrication problems (Table A-1.14); and miscellaneous (random failures of electronic piece-parts or those due to external events) (Table A-1.15). Industry averages are not calculated for this indicator.

2.1.9 Unit Operating Factors

Within the context of its safety mission, the NRC is not normally concerned with the availability and capacity factors of nuclear power plant operations. However, because good availability and capacity factors require close managerial involvement in day-to-day operations, efficient and effective outage management, and attention to detail, which are also important in safe plant operation, they can be indirect indicators of safety performance. Availability, capacity, and outage statistics for the U.S. commercial nuclear industry for 1996 are presented in Tables A-2.10 through A-2.12 of Appendix A-2. The industry average unit availability increased from 66.2 percent in 1986 to 81.0 percent in 1996, excluding the Browns Ferry Units when they were in long-term regulatory shutdown.

2.1.10 Statistical Analysis of Some Trends

As part of an assessment of PI trends, AEOD performed a regression analysis to evaluate the rate of change in the following PIs. This analysis updates a similar analysis in the 1994-1995 Annual Report, and adds data for 1996.

- Automatic reactor scrams
- SSAs
- SEs
- SSFs
- FOR
- EFO

To perform the analysis, an exponential model, $y = Ae^{Bx} + C$, was fitted to each PI. In this model, y is the PI value; x is a time increment index in years (with 1988 = 1); and A, B, and C are parameters estimated from the data. Statistical tests were then performed to determine if the estimated parameters A, B, and C were non-zero. In each case, the C term was not significantly different from zero, and were therefore set to zero. Therefore, the simpler nonlinear model $y = Ae^{Bx}$ was fitted to the data. For comparison, a linear model $y = A + Bx$ was also fitted; the two models produced very similar fits, but the nonlinear model has the conceptual advantage of never being negative.

The results of the regression analysis are summarized in Table 2.1. Figures 2.5 through 2.10 show the six PIs listed in Table 2.1. Scrams, SSAs, SEs, and EFO have statistically significant exponential model fits, indicating that a trend is discernable. The SSF and FOR PIs have neither a non-linear nor a linear trend over the 9-year period; FOR was constant (level) during this period and SSFs were modeled as having two different mean values over the periods before and after January 1, 1994. As seen in Figure 2.8, the number of SSFs are consistently lower after 1994 but increased slightly in 1996. This drop was due primarily to the application of an improved SSF definition in 1994 and a decrease in the number of SSFs reported in LERs. Using the improved SSF definition, a safety system declared inoperable per Technical Specifications, which in the past would have been classified as an SSF, would not be counted as an SSF if the licensee produces and documents in the LER an analysis which demonstrates that the system is capable of performing its safety function. The decrease in LERs reporting SSFs appears to be a consequence of a reduction in the number of SSFs discovered during design basis reconstitution efforts. Because of this notable change, a model different from the nonlinear model described previously was fit to the data. Different means were calculated for the periods from 1988 through 1993 and from 1994 through 1996. These means and their associated confidence intervals are shown in Figure 2.8.

Table 2.1 Summary of PI Regression Analysis Trends

Performance Indicator	**1988-1996**
Automatic reactor scrams	Slowly decreasing nonlinear trend
SSAs	Slowly decreasing nonlinear trend
SEs	Slowly decreasing nonlinear trend
SSFs	Level over each of two periods
FOR	Level
EFO	Slowly decreasing nonlinear trend

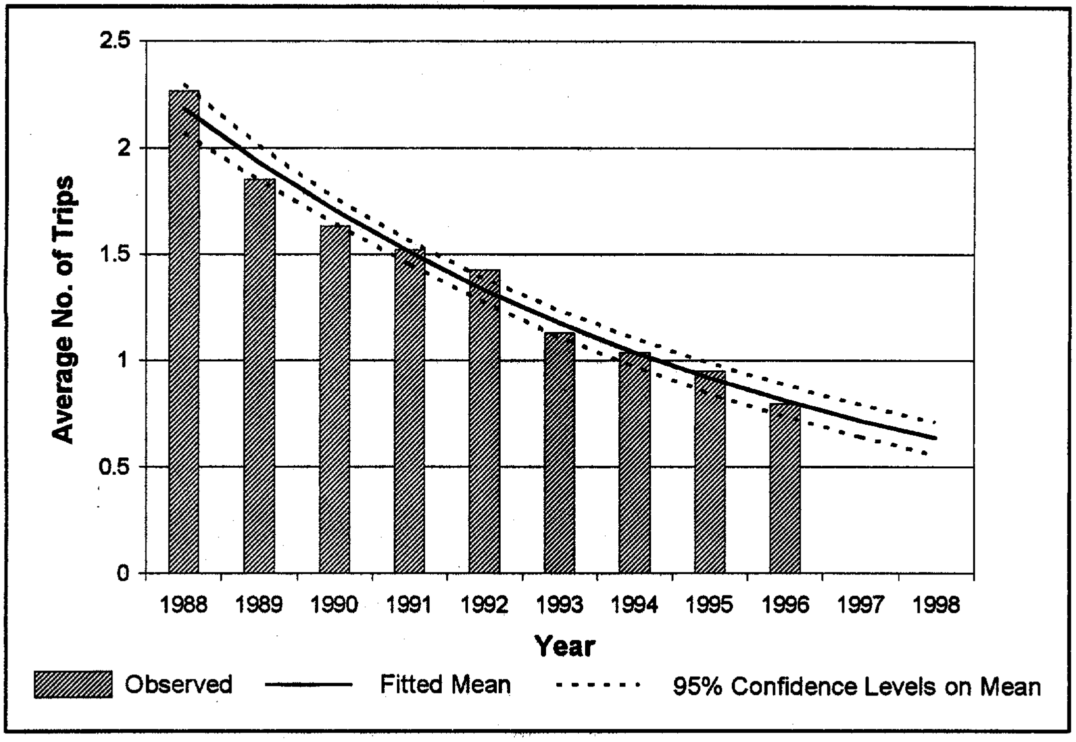


Figure 2.5 Nonlinear Regression Fit for Automatic Scrams ($y = Ae^{Bx}$)

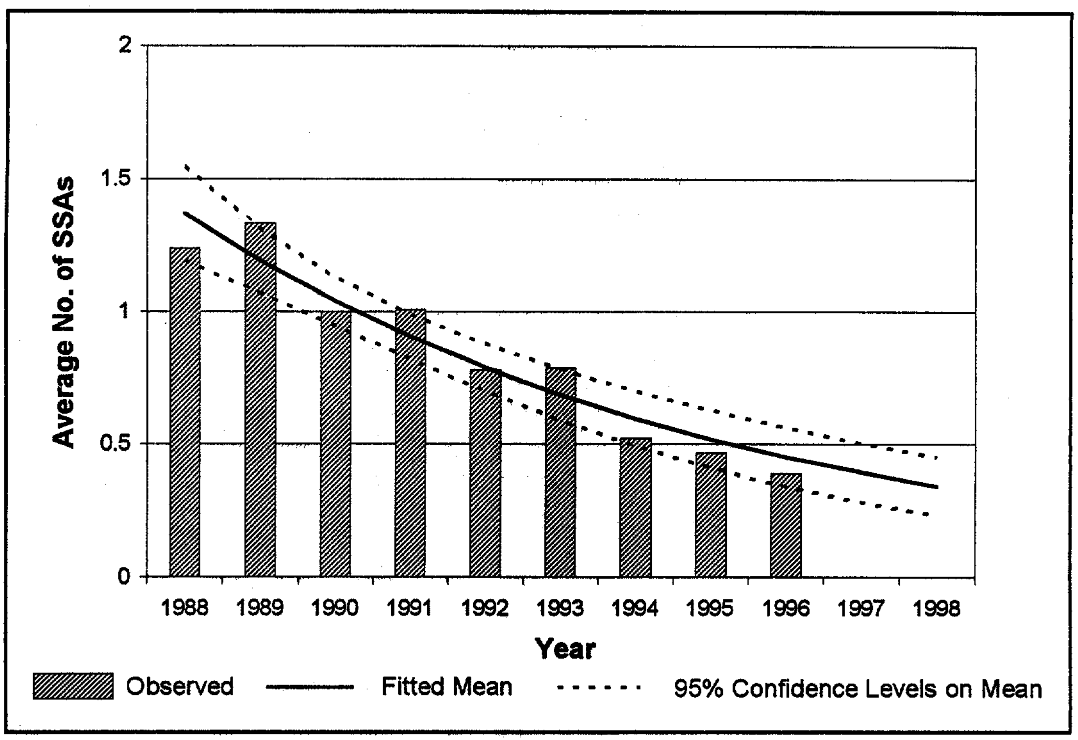


Figure 2.6 Nonlinear Regression Fit for Safety System Actuations ($y = Ae^{Bx}$)

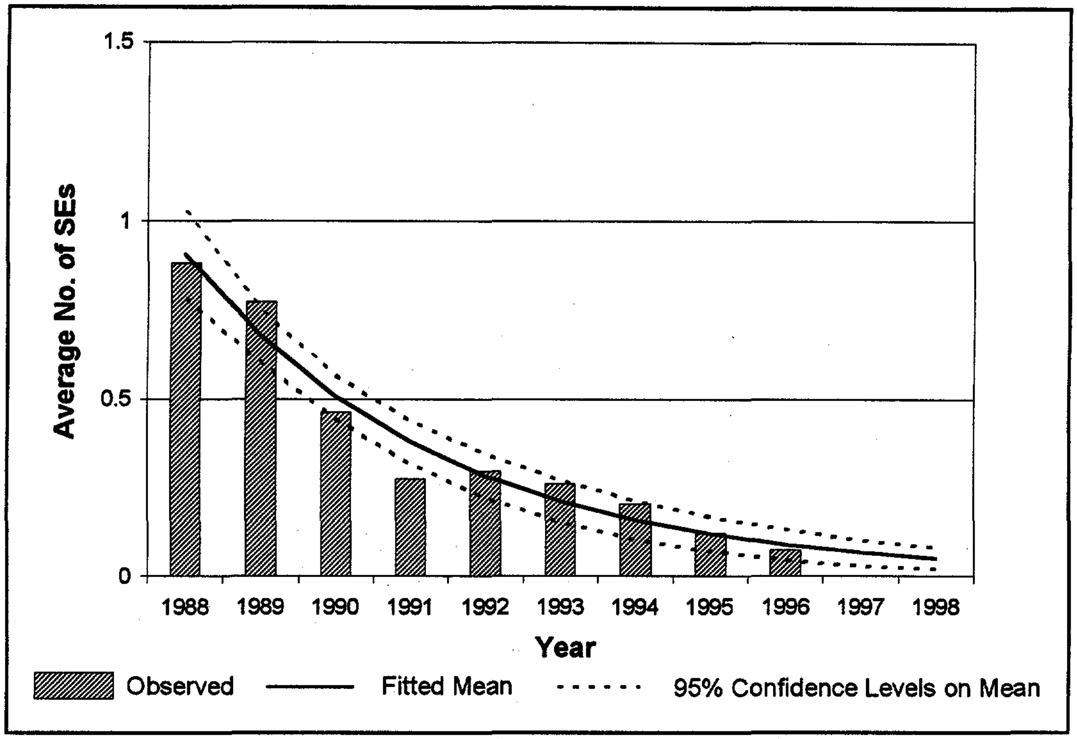


Figure 2.7 Nonlinear Regression Fit for Significant Events ($y = Ae^{Bx}$)

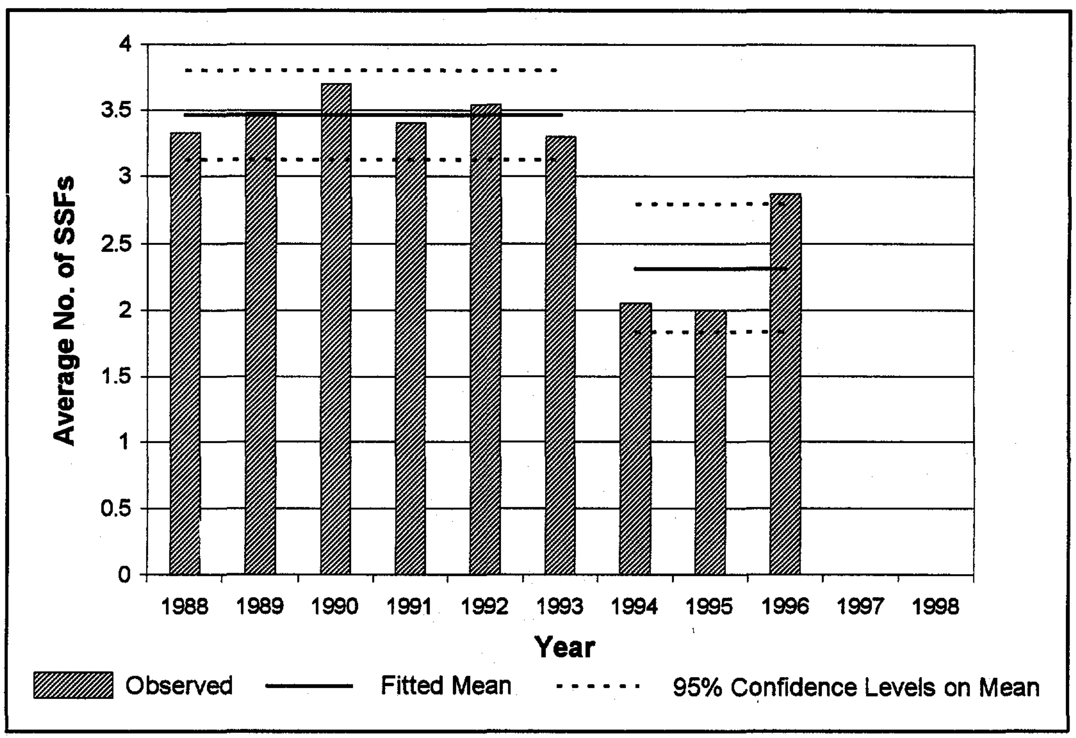


Figure 2.8 Two-Valued Fit for Safety System Failures ($y = A_0$ or A_1)

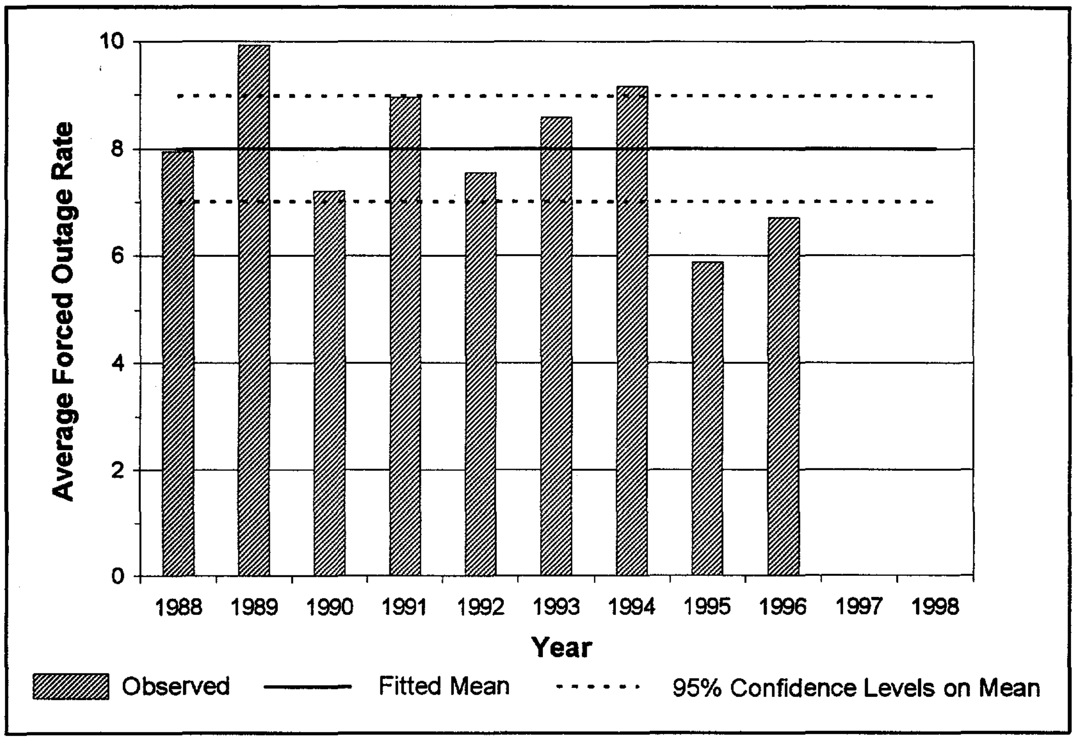


Figure 2.9 Constant Fit for Forced Outage Rate ($y = A$)

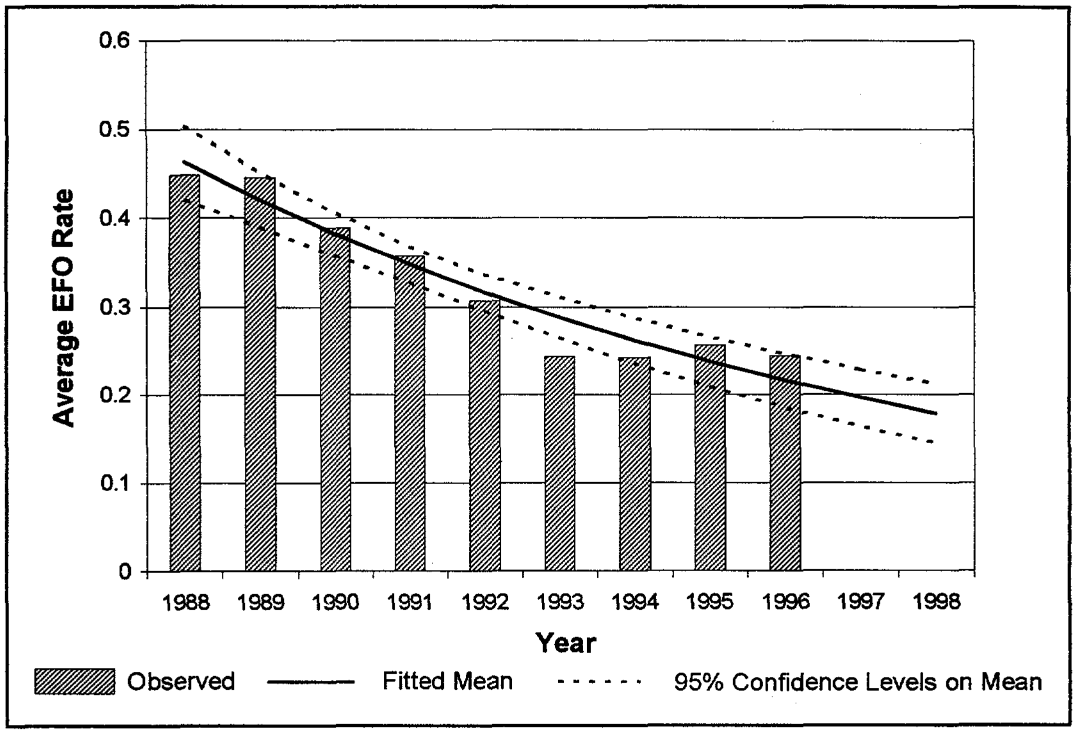


Figure 2.10 Nonlinear Regression Fit for Equipment Forced Outage Rate ($y = Ae^{Bx}$)

2.1.11 Nuclear Reactor Safety Performance

Through the many activities of AEOD, trends in overall safety performance of power reactors may be inferred. The PI and Accident Sequence Precursor Programs of AEOD have been applied to analyze data and information in a consistent manner over a number of years. These programs show a substantial reduction in safety-significant operational events since 1985. The number of initiating events resulting in scrams has declined significantly over the past ten years, and this is reflected in fewer and less complicated plant transients (safety systems actuations and significant events). In 1996 the industry average number of scrams, safety system actuations, and significant events continued to decline slightly. However, equipment problems persist, as evidenced by the percentage of scrams caused by equipment failure (the leading cause of all scrams), and the lack of sustained improvement in safety system failures, forced outage rate, and equipment forced outages per 1000 critical hours. In 1996 safety system failures, forced outage rate, equipment forced outage rate, and collective radiation exposure leveled off or worsened. Although average unit availability has improved considerably over the past 10 years, this has been due not to fewer forced outage hours but to greatly reduced scheduled outage hours. This is a consequence of longer fuel cycles, which result in greater intervals between refueling outages, and of shorter refueling outages. Implementation of the maintenance rule, and the collection and use of equipment reliability and availability data associated with it, should provide a means to reduce the number of safety system failures as well as both the number and duration of forced shutdowns.

2.2 Abnormal Occurrences

Section 208 of the *Energy Reorganization Act* of 1974 defines an abnormal occurrence (AO) as an unscheduled incident or event that NRC determines to be significant from the standpoint of public health or safety. The *Federal Reports Elimination and Sunset Act* of 1995 requires that AOs be reported to Congress annually. Consequently, NRC now publishes the annual AO report on a fiscal year basis.

AEOD identifies AOs using criteria that were initially promulgated in an NRC policy statement that was published in the *Federal Register* on February 24, 1977 (Vol. 42, No. 37, pages 10950-10952). Using these criteria, an event will be considered an AO if it involves a major reduction in the degree of protection of public health and safety. Such an event would involve a moderate or more severe impact on public health and safety and could include, but need not be limited to, (1) moderate exposure to, or release of, radioactive material licensed by or otherwise regulated by the Commission; (2) a major degradation of essential safety-related equipment; or (3) major deficiencies in design, construction, use of, or management controls for licensed facilities or material. This policy statement was published before medical licensees were required to report misadministrations to the NRC, and few of the examples in the policy statement are applicable to medical misadministrations. Therefore, in 1984 NRC adopted additional guidance for reporting medical misadministrations. This guidance was still in effect in 1996 and was used to select events to be included in the 1996 AO report to Congress. On January 27, 1992, new medical misadministration requirements became effective. Consequently the NRC staff developed revised criteria for reporting incidents and events. The revised criteria became effective on November 7, 1996, and relate AOs directly to the requirements in Title 10 of the *Code of Federal Regulations* for protection of public health and safety. The revised criteria will be used to select events to be included in the 1997 AO report to Congress.

The AO report for 1996 (NUREG-0090, Vol. 19) contains two AOs for events at nuclear power plants. One involved a plant trip with multiple complications at Wolf Creek Nuclear Generating Station, and the other involved containment-bypass leakage via disconnected hydrogen-monitor lines at Braidwood Units 1 and 2. These AOs are summarized in Appendix B to this report.

Table 2.2 shows the number of AOs that have occurred at nuclear power plants from CY 1987 to 1996. The number has remained low, averaging just over two per year.

Table 2.2 Abnormal Occurrences per Year at U.S. Nuclear Power Plants

Year	No. of AOs
CY 87	3
CY 88	3
CY 89	4
CY 90	1
CY 91	0
CY 92	3
CY 93	1
CY 94	2*
1995	3*
1996	2

* includes one event from the fourth quarter of CY 94

2.3 Radiation Exposures From Reactors and Nonreactors

2.3.1 Sources of Radiation Exposure

According to the National Council on Radiation Protection and Measurements, the average total effective dose equivalent (TEDE) to a person in the United States is approximately 0.36 centiSieverts (cSv) (360 millirem [mrem]) per year, mostly from natural sources of radiation. The average person in the United States receives a TEDE of about 0.05 cSv (50 mrem) per year from medical applications. The entire fuel cycle, including operation of reactors, contributes less than 0.001 cSv (1 mrem) per year. All other human-controlled sources of radiation combined add up to a TEDE of approximately 0.006 cSv (6 mrem) per year.

The economics of operating a nuclear power plant creates a strong impetus to reduce exposures to plant employees and contractors who work there and to achieve ALARA (as low as reasonably achievable) objectives. As a result, utility violations of NRC limits on personnel exposure are rare, and the vast majority of plant personnel have annual exposures far below the NRC regulatory limits specified in 10 CFR Part 20. The average measurable TEDE per reactor worker has been reduced from 0.94 cSv (940 mrem) per worker in 1973 to 0.31 cSv (310 mrem) per worker in 1995 (the latest year for which data are available). This is believed to result primarily from the licensees' extensive dose-reduction efforts. Some measures that reduce collective exposure are an effective maintenance program, experienced and well-trained personnel, a good water chemistry control program, effective decontamination and cleanup practices, good fuel cladding integrity, effective radiation exposure control programs, good housekeeping, and an alert health physics staff.

2.3.2 Exposures for Reactor and Nonreactor Applications

The NRC regulates both reactor and nonreactor applications of nuclear materials. All NRC licensees are required to monitor employee exposure to radiation and radioactive materials at levels sufficient to demonstrate compliance with the occupational dose limits specified in 10 CFR Part 20. Licensees of power reactors, and those involved in industrial radiography, the manufacture and distribution of radioactive materials, fuel fabrication and processing, low-level radioactive waste disposal, and independent spent fuel storage, are required by 10 CFR 20.2206 to give the NRC annual reports of exposure data for individuals for whom personnel monitoring is required.

Table 2.3 summarizes the information reported by licensees of commercial reactors from 1990 to 1995. For purpose of comparison, 1973 has also been included.

Table 2.3 Annual Occupational Exposure Data for Commercial Reactors for CY 1973 and CY 1990 to CY 1995

Year	No. of Reactors	Collective TEDE (person-cSv [rem])	No. of Workers with Measurable TEDE	Average Measurable TEDE per Worker (cSv [rem])
1973	24	13,962	14,780	0.94
1990	116	36,607	98,802	0.37
1991	115	28,528	91,085	0.31
1992	114	29,298	94,317	0.31
1993	114	26,365	86,187	0.31
1994	109	21,695	73,780	0.29
1995	109	21,674	70,986	0.31

Source: Radiation Exposure Information Report System, funded by the Office of Nuclear Regulatory Research. All reactor data are adjusted to account for multiple counting of transient reactor workers.

Table 2.4 lists the exposure data by licensee category for 1995. For more information on radiation exposures in nuclear materials applications, see the AEOD Annual Report on Nuclear Materials, NUREG-1272, Vol. 10, No. 2. The data in all tables are subject to change as more information becomes available; this may cause minor changes in the data published from year to year.

2.3.3 Comparison of Overexposures for Reactor and Nonreactor Applications

Although commercial reactor occupational exposures have been maintained at a low level, a few overexposures continue to occur. A summary of the number of occupational overexposures in NRC-licensed facilities for reactors and nonreactors for the years CY 1990 through CY 1995 is given in Table 2.5. In every year shown the number of individuals overexposed in nonreactor applications has exceeded the number overexposed at reactor sites. For more information on overexposures in nonreactor applications, see the AEOD Annual Report on Nuclear Materials, NUREG-1272, Vol. 10, No. 2.

The number of overexposures and the number of workers with measurable doses for reactors and NRC-licensed radiographers, the nonreactor licensee category of most concern because of the high rate and magnitude of overexposures, are shown in Table 2.6.

The special radiological problems of industrial radiography have been recognized for some time. The NRC has provided a special guidance and training document, NUREG/CR-0024, "Working Safely in Gamma Radiography," for radiographers for the purpose of reducing overexposures. In addition, AEOD has prepared a videotape on good safety practices in industrial radiography. The tape is entitled, "Taking Control: Safety Procedures for Industrial Radiography," and was released in December 1993.

2.4 Allegations at Commercial Nuclear Power Plants

The NRC receives allegations from individuals or organizations who assert some impropriety or inadequacy in activities regulated by the NRC. Allegations may be received at NRC headquarters or the regional offices. Allegations are entered into the Allegation Management

Table 2.4 Occupational Exposure Data for NRC Licensees for CY 1995

Category	No. of Licensees Reporting	Collective TEDE (person-cSv [rem])	No. of Workers with Measurable TEDE	Average Measurable TEDE per Worker (cSv [rem])
Reactors	109	21,674	70,986	0.31
Industrial Radiography	139	1338	2465	0.54
Manufacture & Distribution	36	595	1222	0.49
Fuel Fabrication & Processing	8	1217	2959	0.41
Low-Level Waste Disposal	2	8	56	0.15
Independent Spent Fuel Storage	1	51	49	1.04

Source: Radiation Exposure Information Report System

Table 2.5 Annual Occupational Overexposures for NRC Licensees for CY 1990 to CY 1995

	1990	1991	1992	1993	1994	1995
Reactors	1	0	5	0	1	0
Industrial Radiography	7	2	1	1	2	1
Medical Facilities	3	2	5	3	0	0
Manufacture & Distribution	0	1	0	5	1	2
Other	1	1	3	3	0	0

Source: Radiation Exposure Information Report System

Note: Occupational overexposures exclude exposures to the general public and to patients in excess of those prescribed for medical procedures.

Table 2.6 Annual Occupational Overexposure Rate at NRC Reactor and Radiography Licensees for CY 1990 to CY 1995

0	Reactors			Radiography		
Year	No. of Workers with Measurable TEDE	No. of Workers Overexposed	Over-exposures per 1,000 Workers	No. of Workers with Measurable TEDE	No. of Workers Overexposed	Over-exposures per 1,000 Workers
1990	98,802	1	0.01	4,458	7	1.57
1991	91,085	0	0.00	4,649	2	0.43
1992	94,317	5	0.05	4,265	1	0.23
1993	86,187	0	0.00	3,007	1	0.33
1994	73,780	1	0.01	2,351	2	0.85
1995	70,986	0	0.00	2,465	1	0.41

Source: Radiation Exposure Information Report System

System (AMS), which is managed by the Office of Nuclear Reactor Regulation (NRR). NRR and regional staff jointly collect the allegations, determine their validity, and track their resolution. AEOD analyzes trends in the numbers of allegations received from each nuclear plant site and publishes the data in such a manner as to not reveal the identity of the alleger. Table A-2.13 of Appendix A-2 provides the number of allegations that were received from each site, those that remain open, those that have been substantiated in any manner, and those that contain harassment and intimidation concerns. Caution should be used in interpreting the table because the existing AMS database does not provide a definitive breakdown between fully and partially substantiated allegations, no differentiation is made in the data between allegations having varying levels of safety significance, and each allegation may contain one or many individual concerns. The AMS database structure has been recently enhanced to improve its capability to track and analyze allegations.

3 AEOD Reliability and Risk Activities

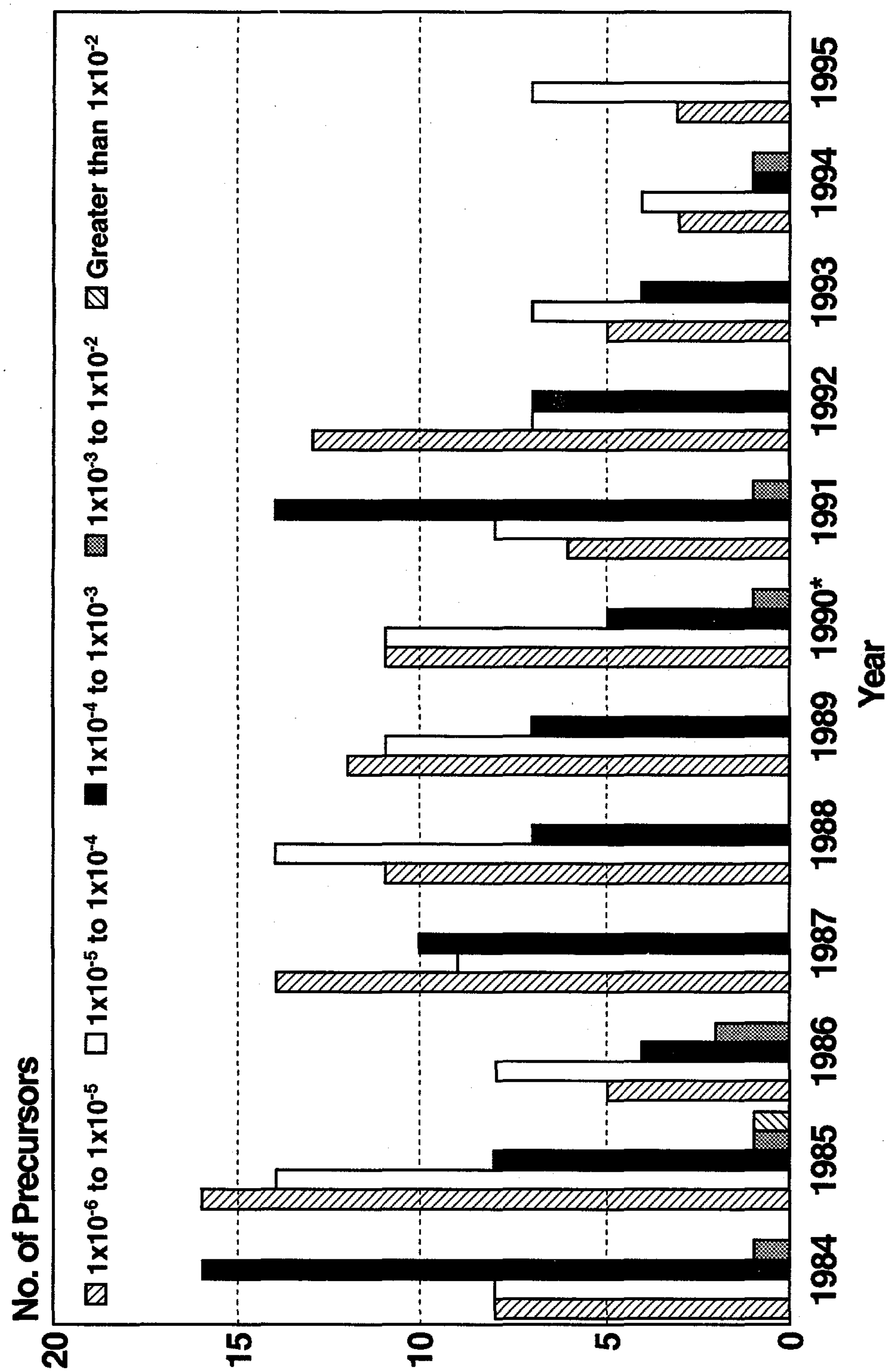
3.1 Accident Sequence Precursor Program

The Accident Sequence Precursor (ASP) Program uses probabilistic risk assessment techniques to evaluate the conditional core damage probabilities of nuclear power plant events and equipment unavailabilities. The purpose of the program is to provide a structured and systematic means of quantitatively evaluating the safety significance of nuclear plant operating experience. The principal objectives of the program are to identify and rank the risk significance of operating reactor events, to determine their generic implications, to characterize risk insights, and to document and disseminate the evaluations for feedback to plant operators to promote learning from experience. ASP Program results are published annually in the NUREG/CR-4674 series.

An Accident Sequence Precursor is an operational event or plant condition that is an important element of a postulated core-damaging accident sequence. Accident sequences considered in the ASP Program are those associated with inadequate core cooling, which would be expected to result in core damage. Precursors can be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition involving inadequate core cooling. The ASP methodology evaluates disparate elements of operational experience by assuming random failures for other branches of the event tree models. These evaluations account for all actual or potential concurrent failures, degradations, or outages of safety systems. The evaluations also include estimates of the likelihood of equipment failures and human errors and of the probability of recovery should they occur. The figure of merit for ASP analyses is conditional core damage probability (CCDP) for initiating events and increase in core damage probability (ΔCDP) for conditions and equipment unavailabilities. Events with CCDPs or ΔCDPs greater than 1.0×10^{-6} are considered Accident Sequence Precursors. Figure 3.1 shows the distribution of ASP events for U.S. nuclear power plants for CY 1984 through CY 1995.

The ASP Program began in 1979. Since then the staff has evaluated and documented over 490 precursors from reported experience for CY 1969 through CY 1995 (final results are also available for two precursors for CY 1996). Over the years, the ASP Program has evolved to the point where the methodology and results are now used routinely by the NRC. The methodology continues to be improved to better account for plant design and operational differences, human reliability, and changes in equipment, and to provide user-friendly analytical tools. Other planned improvements include incorporation of modeling and data uncertainty in each event analysis, a more complete set of accident sequences, and better containment response and consequence evaluation.

To identify potential precursors, the staff reviews licensee event reports (LERs) or other documentation (e.g., inspection reports, Incident Investigation Team reports) of plant problems, equipment failures, or other operational incidents. Event trees model plant responses to challenges such as transients, loss-of-coolant accidents (LOCAs), loss of offsite power (LOOP) events, steam generator tube ruptures, and anticipated transients without scram. Operational occurrences that involve portions of these postulated core damage sequences are identified. Plant equipment and human responses that could affect the progression of an accident are evaluated, including actual failures that have occurred and the probability that other postulated failures could occur. Fault tree linking techniques are used to provide a quantitative estimate of the significance of the reported data.



*The 3/20/90 Vogtle Event has been rounded up from 9.7×10^{-4} and plotted as 1×10^{-3}.

Figure 3.1 Yearly Distribution of ASP Conditional Core Damage Probabilities

The results of the ASP analyses are considered indications of the level of risk associated with operating nuclear power plants based on direct assessment of actual operating experience. The precursor events from the ASP Program comprise a unique database of historical system failures, multiple losses of redundancy, and infrequent core damage initiators. Several of the recorded precursor events involved equipment failure caused by factors, conditions, or phenomena that affected the ability of safety equipment to perform its function. These mechanistic failures are different from "random" failures or unavailabilities of equipment.

Commercial nuclear power reactors in the United States now have a combined total of over 2000 years of operating experience. The ASP Program uses information gained from this experience to provide an ongoing assessment of nuclear plant operation. This assessment helps to identify how well plant designs and capabilities can cope with actual operational events or conditions.

3.1.1 Results for CY 1995

The results of the ASP analyses of CY 1995 events are documented in NUREG/CR-4674, Vol. 23, dated April 1997, and are shown in Table 3.1. Ten events or conditions occurred in CY 1995 which resulted in ten precursors (ten units were affected). This is fewer than in previous years, and the reduction is due in part to specific mitigating equipment and recovery measures that were not previously credited. The preliminary ASP analyses were reviewed by the NRC staff and the affected licensees. They were also independently reviewed by the Sandia National Laboratories under contract to the NRC. On the basis of comments received from the reviewers, the analyses were revised to help the NRC provide more accurate risk assessments of the events.

Of the ten precursors for CY 1995, eight involved discovered conditions or unavailabilities of equipment and two involved initiating events. All except one of them occurred at PWRs. Six of the precursors involved problems with electrical systems, although none involved a total LOOP. This is consistent with the results from the previous five calendar years, for which about 60 percent of the precursor events involved electric power issues.

3.1.2 Results for CY 1996

Table 3.2 presents the results of precursor analyses of two CY 1996 operational events. Both of these events were important precursors. The Catawba 2 LOOP while an emergency diesel generator was out of service for maintenance had a CCDP greater than 1.0×10^{-3}, which was higher than the CCDP of any of the CY 1995 precursor events. The degradation of the essential service water (ESW) system at Wolf Creek caused by the formation of frazil ice under severely cold weather conditions would have resulted in the loss of all decay heat removal capability if the only remaining operable ESW pump had failed. In addition, with the turbine-driven auxiliary feedwater pump out of service for maintenance when the event occurred, the plant was vulnerable to a LOOP.

3.1.3 Analysis of CY 1982-1983 Events

The review and analysis of CY 1982 and CY 1983 events for precursors began in October 1994 to obtain the two years of precursor data that had previously been missing. More than 10,000 LERs were systematically screened for potential precursors and 435 were identified for further analysis. As a result of this analysis, 109 precursors were identified, almost equally distributed between the two years. The final report was published in NUREG/CR-4674, Vol. 24, dated April 1997.

Table 3.1 Accident Sequence Precursors for CY 1995

Precursors Involving an Initiator

Plant	LER No.	Date	CCDP	Description
Comanche Peak 1	445/95-003, -004	10/11/95	2.9×10^{-5}	Reactor trip, auxiliary feedwater (AFW) pump trip, second AFW pump unavailable
Arkansas Nuclear One, Unit 1	313/95-005	04/20/95	2.0×10^{-5}	Reactor trip with emergency feedwater (EFW) problems

Precursors Involving Equipment Unavailabilities

Plant	LER No.	Date	ΔCDP	Description
St. Lucie 1	335/95-004, -005, -006	08/02/95	9.3×10^{-5}	Failed power-operated relief valves, multiple reactor coolant pump seal stage failures, relief valve failure, shutdown cooling unavailable and other problems
Millstone 2	336/95-002,	01/25/95	3.1×10^{-5}	Containment sump isolation valves susceptible to pressure locking
Waterford 3	382/95-002	06/10/95	1.7×10^{-5}	Reactor trip and fire in turbine-generator building
St. Lucie 2	389/95-005	11/20/95	1.3×10^{-5}	Failure of one emergency diesel generator (EDG) with common-cause failure implications
Arkansas Nuclear One, Unit 2	368/95-005	07/19/95	1.1×10^{-5}	Loss of dc bus could fail both EFW trains
D. C. Cook 1	315/95-011,	09/12/95	7.7×10^{-6}	One safety injection pump unavailable for six months
Limerick 1	352/95-008	09/11/95	9.0×10^{-6}	Safety/relief valve fails open, scram, suppression pool strainer fails
Haddam Neck	213/95-010	03/09/95	4.7×10^{-6}	Multiple safety injection valves susceptible to pressure locking during a large break loss-of-coolant accident

Table 3.2 Accident Sequence Precursors for CY 1996

At-Power Precursors Involving an Initiator

Plant	LER No.	Date	CCDP	Description
Catawba 2	414/96-018	02/06/96	2.1×10^{-3}	LOOP with emergency diesel generator B unavailable
Wolf Creek	582/96-001, -002	01/30/96	2.1×10^{-4}	Reactor trip and loss of train A of essential service water with the turbine-driven auxiliary feedwater pump unavailable

3.1.4 Evaluation of ASP Results and Trends

A paper documenting an analysis of trends in the annual occurrence rates of accident sequence precursors that occurred from CY 1984 through CY 1994 was prepared and presented at the Probabilistic Safety Assessment '96 Conference at Park City, Utah, in October 1996. The annual rates for all precursors and for precursors in each CCDP range bin were analyzed and compared. The analysis showed similarly decreasing rates for all CCDP probability bins less than 1.0×10^{-3}. On average, precursors with CCDP greater than or equal to 1.0×10^{-3} appear to be occurring every one to two years.

3.2 System Reliability Studies

AEOD uses operational data to determine the reliability of risk-significant systems in U.S. commercial reactors. The data are obtained from LERs, special reports, and monthly operating reports. Each study covers the period from CY 1987 through CY 1993. Three of them have been completed. A report on the reliability of the high-pressure coolant injection (HPCI) system in the 23 boiling water reactors (BWRs) with HPCI systems was completed in 1995. The results of this study are summarized in NUREG-1272, Vol. 9, No. 1. Studies of the reliability of the emergency diesel generator (EDG) power system in all plants with EDGs, and of the isolation condenser (IC) system at the five BWRS with that system, were completed in 1996 and the results are summarized below. Table 3.3 summarizes the studies completed to date.

Draft reports for the reactor core isolation cooling system and the high pressure core spray system in BWRs were completed this year. Future studies include the auxiliary/emergency feedwater systems at PWRs, the low pressure injection systems at both BWRs and PWRs, and the high pressure safety injection system at PWRs. AEOD is also developing and applying simplified models of the various reactor protection systems for both PWRs and BWRs to estimate their reliability based on actual operating experience.

3.2.1 Emergency Diesel Generator Power System Reliability

This report presents an evaluation of the performance of the EDG trains at all sites with EDGs. Because inconsistences exist in the information available between plants reporting per Regulatory Guide 1.108 (RG-1.108) and those that do not, the report focuses primarily on plants reporting per RG-1.108, with limited analyses and comparisons for non-RG-1.108 plants.

Table 3.3 System Performance Summary

System	Unreliability	Unplanned Demand Trend	Failure Rate Trend	Unreliability Trend	Consistency with PRAs/IPEs	Unreliability vs. Plant Age	Demand vs. Test Failure Differences
HPCI	0.06	Decreasing	Decreasing	Steady	General agreement with exceptions requiring investigation	None	Yes
EDG	0.04	Decreasing	Decreasing	Steady	General agreement except actual data better after 8 hours	None	Yes
IC	0.02	Steady	Steady	Steady	General agreement but different contributors	None	No

The mean unreliability was 0.044 for the population of plants reporting per RG-1.108, assuming an 8 hour mission time and including recovery probabilities from failures not requiring repair. Consistent with AEOD's study of the HPCI system, the overall unreliability remained fairly constant over the 7 year study period, even though the rates of unplanned demands and failures were steadily decreasing (see Figure 3.2). Plants not reporting per RG-1.108 appear to have similar unreliabilities based on the limited data available for comparison.

Failures to start and maintenance-out-of-service (MOOS) while at power were the dominant contributors to EDG train unreliability for the plants reporting per RG-1.108, with the MOOS contribution accounting for 70 percent of the unreliability. The failures to start were primarily caused by failures in a variety of electrical components which were not easily recovered by simple operator actions. MOOS while shut down was about 10 times higher than during power operation. This is an important consideration for shutdown risk studies.

Demand reliability (i.e., failure to start and failure to run) was consistent with the station blackout rule for the plants reporting per RG-1.108. However, the MOOS unreliability observed during this study period was four times as high as the value originally calculated in support of the station blackout rule (0.030 versus 0.007). The average failure-to-start unreliability, including recovery, was 0.01 and the average failure-to-run unreliability was 0.004. These data indicate that the population of diesel generators is achieving a demand reliability (excluding MOOS events) of over 98 percent. The higher MOOS contribution is a reflection of increased maintenance activity during plant operation. While increased time in maintenance or testing adversely impacts the total train reliability, the system effect is more limited due to the importance of common-cause

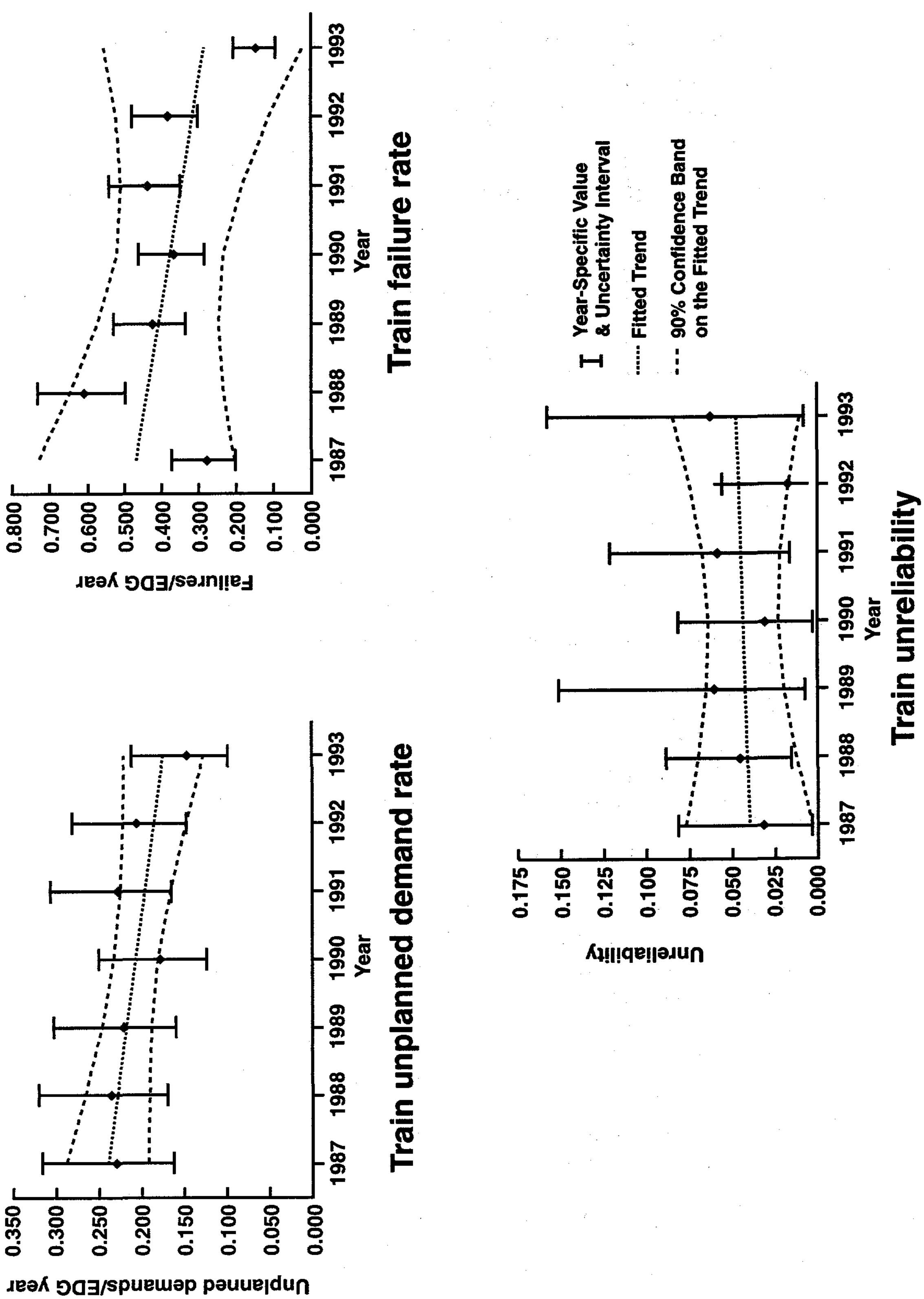


Figure 3.2 EDG Train Trends

failure probabilities when two or more redundant trains are considered.

No common-cause failures of multiple EDG trains were observed during the unplanned demands reported by the RG-1.108 plants. Based on our current understanding of common-cause failure probabilities, no common-cause failure events would have been expected for the number of unplanned demands that occurred in the study period. In the larger population of test demands, some common-cause failure events did occur and these events are discussed in the report.

Three distinct failure-to-run rates were observed from the data reported per RG-1.108, associated with different failure mechanisms occurring at different run times. The failure-to-run rate for the longer mission times (greater than 14 hours) was one-hundredth of that for the shortest times (less than 30 minutes). The observed mean unreliability was generally comparable with the values used in PRAs and Individual Plant Examinations (IPEs) with mission times under 8 hours. However, study results indicate that PRA/IPEs may be overestimating the contribution of failure-to-run events for longer mission times.

No correlation was found between the low power license date and the plant-specific unreliability for the plants reporting per RG-1.108. However, the plants licensed from 1980 to 1990 did experience higher failure rates than the plants licensed earlier. There was insufficient information from the data to determine the reason for this difference but it was observed that most of the failures experienced by the 1980-to-1990 plants occurred during the first 2 years of operation.

The overall nature of the failures experienced by the plants reporting per RG-1.108 during actual demands differed somewhat from those discovered during monthly surveillance tests, engineering and design reviews, and routine inspections. This indicates that the current testing and inspection activities may not be focusing on the dominant contributors to unreliability during actual demands and may need to be modified to better factor in the conditions and experiences gained from actual system demands.

3.2.2 Isolation Condenser System Performance

For the five U.S. BWRs that have the IC system, the best estimate of train unreliability (including recovery), based on operational experience data, is 0.02. The failure-to-operate failure mode of the IC train and the failure to provide makeup water to the isolation condenser contributed equally to the overall unreliability. The recovered and unrecovered train unreliability estimates differ by a factor of five. The difference is primarily attributable to IC train failures to operate due to spurious isolations that occurred during unplanned demands.

The average of the estimates of IC train unreliability based on information contained in PRA/IPEs was generally about a factor of 1.5 lower than the estimate of the mean probability based on operational experience data. All of the PRA/IPE estimates of IC train unreliability are within the uncertainty interval based on operational experience data. The average of the PRA/IPE values of IC train unreliability is approximately 1.3×10^{-2} per demand.

The PRA/IPEs show that the condensate isolation valve failing to open is the important contributor to IC train unavailability. However, this contrasts with the calculations based on operational data, which show that the effect of this type of failure is not as important to IC train unreliability as the spurious isolations of the IC train. Figure 3.3 shows the train unreliabilities and comparisons to the PRA/IPEs.

The probability of MOOS was not estimated in this report. The operating experience is sparse and a lack of MOOS failures (i.e., no failures in 23 demands) does not support postulating this particular failure mode at this time. Based on

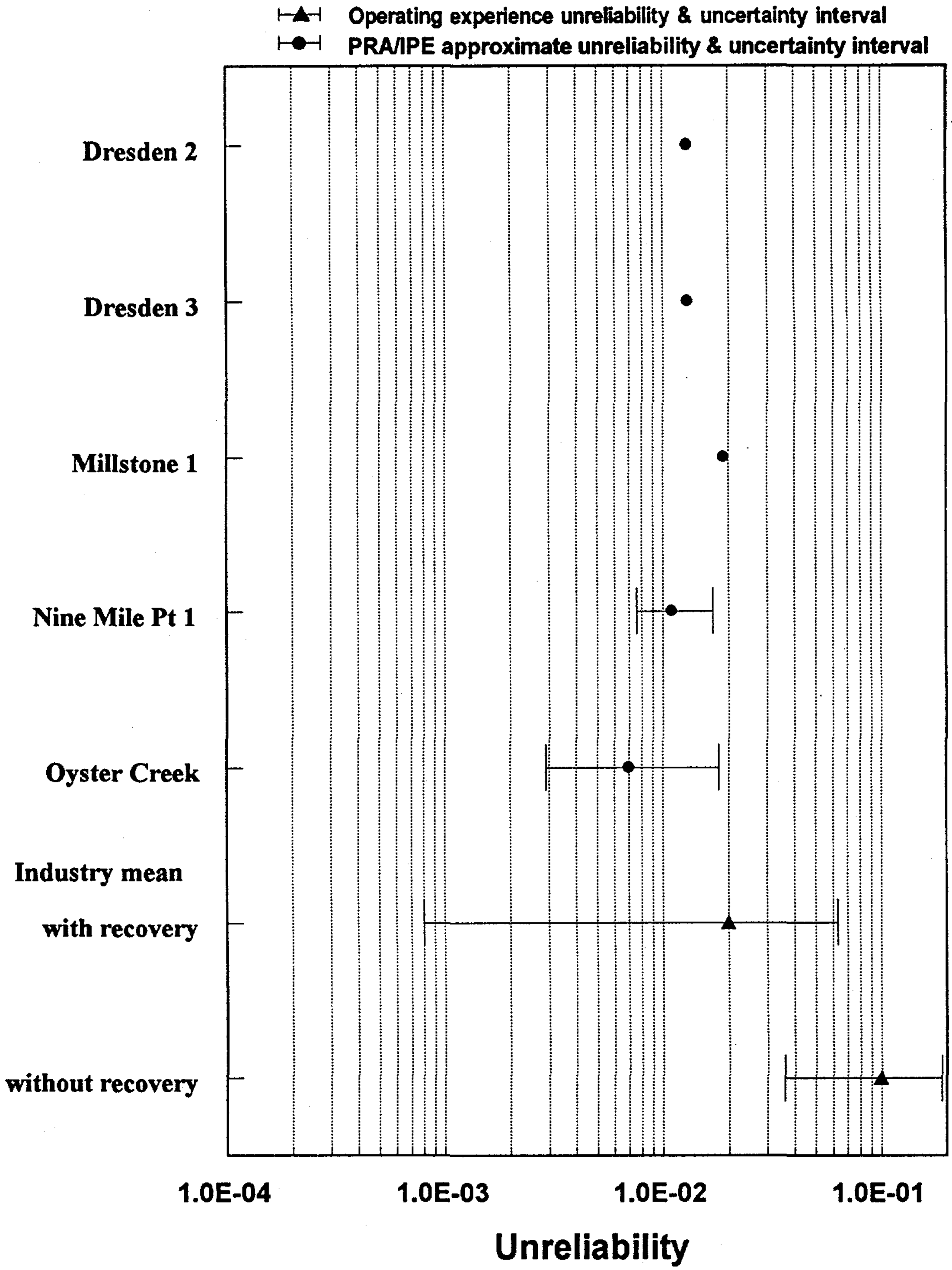


Figure 3.3 Isolation Condenser Unreliabilities

PRA/IPE information, maintenance accounts for approximately 5 per cent of the total unreliability of the IC train.

No statistically significant trends in IC train failure and unplanned demand frequencies or unreliability by calendar year were observed in the operational experience data. Further, IC train unreliability was analyzed against low-power license date for the plants to determine if unreliability was affected by plant age. No such trends were observed.

3.3 Common-Cause Failure Database

AEOD has compiled common-cause failure (CCF) events from LERs and data records contained in the Nuclear Plant Reliability Data System (NPRDS). These events are maintained in a CCF database which, along with its associated technical documentation (6 volumes), was completed in December 1995. The database contains CCF events for over 40 combinations of risk-important systems and components using data from about 1980 through 1995. Technical review of the database and draft technical reports has been completed.

This database represents the most complete collection of CCF events in the world. Because NPRDS plant-specific records are proprietary, the database is also proprietary. AEOD expects to have a limited distribution of the database in 1997. Details of this distribution are being coordinated with INPO.

4 AEOD REPORTS

In 1996 the AEOD staff continued to analyze and evaluate operating experience and publish studies of equipment problems and events as well as analyses of the reliability of important safety systems. Probabilistic risk assessment (PRA) and reliability analyses continue to be applied to a greater range of event studies.

The staff reviewed a broad spectrum of operating experience data, including reports submitted to the NRC by licensees in compliance with 10 CFR 50.72 and 10 CFR 50.73, the database of component failures in the Nuclear Plant Reliability Data System (NPRDS) maintained by the Institute of Nuclear Power Operations, and reports of foreign reactor events. On the basis of the staff's review and analysis of these data, AEOD in 1996 issued seven special studies, four engineering evaluations, and five technical reviews. Appendix C lists the reports issued in 1996, while Appendix D lists those issued from CY 1980 through 1995. Sections 4.2 through 4.4 below summarize the 1996 reports, which are categorized as follows:

- Case studies involve in-depth analyses of significant safety issues and document the bases for AEOD recommendations for regulatory or industry actions. Each case study goes through a rigorous peer review process to ensure technical adequacy.
- Special studies document accelerated investigations and suggest or recommend regulatory actions that are to be completed expeditiously.
- Engineering evaluations document assessments of significant operating events and suggest remedial actions, if appropriate.
- Technical reviews document AEOD studies of issues that the staff concludes have little safety significance, typically concluding that the licensees' or industry's planned or scheduled corrective actions are adequate.

Reports of AEOD studies of operational experience are broadly disseminated. The AEOD staff continued efforts to more effectively communicate the lessons of operating experience through a variety of other forums, including participation in industry code committees, presentation of papers at professional meetings, and attendance at owners groups and international meetings.

4.1 AEOD Activities To Identify and Address Safety Issues

AEOD uses a systematic process to nominate, prioritize, and select safety issues to be studied. Six attributes are considered: risk significance, issue complexity, requirement factors, review factors, industry initiatives, and other considerations. Information is extracted from various databases, including the NRC Sequence Coding and Search System, the Incident Reporting System of the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (see Section 10.1), the NPRDS, the NRC Allegation Management System (AMS), and NRC generic communications. Based on the assembled information, each topic is rated in each attribute. This approach strengthens AEOD's independent means of identifying and studying generic lessons learned from operating experience.

4.2 Special Studies

4.2.1 Office for Analysis and Evaluation of Operational Data Annual Report, 1994 - FY 95

NUREG-1272, Vol. 9, No. 1

NUREG-1272, Vol. 9, was published as a combined calendar year 1994 and fiscal year 1995 report that describes activities conducted between January 1, 1994, and September 30, 1995. NUREG-1272, Vol. 9, comprises three parts. Vol. 9, No. 1, covering power reactors, presents operational data and summarizes the important operating experience of the nuclear power industry from the NRC's perspective. NUREG-1272, Vol. 9, No. 2, covers nuclear materials and presents a review of the events and concerns associated with the use of licensed material in nonpower reactor applications. NUREG-1272, Vol. 9, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission.

4.2.2 Precursors to Potential Severe Core Damage Accidents: 1994 A Status Report

NUREG/CR-4674, Vols. 21 and 22

See Section 3.1 of this volume.

4.2.3 Performance Indicators for Operating Commercial Nuclear Power Reactors, Data Through September 1995

See Section 2.1 of this volume.

4.2.4 Evidence of Aging Effects on Certain Safety-Related Components

NUREG/CR-6442

A generic study was conducted on the effects of aging of active components in nuclear power plants in response to interest shown by the NEA's Principal Working Group Number 1 (PWG-1) of the Committee on the Safety of Nuclear Installations. The study was limited to active components since these are within the mandate of PWG-1. (Passive components are in the mandate of PWG-3.) Representatives from France, Sweden, Finland, Japan, the U.S. and the United Kingdom (U.K.) participated in the study by submitting reports documenting aging studies performed in their countries. This NUREG/CR consists of a discussion of the general theory of aging, summaries of the reports submitted by the participating countries, and a comparison of the various statistical analysis methods used in the studies. Statistical analysis of failure and maintenance data is an important part of efforts to identify potential aging problems and to focus maintenance activities to mitigate those problems.

A summary of the U.K. studies concluded that no major phenomena of degradation caused by aging have been observed at the U.K. nuclear power plants studied and that degradation phenomena, when present, are of a minor nature and do not appear to affect the reliability of plant systems. The Japanese study concluded that their preventative maintenance programs are effective in mitigating potential aging problems, resulting in a lower failure rate at older plants than at newer plants. A U.S. study of overall plant performance found no deleterious age effects. The study concluded that maintenance strategies are effective in managing the effects of aging and that design and licensing criteria produce an inherently rugged plant.

On the basis of this study, the PWG-1 reached a tentative consensus that, with some exceptions, active components generally do not present a significant aging problem in nuclear power plants where maintenance and modification strategies are effective. Design criteria and effective preventive maintenance programs, including timely replacement of components, are effective in mitigating potential aging problems. However, aging studies (such as qualitative and statistical analyses of failure modes and maintenance data) are an important part of efforts to identify and solve potential aging problems. Solving these problems typically includes such strategies as replacing suspect components with improved components and implementing improved maintenance programs.

4.2.5 Isolation Condenser System Reliability, 1987 - 1993

AEOD/S96-01

See Section 3.2.2 of this volume.

4.2.6 Assessment of Spent Fuel Cooling

AEOD/S96-02

As a result of questions that had been raised about the adequacy of spent fuel pools (SFPs), the Executive Director for Operations requested that AEOD perform an independent study of the likelihood and consequences of an extended loss of SFP cooling. AEOD staff conducted an extensive review of more than 12 years of domestic and foreign operating experience data; visited six nuclear sites (with nine nuclear power plants) and the headquarters of Pennsylvania Power and Light, the operator of the Susquehanna Steam Electric Station (SSES); and met with contract engineers who had submitted a 10 CFR Part 21 report about potential defects and noncompliances at SSES. The staff reviewed previous SFP risk assessments and contracted with the Idaho National Engineering and Environmental Laboratory to perform a limited PRA of the SSES SFP. AEOD also performed independent assessments of the electrical systems, instrumentation, heat loads and radiation levels associated with the SFPs.

On the basis of the study findings, the staff concluded that loss of SFP coolant inventory greater than 1 foot has occurred at a rate of about 1 per 100 reactor years, and loss of SFP cooling with a temperature rise greater than 20°F has occurred at a rate of approximately 3 per 1000 reactor years. The consequences of these actual events have not been severe. However, events have occurred that have resulted in the loss of several feet of SFP coolant level and have lasted more than 24 hours. The primary cause of these events has been human error. Both the likelihood and the consequences of loss of SFP cooling events are highly dependent on human performance as well as individual plant design features. From their review of existing SFP risk assessments, the staff found that the relative risk from the loss of spent fuel cooling is low compared to the risk from events involving active fuel in the reactor vessel.

As a result of this study, the staff has determined that the typical U.S. plant may need improvements in SFP instrumentation, operator procedures and training, and/or configuration control. The need for specific corrective actions should be evaluated for those plants where failures of reactor cavity or gate seals or ineffective antisiphon devices could potentially cause sufficient loss of SFP coolant inventory to uncover the fuel or endanger makeup capability. The need for improving configuration controls related to the SFP to prevent and/or mitigate SFP loss of inventory events and loss of cooling events should be evaluated on a plant-specific basis. The need for plant modifications at some multi-unit sites to account for the potential effects of SFP boiling conditions on safe shutdown equipment for the operating unit, particularly during full core off-loads, should be evaluated on a plant-specific basis. The need for improved procedures and training for control room operators to respond to SFP loss of cooling events consistent with the time frames over which events can proceed, recognizing the heat load and the possibility of loss of inventory, should be evaluated on a plant-specific basis. The need for improvements to instrumentation and power supplies to the SFP equipment to aid operator response to SFP events should be evaluated on a plant-specific basis.

4.2.7 Emergency Diesel Generator Power System Reliability, 1987 - 1993

AEOD/S96-03

See Section 3.2.1 of this volume.

4.3 Engineering Evaluations

4.3.1 Motor-Operated Valve Key Failures

AEOD/E96-01

AEOD identified a significant number of motor-operated valve (MOV) failures that involved anti-rotation keys, valve operator-to-valve stem keys, and motor pinion gear keys. Many of the key failures were not detected during surveillance tests but were found upon valve demand, during valve operations, or during maintenance activities, and had existed for some time before they were discovered. These were significant deficiencies and represent a potential common-cause failure mechanism for the associated safety-related systems.

The dominant cause of anti-rotation and valve operator-to-valve stem key failures was attributed to key loosening associated with setscrew loosening or improper staking during installation of the keys. The major contributors to motor pinion gear key failures were high impact loads and improper material. This failure mechanism appears to involve keys made from American Iron and Steel Institute (AISI) 1018 material in high speed and high inertia configurations. The fix was to replace the keys with keys made from harder AISI 4140 material, which may, in some cases, lead to keyway deformation or damage, depending on impact loads and shaft material. This situation may present a complex stress problem that is not completely considered in the design and that could lead to cracking and failure of the shaft. Consequently, the possibility of a failure in the motor pinion-to-shaft connection may not be eliminated by key replacement.

This report describes the importance of plant maintenance programs to verify that MOV keys are staked and secured as required, the importance of plant MOV surveillance and maintenance activities to ensure the early detection of key degradation, and the possibility of shaft cracking as a result of replacement of 1018 keys with harder material when the replacement will involve a relatively soft shaft and high impact loads. To address performance problems with MOVs, including the key failures described in the study, the NRC issued Information Notice (IN) 96-48, "Motor-Operated Valve Performance Issues," in August 1996.

4.3.2 Analysis of Allegation Data

AEOD/E96-02

The NRC manages allegations concerning NRC regulated activities in such a way as to encourage individuals to identify technical, safety, and/or harassment and intimidation concerns. Allegations submitted to the NRC (except for 10 CFR 2.206 petitions and allegations of wrongdoing by NRC employees) are entered into the AMS database. AEOD staff analyzed allegations from CY 90 through CY 94 associated with power plant sites and their vendors, contractors, and consultants to identify those organizations with a disproportionate number of allegations. The organizations that stood out were Watts Bar, Millstone, and Burns Security. Other potential candidates for further NRC review were Cooper, Clinton, Fermi, Maine Yankee, Braidwood, LaSalle, McGuire, Salem/Hope Creek, and Zion.

The report includes cautions regarding inferences drawn from the number of allegations associated with a given facility. It may be that several complainants have prepared one allegation, or that several different concerns have been identified in a single allegation. The number of allegations at any facility may also be influenced by a variety of factors such as the number of real safety issues, the perceived freedom to raise safety issues, experience and familiarity with the allegation process, concerns about job security, poor worker-management relationships, and the economic environment. It should also be kept in mind that the number of allegations that are substantiated at most facilities is small.

4.3.3 Analysis of Allegation Data Supplement 1

AEOD/E96-02, Supplement 1

This supplement is an update of the study documented in AEOD/E96-02 using AMS data from CY 91 through CY 95. The organizations that stood out were Watts Bar, Millstone, and San Onofre. Other potential candidates for further NRC review were Waterford, Riverbend, Washington Nuclear, Maine Yankee, St. Lucie, Sequoyah, Salem/Hope Creek, and Westinghouse.

4.3.4 Steam Generator Tube Failures

NUREG/CR-6365
(AEOD/E96-03)

This report presents a review and summary of the available information on steam generator tube failures and the impact of these failures on plant safety. The sources of information included technical reports by the NRC, the Electric Power Research Institute, various nuclear steam system suppliers, utilities, and U.S. National Laboratories; NRC Bulletins, Notices, and Generic Letters; work-shops and conferences; media publications such as *Nucleonics Week*; the Nuclear Power Experience database; and technical journals. Discussions with technical experts were, in some cases, the only available source of information on certain subjects. The evaluation covered steam generator tube degradation in pressurized water reactors, Canadian deuterium-uranium reactors, and Russian water-moderated and -cooled reactors.

Spontaneous tube ruptures have occurred at the rate of about one every 2 years over the last 20 years, and incipient tube ruptures (tube failures usually identified with leak detection monitors just before rupture) have been occurring at the rate of about one per year. These ruptures have caused complex plant transients that have not always been easy for the reactor operators to control.

Analysis shows that if more than 15 tubes rupture during a main steam line break, the system response could lead to core melt. Although spontaneous and induced steam generator tube ruptures are small contributors to the total core damage frequency calculated in PRAs, tube ruptures are risk significant because the resultant radionuclides are likely to bypass the reactor containment building.

The frequency of steam generator tube ruptures can be significantly reduced through appropriate and timely inspections and repairs or by removal from service. However, what constitutes appropriate and timely inspections and what level of degradation requires removal from service are continuing issues. Also, the most widely used inspection equipment is not able to detect and size all steam generator tube degradations of concern. Many different approaches to solve these problems have been used throughout the world.

4.4 Technical Reviews

4.4.1 Potential Damage to Low-Pressure Injection Valves During Surveillance Testing

AEOD/T95-02

The NRC issued Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," in response to AEOD's publication of NUREG-1275, Volume 9 (AEOD/S92-07). These documents addressed the potential inoperability of certain MOVs under specific conditions. Subsequently, the NRC issued IN 95-30, "Susceptibility of Low-Pressure Coolant Injection and Core Spray Injection Valves to Pressure Locking." The event described in the IN illustrated another mechanism for possible valve damage.

The staff found that some valves susceptible to pressure locking could also be subject to a different loading mechanism which could develop during routine surveillance tests (required by regulation). This mechanism by itself may cause inoperability or potentially damage the MOV so that it would not operate during a design basis event. In addition, the corrective action to prevent pressure locking will not protect the MOV from possible damage caused by this different mechanism. The report includes a suggestion to resolve this conflict.

4.4.2 Review of the National Transportation Safety Board's Safety Study NTSB/SS-94/01, "A Review of Flightcrew-Involved, Major Accidents of U.S. Carriers, 1978 Through 1990"

AEOD/T95-03

The staff performed an assessment of the relevance to the nuclear industry of the National Transportation Safety Board's (NTSB's) Safety Study. The staff reviewed all incident investigation team reports, the 1992 AEOD human performance case study (and supporting event reports, as needed), and CY 1993 through CY 1995 augmented inspection team reports. Events investigated by the NTSB were major accidents with injury and loss-of-life, whereas the nuclear power plant events were precursor events with no radiological consequences although, in some cases, an economic impact occurred because of lost power generation. Each of the nuclear events was reviewed for situations analogous to such relevant NTSB findings as (1) checklist implementation errors, (2) tactical decision errors, (3) insufficient monitoring/challenging, (4) high risk evolutions, (5) production pressure, (6) hours awake, and (7) junior operator in crew. Several nuclear power plant events with similarities to NTSB findings were identified and discussed in the report.

4.4.3 AEOD Technical Reports by Category

AEOD/T96-01, Revision 1

This technical review is an update of a similar effort performed 3 years ago. The updated tables are included in the AEOD Annual Report as Appendix E. The revision contains a modified report abstract and cross reference from an AEOD report to a NUREG series. The 500 technical reports issued since 1980 have been sorted by various categories from components, to systems, to processes.

4.4.4 Target Rock Two-Stage SRV Performance Update

AEOD/T96-02

The NRC has issued several INs related to the Target Rock two-stage safety/relief valves' (SRVs) failure to open at the expected setpoint. The cause of the problem has been identified as one or both of the following: (1) binding in the labyrinth seal area caused by tolerance buildup during manufacturing or (2) disc-to-seat bonding caused by oxides of the disc and seat material forming a continuous film and inhibiting disc movement. Some time after 1990, the Target Rock Two-Stage Owners Group submitted two potential solutions. The preferred solution was to use a platinum alloy disc to cause oxygen and hydrogen in the valve to recombine, thereby reducing the oxygen available to cause corrosion. The alternate solution was the installation of a pressure switch and control circuitry to operate the valve electrically when the set pressure of the pressure transmitter was reached.

The corrosion bonding issue is being addressed by the Owners Group and monitored by the NRC. The updated information on overall setpoint testing shows a gradual increase in the average liftpoint of the SRVs. This report also provides information currently available regarding setpoint testing results on platinate discs; at least one full set of setpoint testing results is necessary to evaluate the test program. The pressure switch fix has NRC's approval for the topical report, however plant-specific submittals are required for licensees to implement it. Should the platinate disc solution not work, the pressure switch fix is ready for immediate use.

On September 11, 1995, an event occurred at Limerick Unit 1 in which pilot disc leakage resulted in a stuck-open Target Rock two-stage SRV and an extended reactor blowdown into the suppression pool. Sludge and fibers in the pool, roiled by the blowdown, landed on and obstructed the emergency core cooling system strainers nearest the SRV tailpipe. The NRC issued IN 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage"; IN 95-47, Rev. 1; and Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," to address issues resulting from this event.

4.4.5 Response of Babcock & Wilcox Company Plants Following a Loss of Nonemergency AC Power

AEOD/T96-03

AEOD reviewed the operating experience of Babcock & Wilcox (B&W) Company nuclear plants to characterize their response to a loss of nonemergency ac power and to determine the impact of that response on the Oconee emergency power system. The concern was that a complete loss of nonemergency ac power at Oconee could cause reactor trips, emergency core cooling system actuations, and automatic start of emergency feedwater at all three units which could result in overcooling of the three reactors and overloading of the emergency power system. A review of the operating experience at B&W nuclear plants from 1987 to 1996 found that complete or partial losses of nonemergency ac power did not result in any full emergency core cooling system actuations or overcooling transients. However, four out of five losses of nonemergency ac power did result in the loss of feedwater or a feedwater transient and the automatic start of the emergency feedwater system from the emergency power supply. These issues are currently under review by NRC staff.

5 OPERATING EXPERIENCE DATA

5.1 Licensee Event Reporting

The primary source of information about an operational event is the licensee event report (LER) submitted as required by 10 CFR 50.73. Safety performance is only one of several factors that affects the number of LERs submitted by a licensee. Therefore, the NRC staff does not base its assessment of safety performance of a plant on the number of LERs that have been submitted. Rather, judgments about safety performance are based on an evaluation of the significance of operational events. For completeness, however, we have included Table 5.1, which shows the total number of LERs (excluding supplementary, canceled, proprietary, voluntary, and safeguards LERs) submitted each year since CY 87 by commercial nuclear power reactor licensees. The overall decrease in the number of LERs from CY 87 to CY 95 appears to be associated with the reduction in initiating events, such as scrams and ESF actuations, and changes to the reporting requirements. Table 5.2 shows the percentage of LERs submitted in accordance with specific sections of 10 CFR 50.73.

Table 5.1 LERs Submitted by Year*

Year	No. of LERs	No. of Units	LERs per Unit
CY 87	2895	111	26
CY 88	2479	110	23
CY 89	2356	112	21
CY 90	2128	111	19
CY 91	1858	111	17
CY 92	1774	111	16
CY 93	1400	109	13
CY 94	1279	109	12
CY 95	1178	109	11
CY 96	1368	109	13

* Counts do not include Dresden Unit 1; Humboldt Bay Unit 3; Three Mile Island Unit 2; Fort St. Vrain after August 29, 1989; LaCrosse after April 30, 1987; Rancho Seco after June 7, 1989; Shoreham after June 6, 1987; Yankee Rowe after February 26, 1992; San Onofre Unit 1 after November 30, 1992; and Trojan after January 4, 1993. Supplemental, canceled, proprietary, voluntary, and safeguards LERs were excluded from all counts.

Table 5.2 Percentage of LERs Submitted in CY 1996 by 10 CFR 50.73 Requirement

10 CFR Section	Requirement	Percentage of LERs
50.73(a)(2)(i)	Technical Specification shutdown or violation	51
50.73(a)(2)(iv)	Engineered safety feature actuation (including reactor trip)	21
50.73(a)(2)(v)	Real/potential safety system loss	12
50.73(a)(2)(ii)	Unanalyzed condition	22
50.73(a)(2)(vii)	Failures in multiple systems	4
50.73(a)(2)(x)	Internal threat	<1
50.73(a)(2)(iii)	External threat	<1

5.2 U.S. Operational Experience Databases

AEOD uses the Sequence Coding and Search System (SCSS) for storing and retrieving LER information. This system, developed in the early eighties and maintained under contract with Oak Ridge National Laboratory, contains an average of 150 items of information for each of the nearly 43,000 LERs submitted since 1980. The LER descriptive text is coded into computer-searchable sequences, with each sequence identified by categories such as components, systems, personnel errors, causes, and corrective actions. Coding the LER in sequences facilitates searches. The SCSS, given a series of failures or errors for an event or event type, can identify previous similar events to support trend analyses.

The SCSS database is the primary source of operating experience information for AEOD studies and for the NRC Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research, and for the regions. In 1996 the AEOD staff also continued to use the LER information from the SCSS database to support certain other NRC activities, such as operating experience reports to support inspections and senior management meetings.

In addition to the SCSS, AEOD also maintains data on LERs, monthly operating reports, and plant outages at the Idaho National Engineering Laboratory to support the NRC's Performance Indicator (PI) Program. This PI database contains plant-specific information on reactor scrams, safety system actuations, safety system failures, forced outage rate, and equipment forced outages per 1000 commercial critical hours (see Section 2.1 of this volume for a detailed discussion of the 1996 PIs). In addition, AEOD uses these databases to prepare special studies, evaluations of selected plants, and briefing packages for Commission site visits.

Since the early 1980s, AEOD has used the Nuclear Plant Reliability Data System (NPRDS), a proprietary database containing approximately 630,000 component engineering records and nearly 164,000 component failure records from commercial nuclear power plants in support of studies of operating experience. Nuclear plant operators provided the data to the Institute of Nuclear Power Operations (INPO) which managed and directed the development of the database. In 1997 INPO will replace NPRDS with a new database, the Equipment Performance and Information Exchange (EPIX) System for reporting component failures. NPRDS records will be archived and available to users through EPIX.

5.3 Reliability and Availability Data

On February 12, 1995, the Commission published a proposed rule, 10 CFR 50.76, "Reporting Reliability and Availability Information for Risk-Significant Systems and Equipment" (61 FR 5318). Reliability and availability information is needed to substantially improve the NRC's ability to make risk-effective regulatory decisions. This is part of a more general move towards risk-informed regulatory approaches that provide a means for the NRC to maintain, and in some cases improve, safety while at the same time reducing impacts on licensees and NRC resource expenditures by focusing regulatory activities on the most risk-significant areas. The draft regulatory guide associated with the proposed rule was published on May 2, 1996 (61 FR 19645). The comment periods for the proposed rule and the draft guide closed on June 11, 1996, and July 5, 1996, respectively. Industry has proposed a voluntary alternative to the rule, and in October 1996 INPO provided a sample of voluntary data from its Safety System Performance Indicator data base. As of the end of 1996, the staff was evaluating the feasibility of the proposed approach as an alternative to rulemaking.[1]

[1]In 1997 the staff recommended and the Commission approved the voluntary alternative to the proposed rule.

6 INCIDENT RESPONSE

AEOD maintains and implements the NRC's Incident Response Program with the support of other headquarters and regional offices. This program includes the receipt of data and reports for both emergency and non-emergency events from licensees, followed by an appropriate NRC response. The response for the more serious emergencies is through an incident response organization that includes representatives from several headquarters offices and the affected regional office. The NRC's response program also includes coordination with other Federal agencies as well as State and local governments.

6.1 NRC Operations Center

The NRC Operations Center, located at Two White Flint North in Rockville, Maryland, provides the focal point for NRC communications with its licensees, State agencies, and other Federal agencies about events that occur in the commercial nuclear sector. It is continuously staffed by a Headquarters Operations Officer who is a nuclear systems engineer trained to receive, evaluate, and respond to all types of events. The Operations Center features a state-of-the-art information management system that integrates voice, video, and data subsystems to provide the timely and effective flow of information during the NRC's response to an incident.

6.2 Emergency Response

NRC-licensed facilities have a variety of Emergency Plan requirements. Both production and utilization facilities (power and non-power reactors) are required to maintain plans for responding to emergencies that could impact the health and safety of the public. Facilities or activities that are licensed for the possession and utilization of byproduct material, source material, or special nuclear material are required to maintain Emergency Plans for responding to a radiological release only if these licensees possess quantities of nuclear material that exceed the amounts specified in 10 CFR Parts 30, 40, and 70. In addition, all NRC-certified gaseous diffusion plants are required to maintain Emergency Plans. The requirements for independent spent fuel storage installations located on the sites of NRC-licensed nuclear power reactors are satisfied by the Emergency Plans required for these sites. Other facilities or activities that the NRC licenses to possess or utilize nuclear materials are not required by the Code of Federal Regulations to maintain Emergency Plans; however, these facilities or activities may be required to maintain Emergency Plans in accordance with their NRC licenses.

NRC-licensed facilities also have various classes of emergencies. Both power and nonpower reactor licensees utilize the following four emergency classes, in order of increasing severity:

- <u>Notification of Unusual Event (NOUE)</u> - a condition involving potential degradation of the level of plant safety that does not represent an immediate threat to public health and safety.

- <u>Alert</u> - a condition involving actual or potential substantial degradation of the level of plant safety where any offsite radiological releases are expected to be limited to small fractions of the Environmental Protection Agency protective action guideline exposure levels.

- <u>Site Area Emergency</u> - a condition involving actual or likely major failures of one or more plant functions required for protection of the public or involving conditions with potential for a significant offsite radiological release but where a core melt situation is not indicated.

- <u>General Emergency</u> - a condition involving actual or imminent substantial core degradation or melting with potential for loss of containment.

Emergencies for nuclear materials licensees are classified into one of the following two levels in order of increasing severity:

- Alert - for an NRC-licensed nuclear materials facility, this indicates that events may occur, are in progress, or have occurred that could lead to a release of radioactive material but that the release is not expected to require a response by offsite response organizations to protect individuals offsite.

- Site Area Emergency - for NRC nuclear materials licensees, this indicates that events may occur, are in progress, or have occurred that could lead to a significant release of radioactive material and that could require a response by offsite organizations to protect individuals offsite.

Although not required by the Code of Federal Regulations, some nuclear materials licensees may also utilize the NOUE emergency classification for events with lower safety significance.

In the event of an emergency at an NRC-licensed facility (or associated with an NRC-licensed activity), the licensee will place an emergency telephone call to the NRC Operations Center immediately after notifying appropriate State and local agencies. For Alert and higher declarations, and for events for which an NRC response may be appropriate, the Regional Administrator and an Executive Team member (typically the Director of the Office of Nuclear Reactor Regulation for reactor events) will be added to the discussion of the event in a conference call with the licensee.

The NRC's response to an event may range from routine follow-up to a complete activation of both the regional Incident Response Center and the NRC headquarters Operations Center. The NRC utilizes the following formal modes for responding to events at its licensed facilities.

For the **Normal Mode**, the lowest level of response, the NRC will not fully staff the headquarters Operations Center or the regional Incident Response Center, but it may take some other action such as sending out a special inspection team or staffing the response centers with a few select experts to monitor the event. The latter is referred to as the Monitoring Phase of the Normal Mode.

Standby Mode, the next level of response, is entered when an event is judged to be sufficiently uncertain or complex that the situation needs to be continuously monitored from the headquarters and regional response centers by teams of experts. During Standby Mode, the NRC response is led from the headquarters Operations Center.

If the event threatens public health and safety, the NRC will enter the **Initial Activation Mode**. Upon entering this mode, the NRC will promptly send a team from the regional office to the site to lead the NRC response. Until the Site Team is in place, the NRC response will be led from the headquarters Operations Center. Within the Operations Center, teams of specialists will evaluate the status of critical safety functions and will independently evaluate protective actions recommended by the licensee for implementation by State and local authorities. All communications with the media, State and Federal officials, Congress, and the White House will also be coordinated from the NRC Operations Center.

Once the NRC site team arrives on the scene and is prepared to accept the authority and responsibility for the Federal response, the NRC enters the **Expanded Activation Mode**. The Director of Site Operations, typically the Regional Administrator, will report to the licensee's Emergency Operations Facility near the site or the Technical Support Center at the site. The lead responsibility for performing assessments of reactor safety and protective measures then shifts from headquarters to the NRC team at the site. The headquarters Operations Center will then provide logistical and technical support to the NRC Site Team as necessary.

6.3 Operations Center Data for 1996

In addition to emergency event notifications, the NRC Operations Center receives many notifications of events that do not meet the threshold for emergency classification. Actions taken by the Headquarters Operations Officer in response to such notifications range from computer and log entries followed by appropriate notifications to establishing emergency conference calls between licensee representatives and senior NRC regional and headquarters representatives. For very significant events, conference calls may result in the activation of the agency's Incident Response Plan.

Table 6.1 shows the total number of events reported to the NRC Operations Center during 1996. These notifications were primarily received from nuclear power plant licensees. A small subset of these notifications involved events classified by licensees into one of the four emergency classes.

Table 6.2 shows the number of each type of emergency event reported annually from CY 89 through 1996. The number of NOUEs reported to the Operations Center has decreased by 66 percent since CY 89. This can be partially attributed to the fact that many licensees have implemented revised procedures for emergency action levels that better reflect the severity of events.

Table 6.3 lists the emergency events reported by power reactor facilities to the NRC Operations Center during 1996 that were categorized at the Alert level. (There were no power reactor events reported at a level higher than Alert.) The NRC entered the Monitoring Phase of the Normal Mode for two of the five Alerts reported. The NRC also entered the Monitoring Phase of Normal Mode for the following four NOUEs:

- Farley Units 1 and 2 - Hurricane Opal
- Wolf Creek - the buildup of frazil ice in the intake structure and the inoperability of the essential service water system
- Catawba Unit 2 - loss-of-offsite power with one emergency diesel generator inoperable
- Brunswick Units 1 and 2 - Hurricane Bertha

6.4 Emergency Exercises

Emergency exercises are held periodically to ensure that the NRC, the licensee, local, State, and other Federal response organizations are proficient in dealing with each type of emergency. Preparation for these exercises includes the development of a postulated accident scenario that usually goes well beyond the facility's design basis and that results in the release of some radioactivity outside the facility's boundary. NRC experts in reactor safety and protective measures follow the progression of the simulated event; communicate with the licensee, State, and Federal responders; and provide recommendations to an NRC Executive Team in the NRC Operations Center.

During 1996 the NRC headquarters and regional offices participated in full scale emergency exercises with the following nuclear power plants: Ginna on December 5, 1995; Duane Arnold on April 9, 1996; Maine Yankee on June 19, 1996; and Arkansas Nuclear One on August 14, 1996. The NRC's primary role in participating in these exercises is to provide an independent assessment of licensee actions, assist the licensee when requested, review the protective action recommendations that the licensee makes to State and local authorities, and facilitate communications between the licensee and other response organizations.

Table 6.1 Events Reported to the NRC Operations Center in 1996

Event Type	Power Reactor	Fuel Facility	Non-Power Reactor	Hospital	Transport/ Materials	Well Logging/ Other	Total
Non-Emergency	1,345	10	2	60	100	82	1,599
Unusual Event	65	1	0	0	1	0	67
Alert	5	5	0	0	0	0	10
Site Area Emergency	0	1	0	0	0	0	1
General Emergency	0	0	0	0	0	0	0
Totals	1,415	17	2	60	101	82	1,677

Table 6.2 Classification of Events Under Licensee Emergency Plans From CY 1989 to 1996

	1989	1990	1991	1992	1993	1994	1995	1996
Unusual Event	197	151	170	135	103	97	66	67
Alert	13	10	9	20	8	4	8	10
Site Area Emergency	0	1	2	1	1	0	0	1
General Emergency	0	0	0	0	0	0	0	0

Table 6.3 Alerts Reported at Power Reactor Facilities in 1996

Name	Event No.	Date	Description	Duration	Response
Salem 1 (W/PWR)	29421	10/04/95	Loss of control room annunciators for greater than 15 minutes (Alert was declared on 10/05/95)	3 hrs 44 mins	N/A
LaSalle 1 (GE/BWR)	29529	10/31/95	High radiation levels in containment due to withdrawal of a traversing incore probe to an unshielded location in the Reactor Building	5 hrs 25 mins	Monitoring
Palo Verde 2 (CE/BWR)	30236	04/04/96	Fire in a control room lighting panel	51 mins	N/A
Quad Cities 1/2	30450	05/10/96	Potential damage to plant due to tornado	10 hrs 34 mins	Monitoring
Clinton 1 (GE/BWR)	30894	08/19/96	Fire on reactor core isolation cooling pump turbine insulation Event was declared and immediately terminated)	0 mins	N/A

Limited participation exercises are also conducted under the State Outreach program objectives. These objectives include more frequent participation in exercises with State organizations. During 1996 limited exercises were conducted with the following facilities: Millstone on October 1, 1995; Beaver Valley on February 27, 1996; Catawba on March 12, 1996; River Bend (Site Team only) on April 17, 1996; Fermi on July 16, 1996; and Seabrook (Site Team Only) on September 18, 1996.

In addition to these full-scale and limited emergency exercises, AEOD initiated a new exercise element this year. A small team of experts (Ingestion Team, or I Team) participated in the following ingestion exercises at the following facilities during 1996: Catawba on March 13, 1996; Duane Arnold on April 10, 1996; and Arkansas Nuclear One on August 15, 1996.

Figures 6.1 through 6.4 show participants in a typical exercise as they receive and evaluate the emergency situation, facility status, and licensee actions to determine the appropriate NRC response, including the appropriate guidance to offer State and local governments.

6.5 State Outreach

During 1996 AEOD continued an aggressive State Outreach Program designed to increase

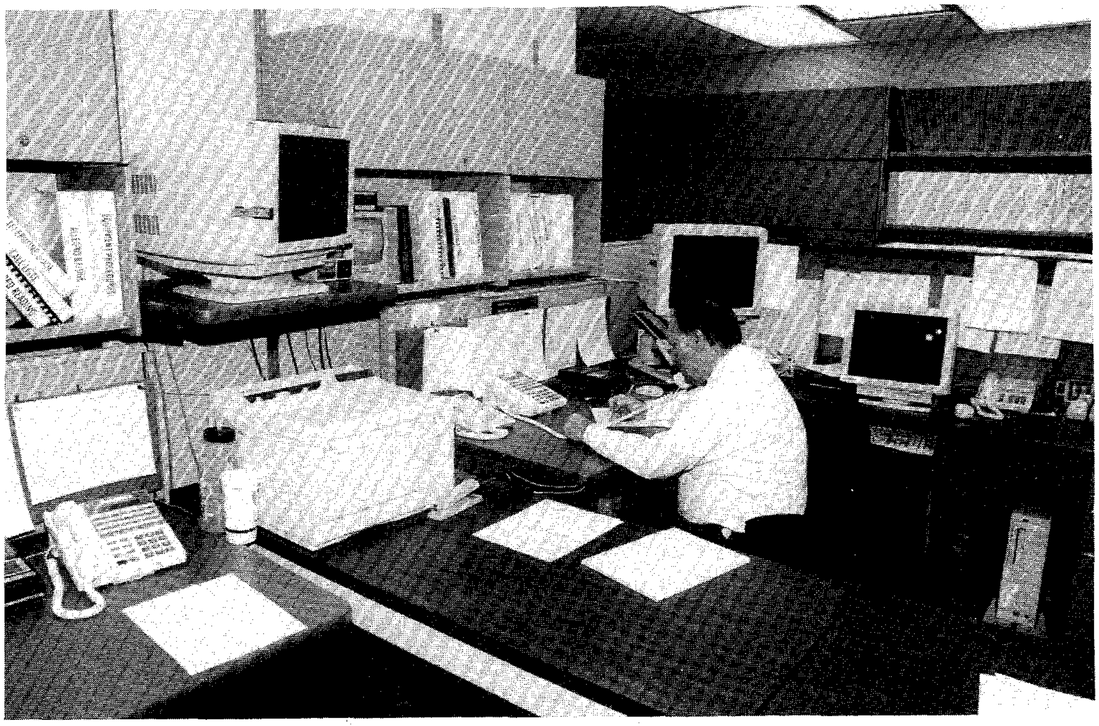


Figure 6.1 A Headquarters Operations Officer Receives an Event Notification Report

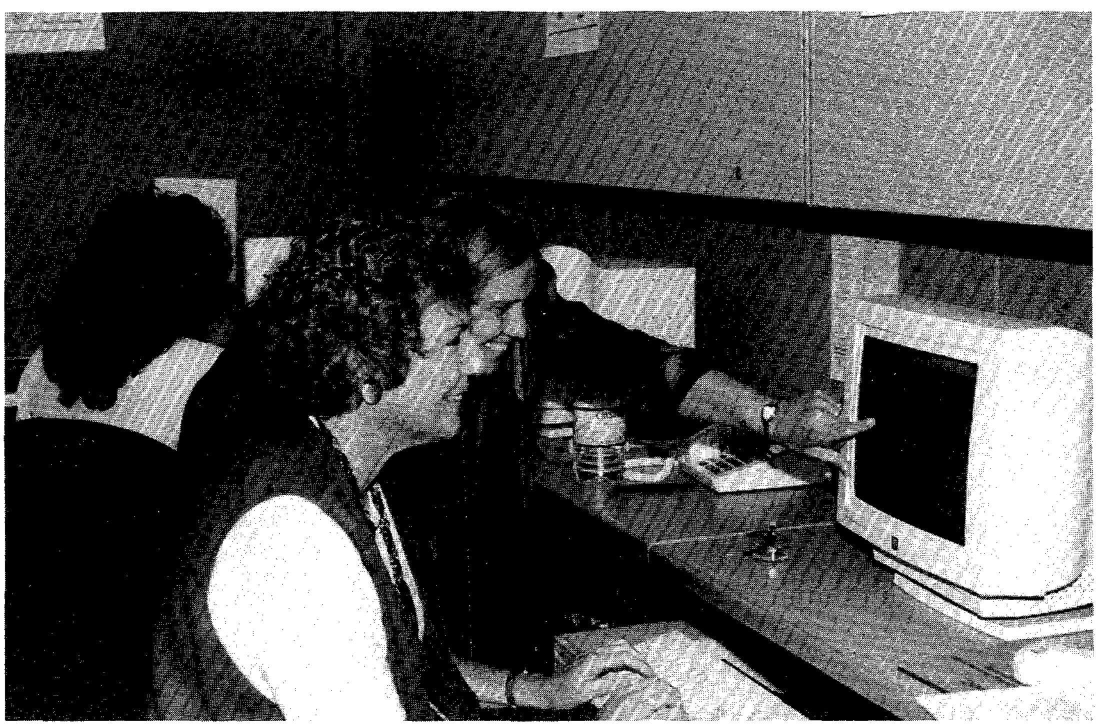


Figure 6.2 Operations Support Team Members Distribute Information Electronically

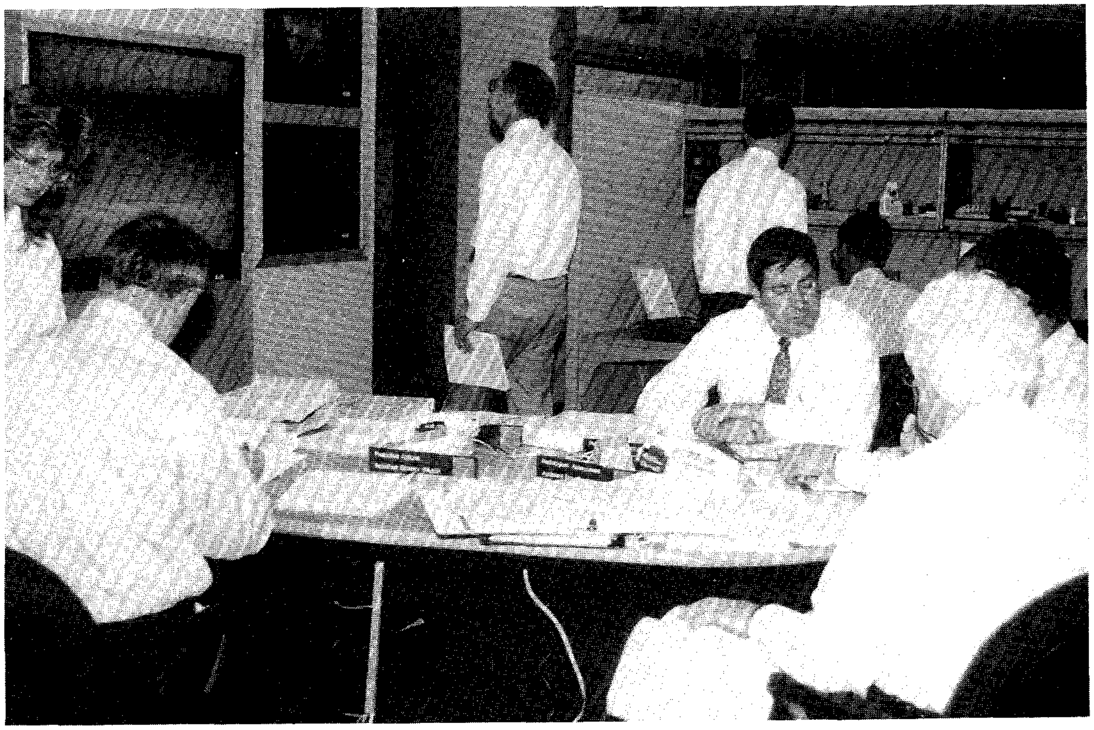


Figure 6.3 The Protective Measures Team Independently Evaluates the Need for Sheltering and Evacuation

Figure 6.4 The Executive Team Being Briefed During an Emergency Exercise

and improve the NRC's interaction with States during events and exercises. The program included briefings of State officials on the NRC and Federal Emergency Response Program, the Emergency Response Data System (ERDS - a real-time data system designed to provide direct transmission of selected nuclear power plant information from licensees' onsite computers to the NRC Operations Center), NRC/State liaison during an emergency, and financial assistance available to responders.

This year, AEOD also expanded the program to include training on the recently published NUREG/BR-0230, "Response Coordination Manual (RCM-96)," and the recently updated NUREG/BR-0150, "Response Technical Manual (RTM-96)." Outreach sessions were conducted with 15 states and numerous licensees. RCM/RTM training was sponsored by Florida Power Corporation for the State of Florida and the utility; by the State of Louisiana for Louisiana, Mississippi, Texas, and Arkansas; and by the Sate of Nebraska for Nebraska, Kansas, Iowa, and Missouri.

Memoranda of Understanding were negotiated to make ERDS available to the States of Louisiana, Wisconsin, and Iowa during 1996. The State of Vermont has requested ERDS, and negotiations are in progress. Twenty-four of the thirty-two States with nuclear power plants have requested access to ERDS.

6.6 Coordination with Other Federal Agencies

During 1996 AEOD continued to participate with other Federal agencies in the issuance of the Federal Radiological Emergency Response Plan (FRERP). AEOD also participated in drafting the Radiological Incident Annex to the Federal Response Plan (FRP), which describes how the FRP and the FRERP are integrated when both plans are used during an emergency. In addition, AEOD participated in an intensive activity to evaluate the adequacy of the Federal plans in response to nuclear, biological, and chemical terrorist events. AEOD continued to train Regional Federal agency representatives on the Federal role in response to a radiological emergency using the Concept of Operations described in the FRERP. Regional Federal responders have also been incorporated in the State Outreach Program events to increase awareness of NRC response methods and to encourage integrated training and planning. AEOD also provided training to selected Congressional staff members on our role during a radiological emergency.

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 materials activities licensed by the NRC. Under the IIP, the NRC responds to an operational event according to its safety significance. For an event of extraordinary safety significance, the Commission may establish an Accident Review Group (ARG), led by an individual from outside the NRC and composed of experts from within and outside the NRC. The ARG reports directly to the Commission and is independent of NRC management. For an event of potentially major safety significance, the Executive Director for Operations (EDO) establishes an Incident Investigation Team (IIT) to investigate the event. 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 follow-up actions can be taken. The EDO assigns staff actions arising from IITs, while the Regional Administrators identify needed actions from AIT findings. AEOD independently reviews AIT reports to provide additional assurance that potential generic lessons are learned and communicated to the industry. For reactor events, the Director of the Office of Nuclear Reactor Regulation (NRR) is responsible for reviewing AIT reports for generic safety implications, initiating follow-up actions, and tracking issues affecting more than one plant, as appropriate. Thus industry-wide safety is enhanced by including the significant lessons learned from AITs with those from engineering studies and reviews of operating experience in generic communications to licensees. As described in NUREG-1303, "Incident Investigation Manual," AEOD has overall responsibility for administration of the IIP, while NRR is responsible for maintaining the procedures for an AIT response.

7.1 Incident Investigation Teams

There were no power reactor events in 1996 that were judged to have a level of significance sufficiently high to warrant an IIT investigation. The status of actions associated with previous IIT findings assigned by the EDO to various NRC offices is documented in Appendix G.

7.2 Augmented Inspection Teams

Seven AITs were established in 1996 to investigate significant incidents at power reactor facilities, as shown in Table 7.1. These AITs helped to improve safety at the affected plants by providing detailed investigations of the problems experienced and identifying their root causes. Lessons learned and communicated to licensees from events investigated by AITs during 1996 include the following:

- On January 30, 1996, after a manual scram from 80 percent power, five control rod assemblies at Wolf Creek failed to fully insert. Two rods remained at 6 steps withdrawn, two at 12 steps, and one at 18 steps. Three of the affected rods drifted to the fully inserted position within 20 minutes, one within 60 minutes, and the last one within 78 minutes. After the scram, the licensee initiated emergency boration, as required when all rods do not fully insert. The five rods were all in 17x17 VANTAGE 5H fuel with burnup greater than 47,000 megawatt-days per metric ton. Westinghouse, the Westinghouse Owners Group, and the licensee pursued the root cause of this event. Possible root causes include the following: debris (foreign matter), corrosion products, control rod or drive line degradation, thimble tube bow, fuel assembly bow and/or twist, reduction in thimble tube

Table 7.1 Reactor Incidents for Which AITs Were Established in 1996

Event Date	Plant	Event
01/30/96	Wolf Creek 1	Control rods failed to fully insert
05/15/96	Dresden	Loss of feedwater flow
05/19/96	Arkansas Nuclear 1	Reactor trip with steam generator dry-out
05/29/96	Point Beach 1, 2	Unanticipated hydrogen gas ignition
06/24/96	LaSalle 1, 2	Foreign material in the intake structure
09/01/96	Haddam Neck	Inadvertent introduction of nitrogen gas into the reactor coolant system during shutdown
09/24/96	Oconee	Scram with complications, rupture of main steam line

diameter, adverse alignment of guide tube cards, and/or design tolerances. As a result of the AIT investigation, the NRC issued Information Notice (IN) 96-12, "Control Rod Insertion Problems," on February 15, 1996.

- On May 28, 1996, a hydrogen gas ignition occurred at the Point Beach Nuclear Station during the welding of the shield lid on a ventilated storage cask (VSC-24) multi-assembly sealed basket (MSB). The gas ignition displaced the shield lid (about 2898 kilograms [6,390 pounds]), leaving the lid in place but tipped at a slight angle, with one edge about 7.6 centimeters (3 inches) higher than normal. The VSC-24 multi-assembly transfer cask (MTC), a shielded lifting device used to transfer the MSB loaded with spent fuel to the ventilated concrete cask, had been placed in the cask decontamination work area in the auxiliary building. Approximately 114 liters (30 gallons) of spent fuel pool water had been drained from the MSB to facilitate welding of the shield lid, creating an air space below the lid. The hydrogen gas ignition occurred during the initiation of the shield lid welding, approximately 11 hours after the loaded MTC had been removed from the spent fuel storage pool. As a result of the AIT investigation, the NRC issued IN 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket," on May 31, 1996.

7.3 Incident Investigation Team Training

The purpose of the IIT Training program is to provide prospective IIT members with comprehensive guidance and methodology for conducting systematic and technically sound investigations. AEOD developed the training program following discussions with representatives of the National Transportation Safety Board, the Federal Aviation Administration, and the National Aeronautics and Space Administration. The seventh IIT training course was conducted in September and October 1996. The class was composed of candidate IIT members and leaders. The 10-day course covered the IIT program requirements and reviewed investigation techniques.

8 INDEPENDENT SAFETY ASSESSMENTS

8.1 Independent Safety Assessment of Maine Yankee Atomic Power Station

In December 1995 the Union of Concerned Scientists forwarded an anonymous allegation to the State of Maine, and the State forwarded the allegation on to the NRC. The allegation was that Yankee Atomic Electric Company knowingly performed inadequate analyses to support an increase in the rated thermal power at which Maine Yankee Atomic Power Station (MYAPS) may operate. After performing a technical review, the NRC Office of Nuclear Reactor Regulation issued a confirmatory order on January 3, 1996, limiting power operation at the plant to the original licensed power level of 2440 MWt.

The NRC Office of the Inspector General (OIG) completed an inquiry into this allegation on May 8, 1996. OIG established that MYAPS had experienced problems with, and made modifications to, the RELAP/5YA computer code that was used in the emergency core cooling analysis for a small-break loss-of-coolant accident. OIG also reported weaknesses in the NRC review and follow-up activities that contributed to the NRC's failure to detect these deficiencies. In response to these findings and to the concerns of the Governor of Maine about the safety and the effectiveness of regulatory oversight of MYAPS, the NRC Chairman initiated an independent safety assessment of the plant.

This assessment was performed on-site during the months of July and August 1996 by a team composed of staff who were independent of any recent or significant regulatory oversight responsibility for MYAPS. Additionally, the assessment was coordinated with the State of Maine to facilitate participation by State representatives, consistent with the Commission's policy on cooperation with States at commercial nuclear power plants.

The Independent Safety Assessment Team found that, while overall performance at Maine Yankee was adequate for operation, a number of deficiencies existed in each of the areas assessed. These deficiencies included poor problem identification and resolution; weaknesses in the scope, rigor, and evaluation of testing; and declining material condition. The root causes of these deficiencies were determined to be (1) economic pressure to reduce costs that caused the licensee to limit the resources for addressing corrective actions and installing some plant improvement upgrades, and (2) a lack of a questioning culture that resulted in the failure to identify or promptly correct significant problems in areas perceived by management to be of low safety significance.

The economic pressures resulted in limitations on resources and interfered with the licensee's ability to complete projects and other work that would improve plant safety and testing activities. Examples include the failure to adequately test safety-related components; long-standing deficient design conditions, such as the undersized atmospheric steam dump valve environmental qualification issues; and the lack of effective improvement programs, such as the design basis reconstitution program. These and other examples discussed in the report illustrate the licensee's willingness to accept existing conditions, many of which became operator "workarounds".

Examples of issues that illustrated complacency and the failure to identify or promptly correct significant problems include the following:

- previously undiscovered deficient conditions of the service water and auxiliary feedwater water systems
- inadequacies in ventilation systems
- post-trip reviews which lacked rigor and completeness

- emergency operating procedures that may not have adequately addressed an inadequate core cooling event and a steam generator tube rupture under certain conditions

- lack of a questioning attitude during test performance and evaluation that was not conducive to discovering equipment problems but rather to accepting equipment performance and licensee self-assessments that occasionally failed to identify weaknesses, or incorrectly characterized the significance of findings

- ineffective or untimely corrective actions, leading to repetitive problems

9 COMMITTEE TO REVIEW GENERIC REQUIREMENTS

The Committee to Review Generic Requirements (CRGR) reviews all generic requirements proposed by the NRC staff that involve one or more classes of power reactors. The CRGR consists of senior managers from various headquarters program offices and, on a rotational basis, from one of the NRC regional offices. The AEOD Director serves as the CRGR Chairman, and the AEOD staff provides support for all of the Committee's activities. The AEOD Director also oversees plant-specific backfit activities of the NRC staff in the headquarters program offices and the regional offices. In 1996 one new member from a region was appointed to the CRGR. The membership of the CRGR as of September 30, 1996, is as follows:

Edward L. Jordan, Director, AEOD (Chairman)

Frank J. Miraglia, Deputy Director, Office of Nuclear Reactor Regulation

Malcolm R. Knapp, Deputy Director, Office of Nuclear Material Safety and Safeguards

Joseph A. Murphy, Executive Assistant to the Director, Office of Nuclear Regulatory Research

Charles W. Hehl, Director, Division of Nuclear Materials Safety, Region I

Dennis C. Dambly, Assistant General Counsel for Materials, Antitrust and Special Proceedings, Office of the General Counsel

While performing the CRGR review function, a CRGR member expresses an individual professional opinion about each item considered, rather than representing the view of his or her respective office. The members of the CRGR determine whether proposed new generic requirements have sufficient merit in terms of safety and are justified in terms of cost (where appropriate) before reaching a consensus recommendation about each issue considered. Each independent CRGR recommendation is given to the EDO for consideration.

In 1994 a staff proposal was submitted to the Commission to reduce the scope of the CRGR review and to evaluate various means of reducing the burden on CRGR members. On April 21, 1994, the EDO transmitted to the Commission SECY-94-109 proposing to reduce the basic scope of CRGR review to include only "high impact" and "controversial" generic correspondence and rules before public comment, issues which the staff has difficulty resolving after public comment, emergency and urgent generic correspondence, and significant proposals with highly expedited schedules. A June 15, 1994, staff requirements memorandum (SRM) directed the staff not to reduce the scope of the CRGR Charter but to consider, and to recommend a course of action for, enlarging the scope of CRGR review to include proposed generic requirements in the nuclear materials area. The SRM also directed the staff to look at measures which would lessen the time spent on CRGR reviews by individual CRGR members. The Committee evaluated this option and agreed to address, on a 1 year trial basis, selected nuclear materials issues identified by the NMSS Director or by the EDO. The Committee will assess whether or not the nuclear materials issues that are presented by the staff for CRGR review warrant CRGR attention and, if so, whether the CRGR review adds significant value. Based on that assessment, the Committee will make appropriate recommendations to the EDO regarding continuation of the CRGR review of nuclear materials issues. This assessment will be included in the CRGR meeting minutes during the trial period, and it will also be reported to the EDO in the CRGR Weekly Items of Interest to be reported to the Commission. This aspect of the expanded scope of CRGR review was included in the ongoing CRGR Charter revision process.

On February 9, 1996, in SECY-96-032, the EDO requested Commission approval for this 1 year trial program to include selected nuclear materials issues. The Commission was also informed that the CRGR has considered and adopted measures to lessen the time spent by members on CRGR reviews. When appropriate, based on lack of controversy, low expected impact, or small potential for error related to the proposed generic actions, the CRGR Chairman may agree to one of three courses of action: (1) defer the CRGR's review pending public comment on the proposal; or (2) agree to a negative consent approach which, in essence, is an abbreviated review; or (3) forgo a second CRGR review, thus reducing the number of dual reviews (i.e., review at both the proposed and final stage). All other staff proposals will be scheduled for regular CRGR review.

On March 22, 1996, the Commission approved Revision 6 to the CRGR Charter, which expanded the scope of CRGR reviews, on a 1 year trial basis, to include selected nuclear materials issues requested by the NMSS Director or the EDO.

In 1996 the CRGR held 16 meetings during which it discussed the following 20 issues, all related to power reactors. The Committee, in its reviews of proposed new generic requirements, continued to place emphasis on less prescriptive, more performance-based and risk-informed regulations. The CRGR supported the expedited review of eight items requested by the staff. Of these, five were proposed urgent bulletins and three were generic letters.

- Proposed urgent bulletin on RHR strainer clogging

  The CRGR supported the issuance of the proposed urgent bulletin subject to the following recommendations:
  - articulate the urgency of the bulletin
  - quantify the risk implications
  - ask licensees to ensure suppression pool cleanliness and demonstrate operability of the emergency core cooling system (ECCS) pumps within 90 days instead of waiting until the next refueling outage
  - focus the bulletin on BWRs only and clearly identify the ECCS pumps to which the strainer clogging concerns apply, and consider including other systems which take suction from the suppression pool
  - request that licensees establish long-term performance measures to demonstrate the cleanliness of the suppression pool and implement a program to ensure operability of pumps taking suction from the suppression pool
  - include in the Maintenance Rule the scope and frequency of suppression pool inspection and cleaning (the Committee noted that this was not a condition for the proposed bulletin but an observation for consideration by NRR at an appropriate time)
  - wait until licensee responses to this urgent bulletin are received, evaluated, and incorporated as appropriate, before issuing the proposed bulletin on post-loss-of-coolant accident clogging of the ECCS pump strainers previously approved by CRGR

  The Committee expressed concern about the lack of effectiveness of the process for disseminating generic safety information and the inadequate utility efforts in analyzing and evaluating the information. The Committee noted that in recent years there have been several generic communications from the NRC, the Institute of Nuclear Power Operations (INPO), and the Boiling Water Reactor (BWR) Owners' Group (BWROG) on strainer clogging at U.S. and foreign reactors. However, utilities are apparently focusing too narrowly on the specific problems and prescriptive measures discussed in those generic communications and are not considering the broader applicability of the information to their plants. The Committee believes that this matter warrants further NRC management attention.

- AEOD briefing on the staff's study of the Oconee Emergency Power Distribution System

The Committee identified possible issues for staff evaluation relating to the unusual aspects of the design of the emergency power distribution system, including the complexity of the interconnections, possible interactions among the three Oconee units and the Keowee hydro unit, and the possible need for operator training to most effectively use this unusual aspect of the design in response to a loss of offsite power.

- CRGR Charter, Commission paper - Office of General Counsel (OGC) comments and changes by the CRGR staff

  The Committee approved by negative consent minor changes to the CRGR Charter recommended by OGC and the CRGR staff. Changes to the Charter approved by the Commission since the last Charter revision include the following:
  - incorporate guidance reflecting the Commission's understanding of the "substantial increase" standard of the backfit rule, specifically with regard to consideration of qualitative factors in justification of proposed backfits
  - expand the CRGR scope, on a trial basis, to include selected nuclear materials items
  - reflect the recent approval by the Commission of the new Regulatory Analysis Guidelines document, NUREG/BR-0058, Revision 2
  - include other modifications, such as changes in staff practices since the last Charter revision that are already being implemented.

- Proposed Generic Letter on Periodic Verification of the Design Basis Capability of Safety-Related MOVs (motor-operated valves)

  The CRGR recommended that the relationship of the generic letter to the Maintenance Rule be clarified.

- Safety Evaluation Report (SER) on BWROG Topical report NEDO 32264, "Application of PSA to GL 89-10 Implementation"

  The CRGR recommended that the SER should clearly indicate that Level 3 probabilistic risk assessment (PRA) expertise is not needed on the Expert Panel. As a related matter, the Committee recommended that the NRC give priority to the proposed 10 CFR 50.55(a) rulemaking regarding the 10 year inservice inspection update, and include incorporation of OMN-1 content in that rule by reference.

- Briefing on the SER on the Electric Power Research Institute Topical report TR103237, "MOV Performance Prediction Program"

- Proposed final Supplement 3 to NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Protective Action Recommendation for Severe Accident"

  The CRGR recommended that the *Federal Register* Notice to be issued in connection with the implementation of Supplement 3 should be changed to (1) more clearly emphasize that the preferred initial protective action is to evacuate promptly, rather than shelter the population in areas near the plant, barring any constraints to evacuation; and (2) to clearly indicate that licensees and offsite response organizations may continue to follow NUREG-0654, Revision 1, guidance to develop the appropriate protective actions for reactor accidents based on severe accident research insights.

- Expedited Bulletin on Control Rod Insertion Problems (Westinghouse plants)

  The CRGR recommended that the wording of the bulletin be revised to clarify that it is a 50.54(f) information request to verify compliance with the existing regulations and not a backfit, and to emphasize that the basis for this bulletin is to ensure compliance with the current licensing basis and that there is no adequate protection concern at this time.

- Expedited Bulletin on Strainer Clogging in BWRs

  The CRGR made the following comments:
  - use of the discharge (rather than suction) strainers (as is done at Nine Mile Point Unit 1) may be a viable option which is not included in the bulletin
  - an acceptable calculational methodology, which must be developed at least 6 months prior to scheduled implementation, is on the critical path, and this should be recognized in the text of the proposed actions
  - licensees would need to do 50.59 evaluations to justify their choice of the replacement insulation material

- Expedited Bulletin on Heavy Loads in BWRs

  The CRGR made several comments on the scope and the urgency of the proposed bulletin. Specifically, the Committee noted that the urgency of the requested actions was not adequately justified in the text of the bulletin as presented. The Committee recommended that the staff rewrite the bulletin to better justify the urgency of the requested actions

- Proposed Regulatory Guide for Reliability and Availability Data Reporting Rule

  The Committee recommended the following changes to the proposed Regulatory Guide:
  - make it clear that the intended scope is 7 to 10 (not 6 to 16) of the most risk-significant systems
  - delete the statement that the centralized database will be open to public access (Reliability analyses made by the NRC based on the collective data will be made publicly available)
  - state explicitly that the limited-scope data collected are intended to be sufficient to qualify the database for only those regulatory applications of PRA that fall within the limitations of data (a separate database will be developed for initiating systems)
  - use the same wording in the regulatory guidance document as in the NRC's enforcement policy regarding "occasional minor errors" and "acceptable level of accuracy"
  - make it clear that the guidance in the regulatory guide is not intended to supersede that contained in Generic Letter 89-17
  - ensure that the definition of the term “risk-significant” is the same in the regulatory guide as in the rule

- Proposed revisions to BWR Emergency Procedure Guidelines (EPGs)

  The CRGR noted that under the provisions of the Backfit Rule, the BWROG could not be required to revise its previously approved reactor pressure vessel water level control strategy, and that the licensees do have the option of using the BWROG approach; however, BWR licensees should be urged to consider the approach included in the revised EPGs. Additionally, the CRGR recommended that any reference to reactor core instabilities should be omitted from the staff's letter to the BWROG, and that the letters to individual licensees should be consistent with that to the BWROG.

- Proposed expedited Generic Letter on Changes to the Operator Licensing Program

  The Committee felt that, with regard to the technical safety aspects of operator licensing, the proposed changes represented a reasonable and workable alternative approach to the licensing process which reflects a substantial and well-coordinated effort on the part of the staff. The Committee endorsed sending the proposed changes forward for the consideration of the Executive Director for Operations (EDO) to support the tight schedule that has been established for consideration by the

Commission. This was done with the understanding that OGC would be asked to consider further the applicability of 50.109 (which appears to differ significantly from the position taken previously by the staff in similar actions affecting operator license renewal and requalification) and the planned use of a generic letter to implement the new staff position(s) as essentially mandatory requirements with no consideration of alternatives being proposed by the licensees.

The Committee noted that the wording in the current package states that licensees *will* implement the proposed changes. (The proposed generic letter did not request licensees to implement the new staff positions, and there is no provision in the generic letter, or elsewhere in the package, for obtaining written licensee commitments in that regard.) The incoming package also explicitly stated that "...NRC will *not* consider alternative testing methodologies...". (The staff believed that this is necessary "...to ensure uniform conditions for licensing operators," as required by the Atomic Energy Act). In effect, this constituted use of a generic letter to mandate new requirements. Even though industry comments appeared to generally favor implementation of the proposed changes, the CRGR questioned whether use of the generic letter mechanism in this manner was appropriate and consistent with Commission policy. The Committee indicated its intent to explore these questions further with OGC subsequent to the meeting. It was understood that OGC's determinations could result in some changes to the package, but CRGR did not object to the staff forwarding the package to the EDO, with a notation regarding these pending questions, in order to maintain the established schedule for the package.

- Briefing and Review of the Proposed Amendments to 10 CFR Part 100

At the 286th and 287th CRGR meetings, the CRGR reviewed the revision the final rulemaking package associated with the revised Reactor Site Criteria which involved revisions to 10 CFR 50 and 10 CFR 100 with respect to site suitability and nuclear power plant seismic design. The CRGR noted that the staff should verify that current plants which might apply for license renewal and license amendments would not be subject to the new (Subpart B) requirements as embodied in the package.

At the 286th CRGR meeting, the Committee noted that the seismic portion of the rule represents a major improvement, but had some comments with which the staff agreed. The Committee expressed concern about wording in the proposed new Appendix S to 10 CFR Part 50 which differs somewhat from comparable wording in Section VI of Appendix A to Part 100, which is applicable to existing operating plants. It was unclear if a subtle regulatory difference is intended by the use of "must" and "will" in Appendix S instead of "shall" that is used in the exisitng Appendix A. The Committee stressed that there should be consistency in the format and context of NRC regulations, unless a change in requirements is intended. The staff did not offer a clear explanation, however, the staff assured the Committee they would review the use of the wording to avoid any unnecessary and unintentional confusion.

At the 287th meeting, the CRGR review of the revised rule largely focused on the policy and guidance relating site acceptability to radiological dose limitations and population distribution. The Committee noted that the portion of this rule which deals with release of fission products into containment does include research and PRA insights, and is much more realistic than the present rule and guidance. However, some other important insights gained from severe accident research are not taken into account in this rule amendment. The CRGR specifically commented on the question of calculating Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary doses during the arbitrarily chosen 2 hour period, rather than cumulative exposure over the entire exposure period, and the assumption that a hypothetical individual would remain in the cloud path and not evacuate. Also, the Committee discussed the fact that the role of emergency planning and emergency response in limiting the dose to individuals at the outer boundary of the EAB or LPZ is not acknowledged, which would no doubt result in lower doses in practice. Furthermore, ground shine was also not taken into account as a significant, or even dominant, dose contributor.

In reference to a disagreement between the staffs of the Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research concerning the "first 2 hours" versus the "worst 2 hours," the CRGR did not take a position but noted that the choice of a 2 hour exposure period itself is arbitrary. Additionally, the Committee noted that the use of the "worst 2 hours" (and to a lesser degree, even the use of the "first 2 hours") could in some cases affect the engineered safety features design requirements (i.e., containment sprays) and, therefore, would couple siting to design. The CRGR believes that siting should be decoupled from design. The Committee was informed that a recommendation was to be made to the Commission in late 1996 regarding a possible follow-up Phase II (Part 100) rulemaking effort - a move that the Committee supported.

- Regulatory Guide 1.153 (proposed final) and Important-to-Safety Issue

  The CRGR recommended minor changes to the safety classification terminology in the draft Regulatory Guide. The Committee also recommended its issuance with the proposed use of the safety classification terminology, namely, the words "important to safety" being substituted for "safety-related" in the footnote. However, with respect to other use of the safety classification terms, the Committee recommended that the general safety classification issue ("important to safety" versus "safety-related") should be considered by the agency in a broader framework.

- Expedited Bulletin on Chemical, Galvanic and Other Reactions in Spent Fuel storage and Transportation Casks

  The Committee recommended the following changes to the bulletin:
  - emphasize that a root cause of hydrogen generation in the spent fuel storage cask is a deficiency in the licensee's design specification and review processes, and that the NRC did not fully consider material reactions and material compatibility in its licensing review of the VSC-24 cask and other storage and transportation casks
  - note explicitly that the gas ignition at Point Beach did not result in damage to the reactor facility
  - delineate more clearly the short-term concerns versus the long-term concerns arising from the Point Beach event, i.e., combustible gases generated by material reactions may create hazardous conditions while loading or unloading a cask (short-term), and products of the reactions between the coating and the water form a precipitate that may degrade the structural integrity of the cask and adversely impact the retrievability of stored spent fuel (long-term)
  - clarify the basis for concerns associated with the cask unloading operation
  - add cleaning agents to the list of materials to be considered by licensees in their evaluation of the potential for hydrogen generation
  - state explicitly that licensees should evaluate the effects of reactions involving carbozinc-11, or other equivalent coatings
  - indicate that, based on the available information and operational experience, it appears that the VSC-24 design is the most susceptible to hydrogen gas generation due to the relatively large use

of carbon steel and anti-corrosion coating
- revise the bulletin to follow more closely the format for a generic communication that includes "Requested Actions," and "Required Response" in accordance with 50.54(f), and allow 45 days for the required response indicating whether the addressees will implement the requested actions
- add a paragraph to the "Required Response" section of the bulletin indicating that licensee reports to the NRC should provide a detailed description of reviews and evaluations performed in response to the actions requested, including any compensatory measures implemented by the licensees
- include a discussion of the Confirmatory Action Letters, including supplements, that have been issued to licensees using VSC-24 so that all addressees will be aware of the actions and measures agreed upon

- Expedited review of 10 CFR 50.54(f) generic letter on Design Basis Information

  A major emphasis of the Committee's review was to ensure that the letter did not involve backfitting at this point, either explicitly or inadvertently, but instead was clearly restricted to requesting information regarding licensees' programs for documenting and maintaining properly the design bases for their facilities.

  The Committee inquired whether the NUMARC (Nuclear Managment and Resources Council) 90-12 guidance which was provided earlier to assist licensees in developing the needed design bases information was considered acceptable guidance in the current context. The staff felt that the guidance would still be useful, but noted that it does not specify reconstitution of any missing information. The Committee recommended that the wording of the letter rely heavily on and follow as closely as possible the wording of the Policy Statement, that the Policy Statement be included with the letter when issued, and that the letter state clearly the staff's current concern that reliance on the industry's voluntary efforts on improving design bases information, consistent with the NUMARC 90-12 guidance, the staff's comments on that industry guidance, and the Commission's Policy Statement, may not have been sufficient to maintain configuration control at some number of plants.

  There was also a discussion regarding the use of the word "required" (rather than "requested") in the context of the information collection. (The usual wording of 50.54 information requests for several years has been to "request" the specific information sought and "require" a licensee response.) This question was referred by the Committee to OGC for final determination.

- Urgent Generic Letter on Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions

  The Committee recommended the following revisions to the draft generic letter:
  - make it clear that the concern being addressed in this generic letter regarding breach of containment is focused on bypass leakage due to the rupture of containment cooling piping, not gross failure of containment
  - make it clear that licensees are being requested to look for possible water hammer or two-phase flow conditions with respect to any of the scenarios referenced in the generic letter and with respect to individual plant postulated accident conditions, not "postulated accident conditions" generally
  - require that licensees who choose not to complete the actions requested in the generic letter provide to the NRC their basis for continued operability of affected systems and components, in addition to the information already specified in the generic letter

- Expedited Generic Letter on Loss of Reactor Coolant Inventory and Associated Potential Loss of Emergency Mitigation Feature While in a Shutdown Condition

  The Committee recommended that the draft generic letter be revised as follows:
  - add a provision that licensees are expected to take appropriate corrective action in accordance with 10 CFR Part 50, Appendix B, if the requested evaluation identifies a susceptibility to common-cause failure from events similar to those described in this generic letter (in order to ensure compliance with GDCs 34 and 35)
  - indicate in the Backfit Discussion that the generic letter includes both a request for information and backfitting - if corrective action must be taken to address identified common-cause susceptibilities - which is justified under the provisions of 10 CFR 50.54(f) and 10 CFR 50.109(a)(4)(i) in order to verify and ensure compliance with applicable regulations
  - follow more closely the approved format for this type of generic communication to delineate more clearly the specific actions, information, and response that are requested, and the schedule to be adhered to by the licensee.

- Proposed Standard Review Plan Chapter Update

The CRGR complimented the staff on the organization of the information in the draft document and in the presentation material. The Committee endorsed the document for issuance for public comments. However, the Committee noted that ANSI/IEEE Std. 279-1971 (IEEE-279), which has been incorporated into the NRC regulations (specifically, 10 CFR 50.55a(h)), still serves as the licensing basis for most operating plants. Also, IEEE-279 is no longer maintained by IEEE and has now been superseded by ANSI/IEEE Std. 603-1991 (IEEE-603). If NRC regulations or staff positions are modified to replace IEEE-279 with IEEE-603, the licensing basis for of the operating plants may not conform with IEEE-603. The updated SRP Chapter 7 would also be used for reviewing license amendment requests for the operating plants; however, these plants would be expected to conform only to IEEE-279. Those plants which conform to IEEE-603 would automatically conform to IEEE-279. The Committee stressed that the staff needs to take the necessary steps to ensure consistency of NRC regulations with the current NRC-endorsed industry standards applicable to instrumentation and control.

10 INTERNATIONAL EXCHANGE OF INFORMATION

The growing use of nuclear power throughout the world and the recognition of the worldwide impact of a major nuclear event in any country 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 Unit 2 in 1979, international agencies developed the 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 on operational experience with the international community. These efforts have provided valuable data for AEOD studies and support for regulatory actions.

10.1 The Incident Reporting System

IRS is a cooperative program of the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA) of the United Nations. The United States and 13 other countries (Belgium, Canada, Finland, France, Germany, Italy, Japan, Republic of Korea, the Netherlands, Spain, Sweden, Switzerland, and the United Kingdom) are members of NEA. Through the IRS, NEA and IAEA member countries exchange information on safety-significant operational events at nuclear power plants that are of generic interest. The exchange takes place via the distribution of reports and quarterly updates of the IRS database that contains summary information about the reports. In this manner, all countries except Taiwan exchange information on operational experience.

In 1996 AEOD prepared and submitted 67 IRS reports. These reports addressed individual operational events and various generic concerns involving nuclear power plants in the United States, which were identified within the NRC's operational experience feedback program. The reports were based on generic communications sent to nuclear power plant licensees in the form of NRC reports (NUREG-series as well as AEOD studies), Information Notices, Bulletins, and Generic Letters. The report topics included precursors to potential severe core damage accidents, steam generator tube failures, clogging of emergency core cooling suction strainers by debris in boiling-water reactors, failures of charging/safety injection pump shafts, control rod insertion problems, pressure locking and thermal binding of safety-related valves, and problems with safety-related electrical equipment.

AEOD also reviews reports of selected foreign reactor events and identifies those that are safety-significant that could be applicable to plants in the United States. It then disseminates those reports to the appropriate NRC staff members. In 1996 AEOD received and reviewed approximately 110 reports from foreign countries.

In 1989 NRC and OECD finalized an agreement whereby NRC assumed responsibility for managing and operating the IRS database. AEOD carries out this function through a contract with the Oak Ridge National Laboratory. Managing the IRS includes the processing of all IRS reports received from NEA and mailing multiple copies of the quarterly updates of the IRS database on diskettes to NEA for further distribution to its member countries and IAEA. IAEA in turn distributes the diskettes to its member countries.

In 1996 the IRS database was modified to incorporate new names for some countries and nuclear power plants that came about as a result of the recent political changes in Eastern Europe. As part of this modification, new nuclear power plants that recently became operational were also added to the IRS database.

10.2 International Support Activities

As part of the NRC's international programs, AEOD exchanges information and ideas on a variety of topics of international interest. For example, AEOD staff provided assistance to foreign countries, to the NEA, and to the IAEA in a number of safety-related areas, including high burnup fuel and control rod insertion problems, undetected safety system failures, and the extended task force on human factors. In addition, AEOD and the Swedish regulatory body are the principal organizers of the international effort to develop a common-cause failure database.

AEOD is also the principal U.S. technical representative on reactor operating experience to the Principal Working Group 1 (PWG-1), "Operating Experience and Human Factors," of the NEA's Committee on the Safety of Nuclear Installations (CSNI). The 15th annual meeting of PWG-1 was held September 17 through 19, 1996, in Paris, France. At this meeting, the Group decided to complete the Nuclear Power Plant Data Collection database as soon as possible and to provide the data on CD ROM to member countries in early 1997. The Group also agreed to recommend to CSNI to approve the distribution of the IRS database to the World Association of Nuclear Operators. In addition, the Group approved the development of the Advanced Incident Reporting System with a number of recommendations. Also, the Study on Undetected Failures in Safety Systems, led by France with input from Belgium, Finland, Spain, and the U.S., was recommended to CSNI for approval and distribution.

AEOD is a participant in the Expert Group on Nuclear Emergency Matters. This group was established by the Committee for Radiation Protection and Public Health in 1989 to improve the quality of national and international nuclear emergency arrangements. Since then, the Expert Group has sponsored a series of international tabletop exercises in 16 countries and three workshops on specific issues identified during those exercises. The Expert Group is also in the process of planning for another series of international exercises. The NRC has been an active participant in each of these activities, all of which mesh with areas currently being worked on in this country.

The Expert Group is now planning for the Second International Nuclear Emergency Exercise (INEX2). INEX2 will be a series of regional exercises simulating an accident in one country and the other participating countries and the International Atomic Energy Agency will respond following their own emergency plan. The simulations, called command post exercises, will use actual nuclear plants, emergency centers, and real-time communications. Canada will host the North American INEX2 in November 1997 and U.S. Federal and State organizations will respond. Canada and the U.S. expect INEX2 to be the test for several projects that have been undergoing development or revision within the U.S./Canada working group, including a bilateral agreement for cooperation and assistance during nuclear emergencies that is being developed. Both the U.S. and Canada will exercise new Federal plans for the first time between countries in this exercise.

10.3 Lisbon Initiative Activities

AEOD is continuing to assist the regulatory authorities of Russia and Ukraine in the improvement of their own capabilities to respond to nuclear power plant emergencies. The AEOD staff, working with counterparts in the Federal Nuclear and Radiation Safety Authority of Russia and the Ministry for Environmental Protection and Nuclear Safety of Ukraine, is helping to establish reliable emergency communications with nuclear power plant sites, to prepare response plans and procedures, and to provide equipment for basic but functional emergency response centers in each country. The AEOD staff is also helping them prepare, conduct, and evaluate exercises so that they will be able to improve their response capabilities after the assistance program ends.

Because reliable communications between the regulatory authorities and each plant is essential, AEOD, the Russians, and the Ukrainians have focused their initial efforts on establishing that capability. A range of possible approaches has been considered. The preferred technology - satellites - is not feasible because of the high recurring usage charges. A combination of telephone systems -- some new and some newly accessible -- now appears to offer the best potential for increasing performance while decreasing initial and recurring costs. In the meantime, Russian and Ukrainian regulators have drafted key documents (concepts of emergency operations, plans, and procedures) and are beginning to build emergency response centers. Both countries recently held internal exercises to help define the requirements for its emergency teams and response centers.

AEOD coordinates its activities with those of the Department of Energy and other agencies of the U.S. Government as well as with related activities of other countries and organizations such as the United Kingdom, France, and the European community. When these tasks are completed in late 1997, each country's regulatory authorities will have an emergency response center and the necessary supporting capabilities to enable them to work quickly and effectively with other organizations to help limit the severity and consequences of a nuclear emergency.

AEOD is also assisting Ukraine in establishing an incident reporting and operating experience feedback system. This system includes strategies for data collection, events analysis and evaluation, regulatory response to events, and experience feedback to nuclear plants as well as information exchange between countries of the former Soviet Union with similar reactors. The Ukrainian effort is near completion. Five information exchange sessions and meetings have taken place. On-the-job training at NRC in response to events, as well as training provided by Idaho National Engineering and Environmental Laboratory (under contract to the NRC) in probabilistic risk assessment of operating events and NRC performance indicators, has taken place. The final step in this system is training in equipment and human performance reliability planned for 1997.

10.4 Limited Participation in the International Nuclear Event Scale

The NRC has participated in a limited manner in the International Nuclear Event Scale (INES) since December 1992. INES is a ranking system that is used to promptly and consistently communicate to the public the safety significance of reported events at nuclear installations worldwide. INES was designed by an international group of experts convened jointly by the IAEA and the NEA. The international scale is currently in use in 54 countries throughout the world.

The NRC usually limits its participation in the INES to rating only events at nuclear power plants that are classified as Alerts or higher on the emergency response scale used in the United States. However, additional events can be rated based upon management discretion. After a trial period of more than two years, the NRC decided to continue indefinitely its limited participation in the INES. Table 10.1 is a summary of the events at power reactors during Fiscal Year 1996 for which INES reports were submitted.

Table 10.1 U.S. Events Reported on the International Scale in 1996

Plant Name (Type)	Event Date	INES Level*	U.S. Classification	Event Description
Salem 1 (W/PWR)	10/04/95	Below Scale	Alert	Loss of control room annunciators for greater than 15 minutes (Alert was declared on 10/5/96)
Lasalle 1 (GE/BWR)	10/31/95	1	Alert	High radiation levels in containment due to withdrawal of a traversing incore probe to an unshielded location in the Reactor Building
Wolf Creek 1 (W/PWR)	01/30/96	2	Unusual Event	Decrease in water levels at the intake for safety-related water systems due to the buildup of frazil ice
Catawba 2 (W/PWR)	02/15/96	1	Unusual Event	Loss of offsite power, reactor trip, and safety-injection due to an electrical fault in isophase ducting
Palo Verde 2 (CE/PWR)	04/04/96	Below Scale	Alert	Fire in a control room lighting panel
Clinon 1 (GE/BWR)	08/19/96	1	Alert	Fire on reactor core isolation cooling pump turbine insulaton

* Events are classified on a scale with seven levels. The lower levels (1-3) are termed "incidents," and the upper levels (4-7) as "accidents." Events which have no safety significance are classified as below scale, or level 0, and are termed "deviations." Events which have no safety relevance are termed "out of scale."

Key to nuclear steam supply system vendors: W - Westinghouse Electric Company
GE - General Electric Company
CE - Combustion Engineering Company

APPENDIX A

Plant Operational Experience Data

A-1 Performance Indicator Program Data

A-2 Other Operational Experience Data

APPENDIX A-1

Performance Indicator Program Data

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Performance Indicator Program Data

Introduction

The NRC program for monitoring Performance Indicators (PIs) for operating commercial nuclear power reactors includes the following eight indicators: (1) the number of unplanned automatic scrams while a reactor is critical, (2) the number of selected safety system actuations, (3) the number of significant events, (4) the number of safety system failures, (5) the forced outage rate, (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. The data for cause code trends are obtained from the Sequence Coding and Search System database maintained at the Oak Ridge National Laboratory. The data for the remaining five PIs are obtained from Trends and Patterns databases maintained at the Idaho National Engineering Laboratory.

Background

In May 1986 an interoffice task group began to develop an NRC program for using quantitative indicators of nuclear power plant safety performance. In July and August 1986, the group conducted a trial program with 50 operating plants, testing 17 prospective performance indicators. For the most part, this trial program used data through calendar year 1985. The group then selected eight performance indicators as candidate 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 of an expanded trial program on 100 operating reactors using data for the first half of 1986. 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 proposed set of indicators. Beginning in February 1987, the AEOD staff provided quarterly PI reports to the Commission and to NRC senior managers. From March 1993 to September 1995, PI reports were issued semiannually, in June and December, with data through March and September respectively. The reports are now published annually in December with data through September. Reports are also placed in the NRC's Public Document Room. Beginning with the PI report for the fourth quarter of calendar year 1989, the staff has routinely provided plant specific information extracted from each PI report to licensee managers.

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

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

Definitions of the Indicators

Automatic Scrams While Critical. This indicator is the number of all unplanned automatic scrams that occur while the reactor is critical. A reactor scram means any actuation of the reactor protection system that results in control rod motion. The PI program also monitors the number of automatic scrams that occur while the reactor is critical at 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. This is the number of manual and automatic actuations of the logic for certain emergency core cooling systems, and actuations of the emergency ac power system that are 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 system, or the low-pressure core spray system are counted. Actuations of the reactor core isolation cooling system are not counted.

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

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

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

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

Collective Radiation Exposure. This indicator is the total radiation dose accumulated by unit personnel. Prior to the third quarter of 1992, all multi-unit sites except Indian Point and Millstone reported site total values, which were divided by the number of units at the site to obtain unit values. Since that time some multi-unit sites have reported individual unit values.

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

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	0	1	1	0	0	1	1
Arkansas 2	0	0	0	2	0	0	0	0
BeaverValley 1	0	0	0	0	0	0	1	0
BeaverValley 2	0	0	0	1	0	0	0	0
BigRockPoint	0	0	0	0	1	0	0	1
Braidwood 1	0	0	1	0	0	0	0	0
Braidwood 2	0	0	0	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	1	2	0	1	0	0	1	0
Browns Ferry 3	0	0	0	0	0	1	2	0
Brunswick 1	0	0	0	2	0	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	0	0	0	0	0	0	0	1
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	1	1	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	1	2	0	0	0	1	0	0
Catawba 1	0	0	0	0	0	0	0	0
Catawba 2	1	1	1	0	0	1	0	0
Clinton 1	0	0	0	0	0	0	1	0
Comanche Peak 1	1	0	1	0	1	1	0	1
Comanche Peak 2	0	0	1	0	1	0	0	1
Cook 1	0	0	0	1	0	1	0	1
Cook 2	1	1	0	3	0	0	1	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	0	0	0	0	0	0	1	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	1	0	0	1	0	0	0	1
Diablo Canyon 2	1	0	0	0	0	0	0	1
Dresden 2	0	0	0	0	0	0	0	0
Dresden 3	0	1	1	1	0	0	1	0
Duane Arnold	0	0	1	0	0	0	0	0
Farley 1	0	1	1	0	0	0	0	0
Farley 2	2	0	1	0	1	0	0	0
Fermi 2	0	0	2	0	0	0	0	0
FitzPatrick	0	0	0	1	0	0	0	1
Fort Calhoun	0	0	0	1	0	0	0	0
Ginna	0	0	0	0	0	0	0	1
Grand Gulf	1	1	0	4	0	0	0	0
Haddam Neck	0	0	0	0	0	0	0	0

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Harris	0	0	0	0	2	0	1	0
Hatch 1	1	0	0	0	0	2	1	0
Hatch 2	0	0	1	0	0	0	0	0
Hope Creek	2	0	0	0	0	0	0	0
Indian Point 2	0	1	1	0	0	1	1	2
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	0	0	1	1	0	1	1	0
LaSalle 1	1	0	0	1	0	0	1	0
LaSalle 2	1	0	0	0	0	0	0	0
Limerick 1	0	1	0	0	0	0	1	1
Limerick 2	1	1	0	2	0	0	1	0
Maine Yankee	0	0	0	0	0	1	0	0
McGuire 1	0	0	0	0	1	0	0	0
McGuire 2	0	0	0	0	0	0	1	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	0	0	0	0	0	0	0	0
Monticello	0	0	0	0	0	0	2	0
Nine Mile Pt. 1	1	0	1	0	0	0	1	0
Nine Mile Pt. 2	0	0	0	0	0	0	0	0
North Anna 1	0	1	0	0	0	0	0	1
North Anna 2	0	0	0	0	1	0	0	0
Oconee 1	0	0	0	0	0	1	0	0
Oconee 2	1	0	1	0	0	0	0	0
Oconee 3	0	0	0	1	0	1	0	0
Oyster Creek	0	0	0	0	1	0	1	1
Palisades	0	0	0	0	0	0	0	0
Palo Verde 1	0	0	1	0	2	0	0	1
Palo Verde 2	1	0	0	1	0	1	0	0
Palo Verde 3	0	0	0	0	0	0	0	1
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	1	0	0	1	1	0	0	0
Perry	0	0	0	3	0	0	1	0
Pilgrim	0	0	0	0	0	0	1	0
Point Beach 1	0	0	0	1	0	0	0	0
Point Beach 2	0	1	0	0	0	0	1	0
Prairie Island 1	0	0	0	0	0	0	1	0
Prairie Island 2	0	0	0	0	0	1	2	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	0	1	0	0	0	0

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
River Bend	1	0	0	0	0	0	0	0
Robinson 2	0	0	1	0	0	0	0	0
Salem 1	0	0	0	0	0	0	0	0
Salem 2	0	0	1	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	1	0	0
Sequoyah 1	1	0	1	1	0	0	1	0
Sequoyah 2	0	1	2	0	0	0	0	0
South Texas 1	0	1	0	1	1	0	0	0
South Texas 2	0	1	0	0	1	0	0	0
St. Lucie 1	1	0	0	1	0	0	0	0
St. Lucie 2	0	1	0	0	0	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	1	0	0	0	0	0	0
Surry 2	0	0	1	0	1	0	1	0
Susquehanna 1	0	0	0	0	0	0	0	1
Susquehanna 2	0	0	1	0	0	0	0	0
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	1	0	0	0	0	1	0	0
Turkey Point 4	1	0	0	0	0	0	0	0
Vermont Yankee	0	0	0	0	1	0	0	0
Vogtle 1	0	0	0	1	0	0	0	0
Vogtle 2	0	0	0	1	0	0	0	0
Wash. Nuclear 2	0	2	1	0	0	0	0	0
Waterford 3	0	0	1	0	0	0	1	0
Watts Bar 1	NYC	NYC	NYC	NYC	NYC	0	2	0
Wolf Creek	0	1	0	0	0	0	1	0
Zion 1	0	0	0	0	0	1	1	1
Zion 2	0	0	0	0	0	0	0	0
Total	25	22	27	37	16	17	34	19

NYC means the plant was not yet critical.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	0	0	0	0	0	0	0
Arkansas 2	0	0	0	0	0	0	0	0
Beaver Valley 1	0	1	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	0	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	0	0	0	0	0	0	0
Braidwood 2	0	0	0	0	0	1	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	0	0	0	1	2	0
Browns Ferry 3	0	0	1	0	0	0	2	0
Brunswick 1	1	0	0	2	0	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	0	0	0	0	0	0	1	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	1	0	0
Calvert Cliffs 2	0	0	0	0	0	1	0	0
Catawba 1	0	0	0	0	0	0	0	0
Catawba 2	0	0	0	0	0	2	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	0	0	0	0	0	1	0	0
Comanche Peak 2	0	0	0	0	0	0	0	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	0	0	0	0	0	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	0	1	0	0	0	0	0	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	1	0	0	0
Diablo Canyon 2	0	0	0	0	0	0	0	0
Dresden 2	0	0	0	0	0	0	0	0
Dresden 3	0	0	0	0	0	0	1	0
Duane Arnold	0	0	0	0	0	0	0	0
Farley 1	0	0	0	0	0	0	0	0
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	0	0	1	0	0	0	0	0
FitzPatrick	0	0	0	1	0	0	0	1
Fort Calhoun	0	0	0	0	0	0	0	0
Ginna	0	0	2	1	0	0	0	0
Grand Gulf	0	0	0	3	0	0	0	0
Haddam Neck	0	0	0	1	0	0	0	0

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	0	0	0	0	2	0	1	0
Hatch 1	0	0	0	0	0	0	0	0
Hatch 2	0	0	1	0	0	0	0	0
Hope Creek	0	0	0	0	1	0	0	0
Indian Point 2	0	1	0	0	0	0	1	0
Indian Point 3	0	1	1	0	0	2	0	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	0
LaSalle 2	0	1	2	0	0	0	0	0
Limerick 1	0	0	0	0	0	0	0	0
Limerick 2	1	1	0	1	0	0	0	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	0	0	0	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	0	0	1	0	0	0	1	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	0	0	0	1	0	0	0	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	0	0	0	1	0	0
Oyster Creek	0	0	0	0	0	0	0	0
Palisades	0	1	0	2	0	0	0	0
Palo Verde 1	0	0	0	0	1	0	0	0
Palo Verde 2	0	0	0	1	1	0	1	0
Palo Verde 3	0	0	0	0	0	0	0	0
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry	0	0	0	3	0	1	0	0
Pilgrim	0	0	0	0	0	0	0	0
Point Beach 1	0	1	0	0	0	0	1	0
Point Beach 2	0	1	0	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	1	0
Prairie Island 2	0	0	0	0	0	0	1	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	1	0	0	0	0	0

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
River Bend	0	0	0	0	0	0	0	0
Robinson 2	0	0	0	0	0	0	0	0
Salem 1	0	0	0	0	0	0	0	0
Salem 2	1	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	0
Sequoyah 2	0	0	0	0	0	0	1	0
South Texas 1	0	0	0	0	1	0	0	0
South Texas 2	0	0	0	0	0	0	0	0
St. Lucie 1	3	1	0	0	0	0	3	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	1	0	0	0	0	0	0
Surry 1	0	0	0	1	0	0	0	0
Surry 2	0	0	0	1	0	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	1
Susquehanna 2	0	0	0	0	0	0	0	1
Three Mile Isl 1	1	0	0	0	0	0	0	0
Turkey Point 3	0	0	0	0	0	0	0	0
Turkey Point 4	0	0	0	0	0	1	0	0
Vermont Yankee	0	0	1	0	0	0	0	0
Vogtle 1	0	0	0	0	0	0	0	0
Vogtle 2	0	1	0	0	0	0	0	0
Wash. Nuclear 2	0	0	0	0	0	0	0	0
Waterford 3	0	0	1	0	0	0	0	0
Watts Bar 1	NYL	NYL	NYL	NYL	0	0	0	0
Wolf Creek	0	0	0	0	1	0	0	0
Zion 1	0	0	0	0	1	1	0	0
Zion 2	0	0	0	0	0	0	0	0
Total	8	12	12	18	9	13	17	3

NYL means the plant was not yet licensed for low power operation.

Table A-1.3 Descriptions of Significant Events for 1996

Plant Name	Event Date	Rx Type	NRC Region	Description of Event
Arkansas 1	05/19/96	PWR	IV	A reactor trip resulted from a main feedwater pump control malfunction. The "B" steam generator boiled dry after one of its main steam safety valves stuck open.
Catawba 2	02/06/96	PWR	II	A reactor trip with complications resulted from a loss-of-offsite power. The complications included one emergency diesel generator out of service, a safety injection on low steam line pressure, a pressurizer power-operated relief valve actuation, the pressurizer filled solid, the pressurizer relief tank rupture disk actuated, and natural circulation established.
Dresden 3	06/11/96	BWR	III	A low-pressure coolant injection pump supply breaker failed to open on demand. The cause was grease hardening in the breaker mechanism due to inadequate maintenance. There were 23 safety-related breakers susceptible to this failure mechanism.
Haddam Neck	08/28/96	PWR	I	An incorrect valve lineup allowed nitrogen gas to displace about 6000 gallons of reactor coolant from the vessel. This could have resulted in common-mode failures of the residual heat removal and charging pumps, and could have impacted the ability to use steam generator cooling.
Hope Creek	03/14/96	BWR	I	Programmatic weaknesses resulted in violations and the imposition of civil penalties. This was a significant event because of repeated failures to plan appropriate post-maintenance equipment testing and to identify and correct problems with safety-related equipment.
Oconee 2	09/24/96	PWR	II	Severe water hammer caused a moisture separator/reheater drain pipe to rupture, resulting in significant injury to plant staff.
South Texas 1	12/18/95	PWR	IV	A turbine trip/reactor trip resulted from a partial loss-of-offsite power. Complications following the scram included three control rods indicating six steps from the bottom and power-operated relief valves actuating three times.
Wolf Creek	01/30/96	PWR	IV	A manual reactor scram was performed in anticipation of a loss of circulating water due to ice buildup on the traveling screens. Five control rods failed to fully insert following the scram.

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Arkansas 1	0	0	0	0	0	0	1	0
Arkansas 2	0	0	0	0	0	0	0	0
Beaver Valley 1	0	0	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	0	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	0	0	0	0	0	0	0
Braidwood 2	0	1	0	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	0	0	0	0	0	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	0	0	0	0	0	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	0	0	0	0	0	0	0	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	0	0	0	0	0	0	0	0
Catawba 1	0	0	0	0	0	0	0	0
Catawba 2	0	0	0	0	0	1	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	0	0	0	0	0	0	0	0
Comanche Peak 2	0	0	0	0	0	0	0	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	0	0	0	0	0	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	1	0	0	0	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	0	0	0	0	0	0	0	0
Dresden 2	0	0	0	0	0	0	0	0
Dresden 3	0	0	0	0	0	0	1	0
Duane Arnold	0	0	0	0	0	0	0	0
Farley 1	0	0	0	0	0	0	0	0
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	0	0	0	0	0	0	0	0
FitzPatrick	0	0	0	0	0	0	0	0
Fort Calhoun	1	0	0	0	0	0	0	0
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	0	0	0	0	0	0
Haddam Neck	0	1	0	0	0	0	0	1

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
Harris	0	0	0	0	0	0	0	0
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	1	0	1	0	0
Indian Point 2	0	0	0	0	0	0	0	0
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	0
LaSalle 2	0	0	0	0	0	0	0	0
Limerick 1	0	0	0	1	0	0	0	0
Limerick 2	0	0	0	0	0	0	0	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	0	0	0	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	1	1	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	0	0	0	0	0	0	0	0
Nine Mile Pt. 2	0	0	0	0	0	0	0	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	1
Oconee 3	0	0	0	0	0	0	0	0
Oyster Creek	0	0	0	0	0	0	0	0
Palisades	0	0	0	0	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	0	0	0	0	0	0	0	0
Peach Bottom 2	1	0	0	0	0	0	0	0
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry	0	0	0	0	0	0	0	0
Pilgrim	0	0	0	0	0	0	0	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	0	0	0	0	0	0

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
River Bend	0	0	0	0	0	0	0	0
Robinson 2	0	0	0	0	0	0	0	0
Salem 1	0	1	1	0	0	0	0	0
Salem 2	0	1	1	0	0	0	0	0
San Onofre 2	0	1	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	0
Sequoyah 2	0	0	0	0	0	0	0	0
South Texas 1	0	0	0	0	1	0	0	0
South Texas 2	0	0	0	0	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	0	0	0	0	0	0	0
Surry 2	0	0	0	0	0	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	0
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	1	0	0	0	0	0	0	0
Turkey Point 4	1	0	0	0	0	0	0	0
Vermont Yankee	0	0	0	0	0	0	0	0
Vogtle 1	0	0	0	0	0	0	0	0
Vogtle 2	0	0	0	0	0	0	0	0
Wash. Nuclear 2	0	0	1	0	0	0	0	0
Waterford 3	0	0	1	0	0	0	0	0
Watts Bar 1	NYL	NYL	NYL	NYL	0	0	0	0
Wolf Creek	0	0	0	0	0	1	0	0
Zion 1	0	0	0	0	0	0	0	0
Zion 2	0	0	0	0	0	0	0	0
Total	6	6	4	2	1	3	2	2

NYL means the plant was not yet licensed for low power operation.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	0	0	0	0	0	1	1
Arkansas 2	1	0	0	1	0	0	0	1
Beaver Valley 1	1	0	0	0	0	0	1	1
Beaver Valley 2	1	0	0	0	0	0	0	0
Big Rock Point	0	1	0	0	0	1	0	0
Braidwood 1	0	0	0	0	1	1	2	0
Braidwood 2	0	1	1	0	0	2	2	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	1	0	1	1	1	0
Browns Ferry 3	0	0	0	0	0	1	0	2
Brunswick 1	0	1	1	0	1	1	1	0
Brunswick 2	0	0	1	0	0	1	0	0
Byron 1	0	0	0	0	0	0	2	0
Byron 2	0	0	0	1	0	0	2	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	1	1	1
Calvert Cliffs 2	0	0	1	0	0	1	0	0
Catawba 1	1	0	1	1	0	0	0	0
Catawba 2	1	0	1	1	0	0	0	1
Clinton 1	0	0	1	0	0	1	1	0
Comanche Peak 1	0	0	0	1	0	2	0	0
Comanche Peak 2	0	0	0	1	0	1	0	0
Cook 1	1	0	0	3	0	0	0	0
Cook 2	0	0	0	0	0	0	0	0
Cooper Station	2	2	0	0	2	2	1	3
Crystal River 3	2	2	0	3	2	6	2	0
Davis-Besse	0	0	0	0	1	0	0	0
Diablo Canyon 1	0	1	0	1	3	2	3	0
Diablo Canyon 2	1	0	0	1	1	2	2	0
Dresden 2	0	1	0	0	1	1	1	1
Dresden 3	0	3	4	1	1	2	0	1
Duane Arnold	1	0	0	1	1	0	0	0
Farley 1	0	0	0	2	0	0	1	0
Farley 2	0	0	0	1	0	0	0	0
Fermi 2	0	1	0	0	0	3	0	0
FitzPatrick	1	1	0	0	0	3	1	1
Fort Calhoun	1	1	0	0	1	0	0	0
Ginna	0	0	0	0	0	1	0	0
Grand Gulf	0	0	0	0	1	0	0	0
Haddam Neck	1	6	0	0	3	1	3	5

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	0	0	0	0	0	0	1	0
Hatch 1	1	0	0	2	0	0	1	0
Hatch 2	0	0	0	0	1	0	1	0
Hope Creek	1	0	2	2	1	3	1	1
Indian Point 2	0	1	0	0	2	1	0	1
Indian Point 3	2	2	0	1	2	3	1	1
Kewaunee	0	0	0	0	0	1	0	1
LaSalle 1	2	2	0	0	0	0	1	0
LaSalle 2	3	1	0	0	1	0	1	0
Limerick 1	0	0	0	1	0	2	1	2
Limerick 2	0	0	0	0	0	2	1	1
Maine Yankee	1	1	2	1	2	0	1	1
McGuire 1	0	1	0	0	0	0	1	0
McGuire 2	0	0	0	1	0	1	0	0
Millstone 1	0	0	2	2	2	10	8	0
Millstone 2	3	5	4	1	1	10	3	0
Millstone 3	0	2	2	0	0	3	7	6
Monticello	1	0	1	0	1	1	1	0
Nine Mile Pt. 1	1	1	0	0	0	0	0	0
Nine Mile Pt. 2	0	1	0	0	0	0	0	1
North Anna 1	0	0	0	0	0	1	0	1
North Anna 2	0	0	0	0	0	0	0	1
Oconee 1	1	0	0	1	0	1	1	0
Oconee 2	0	0	0	1	0	1	1	0
Oconee 3	1	0	0	1	0	1	1	0
Oyster Creek	0	0	0	1	0	0	1	0
Palisades	0	0	1	4	1	3	0	1
Palo Verde 1	0	0	2	0	0	0	1	0
Palo Verde 2	3	0	3	0	0	0	1	1
Palo Verde 3	0	0	2	0	0	0	1	0
Peach Bottom 2	0	0	0	0	0	0	1	0
Peach Bottom 3	0	0	0	1	0	0	1	0
Perry	1	0	0	0	0	0	0	0
Pilgrim	1	2	1	0	1	0	1	0
Point Beach 1	0	0	0	0	0	0	1	1
Point Beach 2	0	1	0	0	0	0	1	0
Prairie Island 1	1	0	1	0	1	0	1	0
Prairie Island 2	1	0	1	0	1	0	1	0
Quad Cities 1	2	3	0	0	1	2	1	1
Quad Cities 2	0	2	0	0	3	1	2	1

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
River Bend	1	0	1	1	2	0	1	0
Robinson 2	1	0	0	0	0	0	0	1
Salem 1	2	1	3	3	4	2	2	4
Salem 2	3	1	1	3	4	2	2	5
San Onofre 2	0	0	0	0	1	0	0	0
San Onofre 3	0	0	0	2	1	0	0	0
Seabrook	0	0	0	0	0	0	1	0
Sequoyah 1	0	0	0	0	0	0	0	0
Sequoyah 2	0	0	0	0	0	0	0	0
South Texas 1	1	0	0	0	0	0	0	0
South Texas 2	1	0	0	0	0	0	0	0
St. Lucie 1	1	0	0	2	0	1	0	2
St. Lucie 2	0	0	0	0	1	0	1	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	0	0	0	1	0	0	0
Surry 2	0	1	0	0	1	0	0	0
Susquehanna 1	0	0	0	0	2	0	0	1
Susquehanna 2	0	0	0	0	1	0	0	1
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	2	1	0	2	0	1	0	0
Turkey Point 4	1	1	0	1	0	1	0	0
Vermont Yankee	4	1	1	1	0	1	2	1
Vogtle 1	0	0	0	1	0	1	1	0
Vogtle 2	0	2	0	0	0	0	1	0
Wash. Nuclear 2	2	0	0	0	0	0	0	0
Waterford 3	0	0	0	0	1	1	2	0
Watts Bar 1	NYL	NYL	NYL	NYL	0	2	1	1
Wolf Creek	0	0	0	0	1	4	1	0
Zion 1	0	1	0	1	0	2	1	1
Zion 2	0	1	0	0	0	3	2	1
Total	60	56	42	56	61	105	93	58

NYL means the plant was not yet licensed for low power operation.

Table A-1.6 Annual Safety System Failures

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
Arkansas 1	PWR	1	2	0	0	2
Arkansas 2	PWR	1	1	1	1	1
Beaver Valley 1	PWR	2	1	1	0	2
Beaver Valley 2	PWR	1	3	1	0	0
Big Rock Point	BWR	2	2	1	1	1
Braidwood 1	PWR	3	2	4	1	4
Braidwood 2	PWR	2	2	3	2	4
Browns Ferry 1	BWR	1	0	0	0	0
Browns Ferry 2	BWR	3	1	3	2	3
Browns Ferry 3	BWR	1	0	0	0	3
Brunswick 1	BWR	4	1	3	3	3
Brunswick 2	BWR	5	3	3	1	1
Byron 1	PWR	1	1	3	0	2
Byron 2	PWR	1	1	2	1	2
Callaway	PWR	0	3	1	0	0
Calvert Cliffs 1	PWR	1	2	2	0	3
Calvert Cliffs 2	PWR	3	2	0	1	1
Catawba 1	PWR	3	3	1	2	0
Catawba 2	PWR	3	3	1	2	1
Clinton 1	BWR	1	0	2	1	2
Comanche Peak 1	PWR	2	0	1	1	2
Comanche Peak 2	PWR	NYL	0	1	1	1
Cook 1	PWR	2	0	2	3	0
Cook 2	PWR	3	0	1	0	0
Cooper Station	BWR	10	11	8	4	8
Crystal River 3	PWR	3	1	3	7	10
Davis-Besse	PWR	1	1	1	1	1
Diablo Canyon 1	PWR	5	3	2	5	8
Diablo Canyon 2	PWR	4	5	2	2	5
Dresden 2	BWR	4	8	9	2	4
Dresden 3	BWR	9	7	5	9	4
Duane Arnold	BWR	6	4	3	2	1
Farley 1	PWR	0	1	0	2	1
Farley 2	PWR	0	1	0	1	0
Fermi 2	BWR	3	3	1	1	3
FitzPatrick	BWR	13	9	2	1	5
Fort Calhoun	PWR	3	6	2	2	1
Ginna	PWR	0	2	1	0	1
Grand Gulf	BWR	4	6	1	1	1
Haddam Neck	PWR	6	7	4	9	12

NYL means the plant was not yet licensed for low power operation.

Table A-1.6 Annual Safety System Failures

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
Harris	PWR	0	3	1	0	1
Hatch 1	BWR	3	2	3	2	1
Hatch 2	BWR	2	4	2	1	2
Hope Creek	BWR	4	4	1	5	6
Indian Point 2	PWR	1	4	0	3	4
Indian Point 3	PWR	4	13	4	5	7
Kewaunee	PWR	1	2	0	0	2
LaSalle 1	BWR	3	8	3	2	1
LaSalle 2	BWR	4	6	5	2	2
Limerick 1	BWR	3	1	0	1	5
Limerick 2	BWR	8	1	0	0	4
Maine Yankee	PWR	6	2	1	6	4
McGuire 1	PWR	1	6	0	1	1
McGuire 2	PWR	2	4	0	1	1
Millstone 1	BWR	8	6	7	6	20
Millstone 2	PWR	4	7	8	11	14
Millstone 3	PWR	6	4	1	4	16
Monticello	BWR	2	5	2	2	3
Nine Mile Pt. 1	BWR	2	0	1	1	0
Nine Mile Pt. 2	BWR	2	3	0	1	1
North Anna 1	PWR	1	3	0	0	2
North Anna 2	PWR	2	3	0	0	1
Oconee 1	PWR	10	5	2	1	2
Oconee 2	PWR	12	6	1	1	2
Oconee 3	PWR	9	6	2	1	2
Oyster Creek	BWR	1	1	3	1	1
Palisades	PWR	8	4	6	6	5
Palo Verde 1	PWR	1	2	0	2	1
Palo Verde 2	PWR	1	2	3	3	2
Palo Verde 3	PWR	2	2	0	2	1
Peach Bottom 2	BWR	3	6	3	0	1
Peach Bottom 3	BWR	3	5	4	1	1
Perry	BWR	5	4	4	0	0
Pilgrim	BWR	4	8	4	4	2
Point Beach 1	PWR	4	4	3	0	2
Point Beach 2	PWR	2	5	2	1	1
Prairie Island 1	PWR	3	0	1	2	2
Prairie Island 2	PWR	3	1	1	2	2
Quad Cities 1	BWR	15	8	7	4	5
Quad Cities 2	BWR	13	13	4	5	7

Table A-1.6 Annual Safety System Failures

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
River Bend	BWR	6	4	4	4	3
Robinson 2	PWR	7	2	3	0	1
Salem 1	PWR	3	2	4	11	12
Salem 2	PWR	1	3	4	9	13
San Onofre 1	PWR	0	PSD	PSD	PSD	PSD
San Onofre 2	PWR	2	1	0	1	1
San Onofre 3	PWR	1	2	0	3	1
Seabrook	PWR	3	4	0	0	1
Sequoyah 1	PWR	5	4	0	0	0
Sequoyah 2	PWR	8	4	1	0	0
South Texas 1	PWR	0	5	4	0	0
South Texas 2	PWR	1	4	2	0	0
St. Lucie 1	PWR	0	1	1	2	3
St. Lucie 2	PWR	0	0	0	1	2
Summer	PWR	0	1	0	0	0
Surry 1	PWR	5	2	0	1	1
Surry 2	PWR	7	1	0	2	1
Susquehanna 1	BWR	2	0	2	2	3
Susquehanna 2	BWR	2	0	3	1	2
Three Mile Isl 1	PWR	0	1	2	0	0
Trojan	PWR	3	PSD	PSD	PSD	PSD
Turkey Point 3	PWR	3	2	2	3	1
Turkey Point 4	PWR	1	1	1	2	1
Vermont Yankee	BWR	5	6	4	3	4
Vogtle 1	PWR	2	1	1	1	2
Vogtle 2	PWR	1	2	0	2	1
Wash. Nuclear 2	BWR	21	11	3	0	0
Waterford 3	PWR	0	3	1	1	4
Watts Bar 1	PWR	NYL	NYL	NYL	0	4
Wolf Creek	PWR	1	3	2	1	6
Yankee-Rowe	PWR	0	PSD	PSD	PSD	PSD
Zion 1	PWR	3	1	4	2	4
Zion 2	PWR	6	1	1	1	6
Total All Plants		384	353	219	215	317
Number of All Plants		111	109	109	110	110
Total All BWR Plants		187	161	110	76	113
Number of BWR Plants		37	37	37	37	37
Total All PWR Plants		197	192	109	139	204
Number of PWR Plants		74	72	72	73	73

NYL means the plant was not yet licensed for low power operation. PSD means the plant was permanently shutdown. Calendar year values are shown for 1992 through 1995. Fiscal year values are used for 1996. Data for October through December 1995 are included in both calendar year 95 and fiscal year 96.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	0	5	2	0	0	7	3
Arkansas 2	0	4	0	3	27	0	0	0
Beaver Valley 1	0	6	0	9	8	0	4	19
Beaver Valley 2	0	0	0	2	0	1	0	0
Big Rock Point	3	8	8	0	7	0	0	7
Braidwood 1	0	0	8	0	0	21	0	2
Braidwood 2	0	0	8	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	2	3	2	1	0	0	6	0
Browns Ferry 3	0	0	0	0	1	3	6	2
Brunswick 1	0	0	0	5	2	9	0	0
Brunswick 2	0	0	0	0	0	12	1	2
Byron 1	21	0	0	0	0	0	0	2
Byron 2	0	0	0	0	0	0	7	22
Callaway	0	0	1	1	10	1	1	0
Calvert Cliffs 1	0	0	5	0	7	0	100	36
Calvert Cliffs 2	10	7	5	1	0	7	0	0
Catawba 1	0	2	1	0	0	4	4	0
Catawba 2	2	11	6	0	27	13	0	10
Clinton 1	0	2	8	3	0	0	7	27
Comanche Peak 1	3	0	10	0	2	18	3	1
Comanche Peak 2	0	0	3	0	4	1	27	3
Cook 1	0	0	0	68	22	2	0	8
Cook 2	10	6	0	16	0	0	1	0
Cooper Station	100	57	0	0	0	0	0	0
Crystal River 3	0	0	0	0	0	40	13	31
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	4	0	0	2	19	0	2	6
Diablo Canyon 2	12	0	0	8	6	0	0	7
Dresden 2	72	29	9	0	0	0	52	66
Dresden 3	0	7	37	92	25	0	41	92
Duane Arnold	0	0	5	0	0	0	0	0
Farley 1	0	5	7	0	7	0	0	0
Farley 2	3	0	17	0	4	0	0	0
Fermi 2	100	70	29	0	0	5	24	0
FitzPatrick	0	0	10	8	0	15	0	8
Fort Calhoun	0	0	12	2	0	2	16	0
Ginna	0	0	0	2	0	4	0	15
Grand Gulf	2	2	26	13	0	0	3	0
Haddam Neck	1	1	2	4	0	0	0	53

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	0	0	0	0	21	9	3	4
Hatch 1	3	0	0	0	0	4	7	0
Hatch 2	0	0	18	3	0	0	3	0
Hope Creek	11	8	0	19	0	0	0	0
Indian Point 2	0	7	21	0	4	4	2	4
Indian Point 3	100	100	100	23	100	100	9	0
Kewaunee	0	0	0	1	0	1	1	0
LaSalle 1	9	6	7	11	0	0	17	24
LaSalle 2	6	0	0	9	0	8	6	19
Limerick 1	1	2	0	18	0	0	3	9
Limerick 2	2	5	0	6	1	0	4	0
Maine Yankee	6	71	0	0	0	3	0	48
McGuire 1	21	2	2	8	2	4	0	0
McGuire 2	0	1	6	0	9	0	77	2
Millstone 1	11	0	4	13	0	0	0	0
Millstone 2	0	0	0	12	6	10	0	0
Millstone 3	0	0	0	0	0	1	100	100
Monticello	2	0	0	0	0	0	18	0
Nine Mile Pt. 1	2	8	2	0	0	0	3	14
Nine Mile Pt. 2	11	9	11	9	0	0	0	0
North Anna 1	0	1	0	0	0	0	0	2
North Anna 2	0	0	0	0	1	0	0	0
Oconee 1	0	0	14	0	0	2	0	0
Oconee 2	5	0	23	0	0	0	2	14
Oconee 3	0	0	0	13	0	11	0	0
Oyster Creek	0	0	0	0	11	3	6	3
Palisades	0	0	11	5	0	15	0	3
Palo Verde 1	5	0	4	0	5	8	0	2
Palo Verde 2	7	0	0	2	0	3	0	0
Palo Verde 3	0	0	0	0	0	1	2	1
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	4	4	0	3	5	2	5	0
Perry	0	0	11	8	8	0	17	0
Pilgrim	52	1	29	0	0	0	0	14
Point Beach 1	0	0	0	5	0	0	0	0
Point Beach 2	0	1	0	0	0	0	1	0
Prairie Island 1	0	0	0	0	0	0	2	1
Prairie Island 2	0	0	0	0	0	2	3	1
Quad Cities 1	98	13	0	0	24	0	0	16
Quad Cities 2	81	0	0	24	37	0	57	49

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
River Bend	37	0	0	0	3	0	10	10
Robinson 2	0	0	1	5	0	0	0	0
Salem 1	0	29	49	100	0	0	100	100
Salem 2	2	53	25	100	100	100	100	100
San Onofre 2	0	0	10	0	2	0	0	0
San Onofre 3	0	0	0	0	0	0	0	8
Seabrook	0	0	13	5	0	4	0	0
Sequoyah 1	8	10	2	2	26	3	2	0
Sequoyah 2	1	3	25	0	4	0	0	0
South Texas 1	0	4	0	2	4	0	0	0
South Texas 2	0	3	0	0	4	3	0	0
St. Lucie 1	7	0	0	69	15	3	0	8
St. Lucie 2	0	4	0	4	0	2	9	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	8	2	0	0	0	0	0
Surry 2	0	0	9	0	16	4	0	2
Susquehanna 1	0	0	0	0	0	0	0	6
Susquehanna 2	0	0	7	0	0	0	0	20
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	4	0	2	0	1	9	0	4
Turkey Point 4	6	6	0	0	0	0	1	0
Vermont Yankee	2	0	0	0	3	0	0	0
Vogtle 1	0	0	0	3	0	0	24	0
Vogtle 2	0	0	0	2	0	0	0	0
Wash. Nuclear 2	0	9	29	0	0	0	53	0
Waterford 3	0	0	21	0	0	0	3	22
Watts Bar 1	NYC	NYC	NYC	NYC	NYC	NYC	1	0
Wolf Creek	0	6	0	0	0	14	2	0
Zion 1	12	0	0	0	0	9	10	27
Zion 2	0	0	0	0	3	0	3	0
Average	9	6	6	6	5	5	8	10

NYC means the plant was not yet commercial.

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Arkansas 1	0.00	0.00	1.41	0.46	0.00	0.00	0.49	0.57
Arkansas 2	0.00	0.58	0.00	0.00	3.93	0.00	0.00	0.00
Beaver Valley 1	0.00	0.00	0.00	0.50	0.00	0.00	0.00	0,55
Beaver Valley 2	0.00	0.00	0.00	0.46	0.00	0.00	0.00	0.00
Big Rock Point	0.00	0.50	0.00	0.00	0.96	0.00	0.00	0.48
Braidwood 1	0.00	0.00	0.50	0.00	0.00	0.00	0.00	0.46
Braidwood 2	0.00	0.00	0.50	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	1.12	0.00	0.00	0.46	0.00	0.00	0.00	0.00
Browns Ferry 3	0.00	0.00	0.00	0.00	0.00	0.47	0.96	0.49
Brunswick 1	0.00	0.00	0.00	0.94	0.00	0.99	0.00	0.00
Brunswick 2	0.00	0.00	0.00	0.00	0.00	1.69	0.46	0.53
Byron 1	0.85	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Byron 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.08
Callaway	0.00	0.00	0.00	0.46	0.50	0.92	0.46	0.00
Calvert Cliffs 1	0.00	0.00	0.48	0.00	0.95	0.00	0.00	0.66
Calvert Cliffs 2	0.50	0.00	0.00	0.45	0.00	0.00	0.00	0.00
Catawba 1	0.00	0.85	0.46	0.00	0.00	0.00	0.59	0.00
Catawba 2	0.00	1.54	0.97	0.00	8.49	0.52	0.00	0.50
Clinton 1	0.00	0.60	1.42	0.46	0.49	0.00	0.49	0.62
Comanche Peak 1	0.92	0.00	1.19	0.00	0.46	1.11	0.47	0.00
Comanche Peak 2	0.00	0.00	0.52	0.00	0.47	0.78	0.76	0.00
Cook 1	0.00	0.00	0.00	2.96	0.00	0.46	0.00	0.00
Cook 2	0.00	0.49	0.00	0.53	0.00	0.00	0.00	0.00
Cooper Station	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Crystal River 3	0.00	0.00	0.00	0.00	0.00	1.45	0.00	0.64
Davis-Besse	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Diablo Canyon 1	0.00	0.00	0.00	0.47	2.58	0.00	0.00	0.00
Diablo Canyon 2	0.00	0.00	0.00	0.49	0.00	0.00	0.00	0.48
Dresden 2	0.00	0.00	0.00	0.00	0.00	0.00	2.02	0.00
Dresden 3	0.00	0.00	0.73	2.35	0.53	0.00	1.50	0.00
Duane Arnold	0.00	0.00	0.61	0.00	0.00	0.00	0.00	0.00
Farley 1	0.00	0.00	0.00	0.00	0.68	0.00	0.00	0.00
Farley 2	0.92	0.00	2.29	0.00	0.00	0.00	0.00	0.00
Fermi 2	0.00	0.00	0.60	0.00	0.00	0.00	0.00	0.00
FitzPatrick	0.00	0.00	0.00	0.00	0.00	0.52	0.00	0.00
Fort Calhoun	0.00	0.00	1.18	0.00	0.00	0.51	0.54	0.00
Ginna	0.00	0.00	0.00	0.92	0.00	0.48	0.00	1.06
Grand Gulf	0.46	0.47	2.84	2.01	0.00	0.00	0.47	0.00
Haddam Neck	0.45	0.00	0.56	0.47	0.00	0.00	0.00	0.00

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Harris	0.00	0.00	0.00	0.00	1.35	0.00	0.94	0.48
Hatch 1	0.00	0.00	0.00	0.00	0.00	0.52	1.34	0.00
Hatch 2	0.00	0.00	0.51	0.51	0.00	0.00	0.47	0.00
Hope Creek	0.50	0.00	0.00	0.55	0.00	0.00	0.00	0.00
Indian Point 2	0.00	1.28	2.80	0.00	0.92	0.00	0.00	0.93
Indian Point 3	0.00	0.00	14.10	1.11	0.00	0.00	0.48	0.00
Kewaunee	0.00	0.00	0.00	0.46	0.00	0.00	0.46	0.00
LaSalle 1	0.98	0.49	0.49	0.99	0.00	0.00	0.67	0.59
LaSalle 2	0.48	0.00	0.00	0.49	0.00	0.96	0.48	0.00
Limerick 1	0.45	0.00	0.00	1.20	0.00	0.00	0.93	0.98
Limerick 2	0.46	0.00	0.00	0.94	0.45	0.00	0.93	0.00
Maine Yankee	0.47	3.12	0.00	0.00	0.00	1.05	0.00	0.00
McGuire 1	1.24	0.47	0.47	0.48	1.14	1.25	0.00	0.00
McGuire 2	0.00	0.52	0.49	0.00	0.49	0.00	6.47	0.00
Millstone 1	0.50	0.00	0.00	0.52	0.00	0.00	0.00	0.00
Millstone 2	0.00	0.00	0.00	0.77	0.48	0.82	0.00	0.00
Millstone 3	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Monticello	0.00	0.00	0.00	0.00	0.00	0.00	2.74	0.00
Nine Mile Pt. 1	0.00	1.09	0.96	0.00	0.00	0.00	0.49	1.55
Nine Mile Pt. 2	0.49	0.50	0.00	0.52	0.00	0.00	0.00	0.00
North Anna 1	0.00	0.47	0.00	0.00	0.00	0.00	0.00	0.46
North Anna 2	0.00	0.00	0.00	0.00	0.46	0.00	0.00	0.00
Oconee 1	0.00	0.00	0.53	0.00	0.00	0.47	0.00	0.00
Oconee 2	2.53	0.00	1.18	0.00	0.00	0.00	0.74	1.57
Oconee 3	0.00	0.00	0.00	2.39	0.00	0.51	0.00	0.00
Oyster Creek	0.00	0.00	0.00	0.00	0.50	0.00	0.00	0.63
Palisades	0.00	0.00	0.81	1.93	0.00	0.54	0.00	0.46
Palo Verde 1	0.47	0.00	1.18	0.00	0.95	0.00	0.00	0.00
Palo Verde 2	1.11	0.00	0.00	0.46	0.00	0.56	0.00	0.00
Palo Verde 3	0.00	0.00	0.00	0.00	0.00	0.00	0.92	0.00
Peach Bottom 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Peach Bottom 3	0.47	0.48	0.00	0.51	0.00	0.46	0.48	0.00
Perry	0.00	0.00	0.49	1.42	0.48	0.00	1.73	0.00
Pilgrim	1.27	0.51	0.00	0.00	0.00	0.00	0.00	0.52
Point Beach 1	0.00	0.00	0.00	0.47	0.00	0.00	0.00	0.00
Point Beach 2	0.00	0.50	0.00	0.00	0.00	0.00	0.00	0.00
Prairie Island 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Prairie Island 2	0.00	0.00	0.00	0.00	0.00	0.46	0.47	0.00
Quad Cities 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	2.29
Quad Cities 2	2.17	0.00	0.00	0.75	1.38	0.46	0.00	0.00

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
River Bend	0.67	0.00	0.00	0.00	0.00	0.00	0.50	0.00
Robinson 2	0.00	0.00	1.05	0.48	0.00	0.00	0.00	0.00
Salem 1	0.00	0.64	0.00	0.00	0.00	0.00	0.00	0.00
Salem 2	0.00	1.19	0.61	0.00	0.00	0.00	0.00	0.00
San Onofre 2	0.00	0.00	0.00	0.00	0.46	0.00	0.00	0.00
San Onofre 3	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.49
Seabrook	0.00	0.00	0.52	0.00	0.00	0.93	0.00	0.00
Sequoyah 1	0.49	0.51	0.00	0.60	3.78	0.00	0.46	0.00
Sequoyah 2	0.00	0.47	2.34	0.00	0.92	0.00	0.00	0.00
South Texas 1	0.00	0.69	0.00	0.00	0.00	0.00	0.00	0.00
South Texas 2	0.00	0.48	0.00	0.00	0.64	0.00	0.00	0.00
St. Lucie 1	0.00	0.00	0.00	10.24	0.52	0.47	0.00	1.88
St. Lucie 2	0.00	0.48	0.00	0.47	0.00	0.47	1.49	0.00
Summer	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Surry 1	0.00	0.50	0.47	0.00	0.00	0.00	0.00	0.00
Surry 2	0.00	0.00	1.49	0.00	0.54	0.48	0.00	0.00
Susquehanna 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Susquehanna 2	0.00	0.00	0.49	0.00	0.00	0.00	0.00	0.00
Three Mile Isl 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Turkey Point 3	0.46	0.00	0.47	0.00	0.49	0.99	0.00	0.47
Turkey Point 4	1.65	0.49	0.00	0.00	0.00	0.00	0.49	0.00
Vermont Yankee	0.00	0.00	0.00	0.00	0.46	0.00	0.00	0.00
Vogtle 1	0.00	0.00	0.00	0.00	0.00	0.00	0.74	0.00
Vogtle 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Wash. Nuclear 2	0.00	0.50	1.91	0.00	0.00	0.00	2.99	0.00
Waterford 3	0.00	0.00	0.57	0.00	0.00	0.00	0.47	0.58
Watts Bar 1	NYC	NYC	NYC	NYC	NYC	NYC	0.00	0.00
Wolf Creek	0.00	0.49	0.00	0.00	0.00	1.43	0.50	0.00
Zion 1	1.03	0.00	0.00	0.00	0.00	0.66	0.47	1.21
Zion 2	0.00	0.00	0.00	0.00	0.46	0.00	0.47	0.00
Average	0.19	0.16	0.29	0.29	0.27	0.20	0.30	0.20

NYC means the plant was not yet commercial.

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
Arkansas 1	4	195	11	3	2	4	4	NA
Arkansas 2	5	34	7	31	103	3	2	NA
Beaver Valley 1	6	247	13	13	4	56	207	NA
Beaver Valley 2	2	41	133	2	1	2	1	NA
Big Rock Point	75	9	8	22	14	172	25	NA
Braidwood 1	48	18	6	5	22	64	68	NA
Braidwood 2	48	18	6	5	22	64	68	NA
Browns Ferry 1	15	9	8	8	NLA	NLA	NLA	NA
Browns Ferry 2	418	180	16	14	27	126	133	NA
Browns Ferry 3	102	97	82	63	57	12	12	NA
Brunswick 1	52	53	217	34	38	49	16	NA
Brunswick 2	52	53	217	34	38	308	40	NA
Byron 1	65	78	5	2	43	3	122	NA
Byron 2	65	78	5	2	43	3	122	NA
Callaway	14	23	142	1	20	3	3	NA
Calvert Cliffs 1	5	33	77	4	5	4	100	NA
Calvert Cliffs 2	5	33	77	4	5	4	100	NA
Catawba 1	3	143	57	4	75	5	32	NA
Catawba 2	3	143	57	4	75	5	32	NA
Clinton 1	15	135	147	11	23	9	21	NA
Comanche Peak 1	30	66	14	1	1	53	8	NA
Comanche Peak 2	30	66	14	1	1	53	8	NA
Cook 1	52	6	3	72	21	25	74	NA
Cook 2	52	6	3	72	21	25	74	NA
Cooper Station	26	33	9	16	174	10	22	NA
Crystal River 3	10	3	1	1	3	257	69	NA
Davis-Besse	146	2	2	1	1	2	155	NA
Diablo Canyon 1	76	3	3	6	141	52	80	NA
Diablo Canyon 2	76	3	3	6	141	52	80	NA
Dresden 2	35	24	47	272	95	79	54	NA
Dresden 3	35	24	47	272	95	79	54	NA
Duane Arnold	39	236	90	22	25	17	13	NA
Farley 1	2	58	49	49	74	2	1	NA
Farley 2	2	58	49	49	74	2	1	NA
Fermi 2	39	4	8	6	9	9	14	NA
FitzPatrick	143	249	33	28	19	27	19	NA
Fort Calhoun	5	127	132	5	2	13	24	NA
Ginna	7	37	92	4	3	6	152	NA
Grand Gulf	13	17	290	26	10	16	18	NA
Haddam Neck	20	409	20	6	6	12	13	NA

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	8	6	16	143	9	4	7	NA
Hatch 1	174	19	34	59	164	76	102	NA
Hatch 2	174	19	34	59	164	76	102	NA
Hope Creek	63	17	14	22	143	138	9	NA
Indian Point 2	14	339	170	12	21	25	12	NA
Indian Point 3	22	16	38	4	9	6	8	NA
Kewaunee	1	1	104	2	2	2	2	NA
LaSalle 1	18	158	64	19	15	176	28	NA
LaSalle 2	18	158	64	19	15	176	28	NA
Limerick 1	5	15	13	25	16	83	11	NA
Limerick 2	5	163	13	25	16	83	11	NA
Maine Yankee	4	302	124	75	150	17	7	NA
McGuire 1	93	16	20	6	43	35	76	NA
McGuire 2	93	16	20	6	43	35	76	NA
Millstone 1	13	11	14	21	552	180	156	NA
Millstone 2	91	62	51	15	9	35	23	NA
Millstone 3	3	7	222	4	55	4	18	NA
Monticello	156	9	10	14	11	18	177	NA
Nine Mile Pt. 1	13	0	24	11	8	10	18	NA
Nine Mile Pt. 2	17	33	333	19	15	16	13	NA
North Anna 1	8	22	157	2	3	94	2	NA
North Anna 2	8	22	157	2	3	94	2	NA
Oconee 1	46	7	50	18	27	6	37	NA
Oconee 2	46	7	50	18	27	6	37	NA
Oconee 3	46	7	50	18	27	6	37	NA
Oyster Creek	348	32	19	17	22	21	18	NA
Palisades	13	10	190	166	9	14	6	NA
Palo Verde 1	19	48	150	3	4	4	17	NA
Palo Verde 2	19	48	2	3	3	59	95	NA
Palo Verde 3	19	48	3	5	177	7	4	NA
Peach Bottom 2	104	33	20	65	81	18	29	NA
Peach Bottom 3	104	33	20	65	81	18	29	NA
Perry	10	13	15	9	27	279	11	NA
Pilgrim	55	94	345	24	19	13	29	NA
Point Beach 1	47	39	16	2	38	3	23	NA
Point Beach 2	47	39	16	2	38	3	23	NA
Prairie Island 1	2	1	48	1	3	52	52	NA
Prairie Island 2	2	1	48	1	3	52	52	NA
Quad Cities 1	40	74	214	38	42	274	185	NA
Quad Cities 2	40	74	214	38	42	274	185	NA

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
River Bend	73	18	15	18	28	335	35	NA
Robinson 2	63	10	192	9	4	5	5	NA
Salem 1	6	22	8	98	41	62	13	NA
Salem 2	152	24	3	8	23	57	42	NA
San Onofre 2	4	93	12	113	6	2	4	NA
San Onofre 3	4	93	12	113	6	2	4	NA
Seabrook	1	1	6	4	92	4	2	NA
Sequoyah 1	17	7	9	86	149	13	11	NA
Sequoyah 2	17	7	9	86	2	6	213	NA
South Texas 1	1	108	35	3	2	1	190	NA
South Texas 2	1	2	2	4	150	2	1	NA
St. Lucie 1	160	22	6	26	152	7	158	NA
St. Lucie 2	160	22	6	26	152	7	158	NA
Summer	221	4	4	2	2	12	91	NA
Surry 1	34	80	7	72	43	9	74	NA
Surry 2	34	80	7	72	43	9	74	NA
Susquehanna 1	12	40	99	62	37	11	11	NA
Susquehanna 2	12	40	99	62	37	11	11	NA
Three Mile Isl 1	7	10	5	159	37	5	3	NA
Turkey Point 3	122	8	4	87	8	80	6	NA
Turkey Point 4	122	8	4	87	8	80	6	NA
Vermont Yankee	15	76	89	8	25	25	48	NA
Vogtle 1	23	77	15	4	4	88	41	NA
Vogtle 2	23	77	15	4	4	88	41	NA
Wash. Nuclear 2	33	40	359	34	23	35	299	NA
Waterford 3	2	1	12	39	101	4	3	NA
Watts Bar 1	NYB	NYB	NYB	NYB	NYB	NYB	NYB	NA
Wolf Creek	235	9	4	3	2	99	8	NA
Zion 1	8	109	4	63	221	25	4	NA
Zion 2	8	109	4	63	221	25	4	NA
Total	5418	6267	6412	3653	5292	5288	5358	NA

These data were obtained from the Institute of Nuclear Power Operations (INPO).
NLA means the data was no longer available from INPO.
NYB means the plant had not yet begun its first full calendar year of commercial operation.
Data were not available for 96-3.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	1	2	2	1	0	4	1	0
Arkansas 2	1	1	0	2	2	1	2	1
Beaver Valley 1	1	1	0	0	2	3	3	1
Beaver Valley 2	0	0	0	0	0	2	1	2
Big Rock Point	2	2	0	0	1	1	3	1
Braidwood 1	0	0	0	4	2	2	3	0
Braidwood 2	2	1	1	2	0	3	3	1
Browns Ferry 1	0	0	0	0	0	0	1	0
Browns Ferry 2	2	0	0	0	1	0	2	0
Browns Ferry 3	0	0	0	1	1	0	1	0
Brunswick 1	1	1	7	2	2	1	2	0
Brunswick 2	1	2	2	1	2	0	2	0
Byron 1	0	0	0	2	3	2	1	2
Byron 2	0	2	0	2	1	1	1	4
Callaway	0	0	1	0	0	0	0	0
Calvert Cliffs 1	0	0	0	1	0	0	1	1
Calvert Cliffs 2	1	2	2	0	0	0	0	0
Catawba 1	1	1	3	3	0	1	3	0
Catawba 2	2	0	3	3	0	1	2	0
Clinton 1	1	1	1	2	2	1	1	1
Comanche Peak 1	0	1	0	3	0	2	0	0
Comanche Peak 2	3	1	0	2	0	2	0	0
Cook 1	3	0	0	3	2	1	0	1
Cook 2	4	0	0	3	0	1	0	0
Cooper Station	8	6	1	0	6	1	2	2
Crystal River 3	6	3	2	2	2	2	5	0
Davis-Besse	2	0	0	0	1	1	2	0
Diablo Canyon 1	1	1	1	2	1	1	2	0
Diablo Canyon 2	2	1	1	2	0	1	2	0
Dresden 2	1	4	2	4	0	2	2	1
Dresden 3	2	6	6	5	2	1	7	5
Duane Arnold	2	1	1	0	0	0	0	0
Farley 1	0	0	0	1	2	0	2	0
Farley 2	1	0	0	1	0	0	3	0
Fermi 2	3	1	1	1	0	2	1	0
FitzPatrick	1	4	1	0	0	1	1	1
Fort Calhoun	3	1	1	0	1	2	1	0
Ginna	0	0	2	1	0	1	1	2
Grand Gulf	0	2	2	0	1	1	0	0
Haddam Neck	3	5	1	1	2	4	1	3

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	2	1	2	1	6	2	4	3
Hatch 1	0	1	0	0	0	1	3	1
Hatch 2	0	0	0	0	1	0	1	1
Hope Creek	2	5	4	5	11	8	3	1
Indian Point 2	2	4	1	3	0	3	0	2
Indian Point 3	2	6	4	8	1	2	2	1
Kewaunee	0	0	0	0	1	1	0	0
LaSalle 1	2	4	1	3	2	1	3	3
LaSalle 2	1	8	4	1	1	1	1	3
Limerick 1	1	0	0	3	0	3	3	2
Limerick 2	2	1	0	0	0	2	2	1
Maine Yankee	0	2	0	0	1	0	5	3
McGuire 1	1	0	0	0	0	0	0	0
McGuire 2	1	0	0	0	1	1	1	0
Millstone 1	2	4	1	5	4	12	9	0
Millstone 2	11	5	6	4	6	10	4	1
Millstone 3	1	2	5	2	3	3	5	6
Monticello	4	1	1	0	1	0	1	0
Nine Mile Pt. 1	2	1	2	0	0	2	3	1
Nine Mile Pt. 2	3	3	3	0	0	2	3	1
North Anna 1	1	1	0	0	0	0	0	1
North Anna 2	1	0	0	1	1	0	0	1
Oconee 1	1	0	0	0	0	1	1	1
Oconee 2	1	0	0	0	0	0	2	2
Oconee 3	0	0	0	0	0	1	1	1
Oyster Creek	1	1	1	1	2	3	2	1
Palisades	3	1	0	5	0	1	0	3
Palo Verde 1	1	2	0	1	2	0	1	1
Palo Verde 2	2	2	0	2	1	1	2	0
Palo Verde 3	2	2	0	1	1	0	1	0
Peach Bottom 2	0	0	2	2	0	1	0	0
Peach Bottom 3	0	0	1	2	1	1	0	0
Perry	2	1	0	1	0	0	1	1
Pilgrim	1	2	1	2	2	0	2	0
Point Beach 1	1	2	1	0	0	0	3	2
Point Beach 2	2	2	0	0	1	0	0	1
Prairie Island 1	2	4	1	3	0	4	1	1
Prairie Island 2	3	4	2	3	0	2	0	1
Quad Cities 1	5	0	0	0	1	3	2	6
Quad Cities 2	4	0	1	1	1	1	2	4

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
River Bend	4	2	0	0	2	2	2	1
Robinson 2	1	1	0	0	1	2	0	0
Salem 1	0	2	3	8	1	2	3	11
Salem 2	2	2	2	6	2	3	4	11
San Onofre 2	1	0	0	1	1	1	1	1
San Onofre 3	1	0	0	1	1	1	1	0
Seabrook	2	0	2	1	2	1	2	2
Sequoyah 1	5	1	1	1	3	1	0	2
Sequoyah 2	3	1	3	1	1	1	1	2
South Texas 1	3	2	0	1	0	0	0	0
South Texas 2	3	2	1	0	0	1	0	0
St. Lucie 1	1	1	1	2	0	3	4	3
St. Lucie 2	1	0	1	0	0	0	1	0
Summer	0	0	1	1	0	1	0	1
Surry 1	0	1	2	1	1	0	4	0
Surry 2	0	3	1	0	1	0	3	0
Susquehanna 1	0	3	2	0	5	0	1	2
Susquehanna 2	1	4	2	0	4	0	1	4
Three Mile Isl 1	1	0	0	0	1	0	0	0
Turkey Point 3	1	1	0	2	0	2	1	1
Turkey Point 4	1	1	0	1	0	2	2	1
Vermont Yankee	6	1	6	4	3	5	3	5
Vogtle 1	2	0	0	0	2	0	3	0
Vogtle 2	0	3	1	0	0	0	2	0
Wash. Nuclear 2	4	1	2	1	0	0	2	1
Waterford 3	2	0	1	1	1	3	1	2
Watts Bar 1	NYL	NYL	NYL	NYL	3	5	3	2
Wolf Creek	0	0	1	2	0	2	2	2
Zion 1	3	3	6	10	3	4	6	3
Zion 2	3	4	6	7	2	2	6	2
Total	184	164	133	169	131	166	198	145

NYL means the plant was not yet licensed for low power operation.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	1	0	0	0	0	0	0
Arkansas 2	0	0	0	1	0	1	0	0
Beaver Valley 1	0	1	0	0	1	0	0	0
Beaver Valley 2	0	1	0	1	0	0	0	0
Big Rock Point	0	1	0	0	1	0	1	0
Braidwood 1	0	1	0	1	2	0	0	0
Braidwood 2	2	1	0	1	1	0	1	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	1	0	0	0	0	1	0	0
Browns Ferry 3	0	0	0	0	1	0	0	0
Brunswick 1	0	0	0	1	0	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	3	0	0	1	0	0	0	0
Byron 2	1	1	0	0	0	0	0	1
Callaway	0	0	1	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	0	1	0	0	0	0	0	0
Catawba 1	0	0	1	0	0	0	0	0
Catawba 2	1	0	2	0	1	0	0	0
Clinton 1	0	2	1	1	0	0	1	0
Comanche Peak 1	0	0	1	0	0	1	0	0
Comanche Peak 2	1	0	1	0	0	0	0	0
Cook 1	0	0	0	1	0	0	0	0
Cook 2	0	0	0	1	0	0	2	0
Cooper Station	0	0	0	0	2	0	0	0
Crystal River 3	3	0	1	0	0	2	1	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	1	1	0	0	2	1	2	0
Diablo Canyon 2	2	1	1	0	0	1	1	0
Dresden 2	0	2	0	1	1	1	1	1
Dresden 3	0	1	1	1	0	0	1	2
Duane Arnold	0	0	0	0	0	0	0	0
Farley 1	0	0	0	1	0	1	0	0
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	0	0	0	0	0	1	0	0
FitzPatrick	2	0	1	1	0	0	0	1
Fort Calhoun	1	0	0	1	0	0	0	0
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	0	0	0	0	0	0
Haddam Neck	0	0	1	2	0	0	0	2

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	1	0	0	2	3	1	2	2
Hatch 1	0	0	1	0	0	1	1	0
Hatch 2	0	0	2	0	1	1	0	0
Hope Creek	1	1	0	4	3	0	1	0
Indian Point 2	0	2	1	0	0	0	1	0
Indian Point 3	1	0	0	2	1	2	0	0
Kewaunee	0	0	1	0	0	1	0	0
LaSalle 1	0	2	0	0	0	0	1	0
LaSalle 2	0	2	0	0	1	0	0	0
Limerick 1	0	0	0	1	0	1	0	0
Limerick 2	1	1	0	0	0	0	0	0
Maine Yankee	0	0	0	0	1	1	0	0
McGuire 1	0	0	0	0	0	1	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	0	1	3	0	0	0
Millstone 2	1	0	2	1	2	2	0	0
Millstone 3	1	0	1	0	1	0	0	0
Monticello	1	0	0	0	0	0	0	0
Nine Mile Pt. 1	0	0	0	0	0	0	0	1
Nine Mile Pt. 2	0	1	1	0	0	0	2	0
North Anna 1	0	0	0	0	0	1	0	0
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	0	0	0	1	0	0
Oconee 2	0	0	0	0	0	1	1	0
Oconee 3	0	0	0	0	0	2	0	0
Oyster Creek	3	0	1	0	0	0	1	0
Palisades	2	0	0	0	0	2	0	1
Palo Verde 1	0	1	0	0	0	0	0	0
Palo Verde 2	1	0	0	0	0	1	2	1
Palo Verde 3	0	0	0	1	0	0	0	0
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry	0	1	0	2	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	1	0	0	0	0	2
Prairie Island 2	0	0	1	0	0	0	0	1
Quad Cities 1	0	0	0	0	0	0	0	1
Quad Cities 2	0	0	0	0	0	0	0	1

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
River Bend	1	0	0	0	0	0	0	0
Robinson 2	0	0	0	0	1	1	0	0
Salem 1	2	1	1	0	0	0	0	0
Salem 2	3	1	1	0	0	0	0	0
San Onofre 2	1	1	1	0	0	1	0	1
San Onofre 3	1	0	0	0	0	1	0	1
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	1	0	3	0	0	0
Sequoyah 2	0	0	0	0	1	0	0	0
South Texas 1	0	1	0	2	2	0	0	0
South Texas 2	0	0	0	0	1	0	0	0
St. Lucie 1	0	0	0	1	1	0	1	2
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	0	0	1	0	1	0	0
Surry 2	0	0	0	1	0	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	2
Susquehanna 2	0	0	0	0	0	0	0	2
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	1	0	0	0	0	1	0	0
Turkey Point 4	1	0	0	0	0	0	0	0
Vermont Yankee	1	0	2	0	1	0	0	0
Vogtle 1	1	0	0	1	1	1	0	0
Vogtle 2	0	0	0	0	0	1	0	1
Wash. Nuclear 2	0	0	1	0	0	0	0	0
Waterford 3	0	0	0	0	0	0	0	0
Watts Bar 1	NYL	NYL	NYL	NYL	1	2	2	1
Wolf Creek	0	0	0	0	0	3	0	0
Zion 1	0	1	0	1	0	1	0	0
Zion 2	0	1	0	1	2	1	1	1
Total	43	32	31	38	42	44	29	28

NYL means the plant was not yet licensed for low power operation.

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	1	2	1	0	0	1	0
Arkansas 2	1	1	0	2	1	0	0	0
Beaver Valley 1	0	1	0	1	0	2	0	0
Beaver Valley 2	0	1	0	1	1	2	0	0
Big Rock Point	1	1	1	0	1	1	1	1
Braidwood 1	0	0	0	1	0	1	0	1
Braidwood 2	0	1	1	1	0	0	2	0
Browns Ferry 1	0	0	0	0	0	1	0	0
Browns Ferry 2	0	1	0	0	0	1	0	0
Browns Ferry 3	1	0	0	2	2	0	1	0
Brunswick 1	3	0	0	0	0	0	0	1
Brunswick 2	2	1	0	0	1	0	0	0
Byron 1	1	0	0	0	1	0	1	1
Byron 2	1	0	0	0	0	1	1	1
Callaway	0	0	1	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	1	0	1
Calvert Cliffs 2	0	2	1	0	0	1	2	0
Catawba 1	0	0	0	0	0	0	1	0
Catawba 2	1	0	2	0	0	1	1	0
Clinton 1	0	0	0	1	0	0	3	0
Comanche Peak 1	0	0	0	1	0	1	0	0
Comanche Peak 2	2	0	0	1	0	3	1	0
Cook 1	2	0	0	3	0	0	0	1
Cook 2	2	1	0	3	0	2	0	0
Cooper Station	1	0	0	0	1	0	2	0
Crystal River 3	2	3	0	1	0	3	4	0
Davis-Besse	3	0	0	1	0	0	1	0
Diablo Canyon 1	0	1	0	2	1	1	2	1
Diablo Canyon 2	1	1	0	1	1	0	3	1
Dresden 2	0	0	2	0	1	1	3	1
Dresden 3	1	3	4	0	0	1	4	3
Duane Arnold	0	1	0	1	0	0	1	0
Farley 1	0	0	1	0	0	0	1	1
Farley 2	0	0	1	0	1	0	1	0
Fermi 2	2	1	0	0	0	0	1	1
FitzPatrick	0	1	0	1	0	1	0	2
Fort Calhoun	0	0	0	0	1	1	1	0
Ginna	0	0	0	1	0	0	1	3
Grand Gulf	0	1	1	0	0	0	0	0
Haddam Neck	0	1	0	0	0	2	0	2

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Harris	1	0	0	0	1	2	0	1
Hatch 1	2	0	0	1	2	0	0	0
Hatch 2	0	0	0	0	2	0	0	1
Hope Creek	1	1	3	2	3	2	0	2
Indian Point 2	0	3	2	0	2	0	1	0
Indian Point 3	0	2	1	2	0	1	0	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	1	0	1	0	0	2	3	1
LaSalle 2	1	1	3	0	2	1	1	3
Limerick 1	1	0	0	4	0	6	1	1
Limerick 2	0	0	0	3	0	3	0	1
Maine Yankee	1	0	0	0	0	0	3	3
McGuire 1	0	0	0	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	1	2	4	4	5	1
Millstone 2	2	2	3	1	0	4	0	1
Millstone 3	0	0	2	0	0	0	4	3
Monticello	0	1	1	0	0	0	1	1
Nine Mile Pt. 1	1	0	0	0	0	0	1	1
Nine Mile Pt. 2	0	1	0	0	1	1	0	1
North Anna 1	1	0	0	1	0	0	0	0
North Anna 2	1	0	0	1	1	0	0	1
Oconee 1	0	0	0	0	0	1	0	1
Oconee 2	1	0	0	0	0	1	0	0
Oconee 3	0	0	0	1	0	1	0	1
Oyster Creek	2	0	0	0	0	1	1	0
Palisades	0	1	0	1	1	0	0	0
Palo Verde 1	1	1	1	0	1	0	0	1
Palo Verde 2	0	0	1	0	0	0	0	0
Palo Verde 3	1	0	0	0	0	0	1	0
Peach Bottom 2	0	0	1	1	0	1	2	0
Peach Bottom 3	0	0	1	1	0	1	2	0
Perry	1	0	0	0	0	0	0	0
Pilgrim	1	0	0	1	0	0	2	0
Point Beach 1	0	2	1	0	0	0	0	0
Point Beach 2	1	3	0	0	2	0	0	0
Prairie Island 1	0	1	0	0	0	2	1	1
Prairie Island 2	0	1	0	0	0	0	0	1
Quad Cities 1	1	0	1	0	0	0	0	1
Quad Cities 2	0	0	1	0	0	0	0	0

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
River Bend	2	2	0	0	1	4	1	0
Robinson 2	0	0	0	0	0	1	0	0
Salem 1	1	0	0	3	0	1	0	2
Salem 2	2	0	0	4	0	2	2	1
San Onofre 2	0	1	1	1	0	1	1	0
San Onofre 3	0	1	0	1	0	2	1	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	4	1	1	0	0	2	0	1
Sequoyah 2	2	0	1	0	0	2	1	1
South Texas 1	0	0	1	1	1	0	2	0
South Texas 2	0	0	1	1	0	1	0	0
St. Lucie 1	0	0	0	2	0	2	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	1	0	0	0	1	1	0
Surry 1	0	0	0	0	1	0	1	1
Surry 2	1	0	1	0	0	0	0	2
Susquehanna 1	0	1	1	0	0	1	0	6
Susquehanna 2	1	2	1	0	0	1	1	3
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	0	0	0	0	0	2	0	0
Turkey Point 4	1	0	0	0	0	1	0	0
Vermont Yankee	4	1	2	1	2	1	0	2
Vogtle 1	0	0	0	0	1	0	1	0
Vogtle 2	0	0	0	0	0	0	0	1
Wash. Nuclear 2	0	3	0	0	0	0	1	0
Waterford 3	1	0	0	0	2	1	0	1
Watts Bar 1	NYL	NYL	NYL	NYL	2	2	2	0
Wolf Creek	0	0	0	0	1	0	1	1
Zion 1	0	0	0	1	0	3	1	3
Zion 2	0	0	0	0	0	2	0	2
Total	68	57	51	63	46	95	86	77

NYL means the plant was not yet licensed for low power operation.

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Arkansas 1	1	3	3	2	0	3	2	0
Arkansas 2	2	2	0	2	3	0	2	1
Beaver Valley 1	1	4	1	1	3	3	3	0
Beaver Valley 2	1	2	1	2	2	3	1	1
Big Rock Point	2	1	1	0	1	4	1	1
Braidwood 1	0	1	0	4	5	4	1	1
Braidwood 2	1	2	2	4	0	2	4	2
Browns Ferry 1	0	0	0	0	1	1	1	0
Browns Ferry 2	3	4	1	1	2	1	2	0
Browns Ferry 3	1	0	0	2	3	0	1	1
Brunswick 1	2	1	5	4	3	2	2	1
Brunswick 2	1	2	3	1	3	0	2	0
Byron 1	3	0	0	2	4	1	2	2
Byron 2	1	2	0	2	0	2	1	3
Callaway	0	0	3	1	1	0	0	0
Calvert Cliffs 1	0	0	2	1	2	2	0	1
Calvert Cliffs 2	1	2	3	0	0	1	1	0
Catawba 1	1	1	3	2	0	1	3	2
Catawba 2	2	1	4	2	1	2	2	1
Clinton 1	1	3	0	2	0	2	5	2
Comanche Peak 1	1	1	2	3	0	3	0	0
Comanche Peak 2	9	1	1	2	1	3	1	1
Cook 1	4	0	0	6	1	1	0	1
Cook 2	5	2	0	4	0	4	1	0
Cooper Station	7	6	0	0	8	1	3	2
Crystal River 3	6	3	2	1	2	4	7	0
Davis-Besse	3	0	0	1	1	1	1	0
Diablo Canyon 1	2	1	0	3	1	4	3	1
Diablo Canyon 2	7	1	1	4	0	3	5	2
Dresden 2	0	8	6	4	1	3	4	2
Dresden 3	2	7	6	3	3	1	3	5
Duane Arnold	2	3	1	4	2	0	1	0
Farley 1	0	1	2	2	1	1	3	0
Farley 2	1	1	3	1	1	0	3	0
Fermi 2	2	3	1	1	0	1	2	1
FitzPatrick	1	8	3	1	0	3	1	2
Fort Calhoun	2	2	1	1	0	1	1	0
Ginna	0	0	2	1	0	3	3	3
Grand Gulf	1	4	2	5	1	1	0	0
Haddam Neck	3	5	3	2	3	2	0	5

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

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	2	1	2	3	5	2	7	5
Hatch 1	4	2	2	1	2	1	4	1
Hatch 2	1	1	2	1	6	0	1	2
Hope Creek	2	5	5	7	12	7	3	3
Indian Point 2	2	8	5	3	2	2	2	3
Indian Point 3	2	3	4	6	1	2	2	1
Kewaunee	0	0	3	1	2	2	1	1
LaSalle 1	5	6	1	4	2	3	3	2
LaSalle 2	5	7	4	1	2	1	1	5
Limerick 1	2	2	0	5	2	5	4	2
Limerick 2	4	7	0	5	1	4	3	1
Maine Yankee	1	4	0	0	1	2	3	5
McGuire 1	1	1	0	2	0	1	1	0
McGuire 2	1	1	0	2	0	1	1	0
Millstone 1	3	3	2	5	11	9	11	1
Millstone 2	10	4	6	5	5	7	3	1
Millstone 3	1	3	5	2	4	3	7	7
Monticello	3	1	2	0	1	1	3	1
Nine Mile Pt. 1	2	1	3	0	0	1	0	2
Nine Mile Pt. 2	3	0	3	2	1	2	3	1
North Anna 1	1	2	0	1	0	2	0	1
North Anna 2	1	0	1	1	2	0	0	2
Oconee 1	1	0	1	1	1	2	1	2
Oconee 2	2	1	0	0	0	1	2	3
Oconee 3	0	0	0	1	0	1	1	2
Oyster Creek	3	1	1	1	3	3	2	1
Palisades	3	1	1	3	1	2	0	2
Palo Verde 1	3	1	2	1	4	0	0	1
Palo Verde 2	2	1	2	2	2	1	1	1
Palo Verde 3	2	1	1	0	1	0	1	0
Peach Bottom 2	0	0	2	2	1	1	2	0
Peach Bottom 3	1	1	1	3	4	1	1	0
Perry	2	1	0	3	0	0	1	1
Pilgrim	1	2	2	3	2	1	4	0
Point Beach 1	1	4	1	1	0	0	2	2
Point Beach 2	4	5	0	0	2	0	1	1
Prairie Island 1	3	4	3	3	0	6	1	3
Prairie Island 2	3	4	2	4	0	2	1	2
Quad Cities 1	4	2	0	1	2	2	1	4
Quad Cities 2	3	2	1	3	3	0	1	3

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

Plant Name	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
	Calendar Year-Quarter							
River Bend	4	2	1	0	5	5	2	1
Robinson 2	1	1	1	1	1	1	0	1
Salem 1	3	2	3	6	1	3	1	7
Salem 2	4	1	2	6	1	4	4	7
San Onofre 2	0	2	3	2	1	2	1	1
San Onofre 3	0	0	1	3	1	3	2	1
Seabrook	2	0	3	1	3	1	1	2
Sequoyah 1	4	2	3	3	3	2	3	1
Sequoyah 2	3	1	3	1	1	2	3	1
South Texas 1	3	3	2	4	2	0	2	0
South Texas 2	3	3	2	1	1	1	0	0
St. Lucie 1	2	1	1	4	2	4	4	4
St. Lucie 2	2	1	1	0	2	1	1	0
Summer	1	1	0	1	0	2	2	1
Surry 1	0	2	3	2	3	0	4	1
Surry 2	1	3	3	2	3	1	4	2
Susquehanna 1	0	4	5	1	4	0	1	6
Susquehanna 2	1	5	4	2	5	1	0	5
Three Mile Isl 1	2	0	0	2	1	0	0	0
Turkey Point 3	1	1	0	1	0	5	1	1
Turkey Point 4	2	1	0	1	0	2	2	1
Vermont Yankee	7	2	7	3	2	4	1	5
Vogtle 1	1	0	0	0	1	0	3	1
Vogtle 2	0	4	1	0	0	0	1	3
Wash. Nuclear 2	3	3	3	1	0	0	3	1
Waterford 3	2	0	3	1	3	3	1	3
Watts Bar 1	NYL	NYL	NYL	NYL	5	7	3	1
Wolf Creek	0	1	1	2	1	4	1	2
Zion 1	3	1	6	10	3	6	4	4
Zion 2	3	4	6	7	2	5	5	4
Total	234	232	206	241	205	224	224	186

NYL means the plant was not yet licensed for low power operation.

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Arkansas 1	0	1	1	0	0	0	1	0
Arkansas 2	0	0	0	1	0	0	0	0
Beaver Valley 1	0	3	0	1	0	0	2	1
Beaver Valley 2	0	1	0	0	0	0	1	0
Big Rock Point	0	1	1	0	0	0	0	0
Braidwood 1	0	1	1	0	1	0	0	1
Braidwood 2	0	1	1	0	1	1	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	1	0	0	1	0	1	0	0
Browns Ferry 3	0	0	0	0	0	1	0	0
Brunswick 1	1	2	3	0	1	3	0	0
Brunswick 2	0	1	0	0	1	2	0	0
Byron 1	0	0	0	1	1	0	0	2
Byron 2	0	0	0	0	1	0	0	0
Callaway	0	0	1	0	0	0	0	0
Calvert Cliffs 1	0	0	0	1	1	0	1	0
Calvert Cliffs 2	0	1	1	0	0	0	1	0
Catawba 1	0	1	0	1	0	0	1	0
Catawba 2	0	0	0	1	0	0	1	0
Clinton 1	0	0	1	0	1	0	0	0
Comanche Peak 1	0	0	1	1	1	1	0	0
Comanche Peak 2	1	0	1	1	0	1	0	0
Cook 1	0	1	0	1	0	1	0	0
Cook 2	0	0	0	1	0	1	1	0
Cooper Station	6	2	3	0	2	2	1	2
Crystal River 3	0	2	0	5	6	5	3	0
Davis-Besse	0	1	0	0	0	2	1	0
Diablo Canyon 1	0	1	0	2	5	1	0	0
Diablo Canyon 2	1	1	0	1	1	2	0	0
Dresden 2	1	3	0	0	2	0	2	0
Dresden 3	3	2	1	1	3	1	3	0
Duane Arnold	1	0	0	0	0	0	0	0
Farley 1	2	1	1	0	0	0	1	0
Farley 2	3	1	0	0	1	0	1	0
Fermi 2	1	1	2	0	1	1	1	0
FitzPatrick	1	1	1	0	1	2	0	1
Fort Calhoun	3	0	0	0	2	0	1	0
Ginna	0	2	2	0	0	0	0	1
Grand Gulf	1	0	0	0	1	1	1	0
Haddam Neck	1	7	0	1	1	1	3	5

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
Harris	0	1	1	1	1	2	0	0
Hatch 1	0	0	0	0	0	0	2	0
Hatch 2	0	0	0	0	2	0	1	0
Hope Creek	3	1	0	1	1	3	2	0
Indian Point 2	1	2	1	0	0	1	0	2
Indian Point 3	1	2	1	2	1	3	1	1
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	1	2	0	1	0	0	1	0
LaSalle 2	3	4	0	1	0	0	0	1
Limerick 1	0	1	0	0	1	2	0	0
Limerick 2	0	1	0	0	0	1	1	0
Maine Yankee	0	2	2	3	2	1	5	8
McGuire 1	2	1	1	0	0	0	1	0
McGuire 2	1	1	1	0	0	1	1	0
Millstone 1	1	1	5	2	2	14	6	2
Millstone 2	4	8	5	2	2	10	3	1
Millstone 3	0	1	1	0	0	2	7	6
Monticello	0	0	0	0	0	0	4	0
Nine Mile Pt. 1	1	0	1	0	1	1	2	0
Nine Mile Pt. 2	0	3	1	1	2	0	1	1
North Anna 1	0	1	0	0	0	0	1	0
North Anna 2	0	1	0	0	0	0	1	0
Oconee 1	1	2	0	1	2	2	1	0
Oconee 2	0	2	1	1	0	1	1	0
Oconee 3	1	2	0	1	0	2	1	0
Oyster Creek	1	0	0	1	0	0	1	0
Palisades	1	1	1	4	2	2	0	2
Palo Verde 1	1	1	2	1	2	0	1	0
Palo Verde 2	1	1	3	2	2	0	1	0
Palo Verde 3	0	1	3	1	1	0	1	0
Peach Bottom 2	1	0	2	1	0	1	1	0
Peach Bottom 3	0	0	1	0	1	1	2	0
Perry	1	2	1	0	0	0	1	2
Pilgrim	1	1	0	0	1	0	1	0
Point Beach 1	0	1	0	0	0	0	1	3
Point Beach 2	0	0	0	0	0	0	0	3
Prairie Island 1	3	1	1	2	1	1	1	1
Prairie Island 2	3	1	1	2	1	1	1	1
Quad Cities 1	2	2	0	0	1	0	3	2
Quad Cities 2	3	2	0	1	1	0	1	2

Footnote at end of table.

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

	Calendar Year-Quarter							
Plant Name	**94-4**	**95-1**	**95-2**	**95-3**	**95-4**	**96-1**	**96-2**	**96-3**
River Bend	2	1	0	1	1	1	1	0
Robinson 2	0	0	1	0	0	0	0	0
Salem 1	2	2	1	2	1	2	2	8
Salem 2	1	1	0	2	1	2	2	8
San Onofre 2	1	0	0	1	1	0	1	0
San Onofre 3	0	0	0	1	1	0	1	0
Seabrook	1	0	1	1	0	0	1	0
Sequoyah 1	2	1	1	0	1	0	0	1
Sequoyah 2	1	1	2	0	1	0	0	1
South Texas 1	1	1	0	0	1	0	0	0
South Texas 2	1	0	0	0	0	1	0	0
St. Lucie 1	3	0	0	1	0	0	0	1
St. Lucie 2	1	0	0	0	0	0	1	0
Summer	0	0	0	0	0	1	0	2
Surry 1	0	0	1	2	0	0	0	0
Surry 2	0	1	1	0	0	0	1	0
Susquehanna 1	0	1	0	0	1	1	1	0
Susquehanna 2	0	2	0	0	1	1	0	1
Three Mile Isl 1	0	0	0	0	0	0	0	0
Turkey Point 3	3	0	1	0	1	1	0	0
Turkey Point 4	2	0	0	0	0	0	0	0
Vermont Yankee	1	1	3	1	1	2	4	5
Vogtle 1	1	0	1	1	1	1	1	0
Vogtle 2	0	0	1	0	0	0	1	0
Wash. Nuclear 2	3	1	0	0	0	0	1	0
Waterford 3	1	0	0	0	1	1	2	1
Watts Bar 1	NYL	NYL	NYL	NYL	1	4	4	0
Wolf Creek	0	0	0	0	1	1	1	0
Zion 1	0	1	0	0	3	2	0	1
Zion 2	0	2	0	0	2	3	0	1
Total	92	108	72	66	86	106	111	81

NYL means the plant was not yet licensed for low power operation.

Table A-1.15 Cause Codes — Miscellaneous — Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Arkansas 1	0	0	0	0	0	0	0	1
Arkansas 2	0	0	0	0	0	0	0	0
Beaver Valley 1	0	1	0	0	0	0	0	0
Beaver Valley 2	0	1	0	0	0	0	0	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	0	1	0	2	0	0	0
Braidwood 2	1	0	0	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	1	0	0
Browns Ferry 2	0	0	0	0	0	0	0	0
Browns Ferry 3	0	0	1	0	0	1	0	1
Brunswick 1	0	0	2	1	0	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	0	0	0	0	0	0	1	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	1	0
Calvert Cliffs 1	0	0	0	1	0	1	0	0
Calvert Cliffs 2	0	0	0	0	0	1	0	0
Catawba 1	0	0	0	0	0	0	2	0
Catawba 2	0	0	0	0	0	0	0	1
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	0	0	0	0	1	0	1	1
Comanche Peak 2	0	0	0	0	0	1	0	0
Cook 1	0	0	0	0	0	1	0	0
Cook 2	0	1	0	0	0	1	0	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	0	0	0	0	0	0	0	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	1	0	0	1
Diablo Canyon 2	0	0	0	0	1	0	0	1
Dresden 2	0	0	0	0	0	0	0	0
Dresden 3	0	0	0	1	0	0	0	0
Duane Arnold	0	1	1	0	0	0	0	1
Farley 1	0	0	0	0	0	0	0	0
Farley 2	0	0	0	0	1	0	0	0
Fermi 2	0	0	1	0	0	1	0	0
FitzPatrick	0	0	0	0	0	0	1	1
Fort Calhoun	0	0	0	0	0	0	0	0
Ginna	0	1	1	1	0	0	0	0
Grand Gulf	0	0	0	0	0	0	1	0
Haddam Neck	0	0	0	0	0	0	0	0

Table A-1.15 Cause Codes — Miscellaneous — Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
Harris	0	0	0	0	0	0	0	0
Hatch 1	0	0	0	1	0	1	1	0
Hatch 2	0	0	0	1	0	0	0	0
Hope Creek	1	0	0	0	0	0	2	0
Indian Point 2	0	0	0	0	0	0	0	0
Indian Point 3	0	0	0	0	0	1	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	1	0	0	0	0	0	0
Limerick 2	0	1	0	0	0	0	0	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	0	0	0	0	1	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	0	1	0	0	0	0
Millstone 2	0	0	0	1	0	0	0	0
Millstone 3	0	0	1	0	0	0	0	1
Monticello	2	0	0	0	0	0	0	0
Nine Mile Pt. 1	0	0	0	0	0	0	0	0
Nine Mile Pt. 2	0	0	0	0	0	0	1	0
North Anna 1	0	0	0	0	0	0	0	1
North Anna 2	0	0	0	0	0	0	0	0
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	1	0	0	0	0
Oyster Creek	0	0	0	0	0	0	0	0
Palisades	0	1	0	0	0	0	0	0
Palo Verde 1	0	0	0	1	0	0	0	1
Palo Verde 2	1	0	0	0	0	0	0	0
Palo Verde 3	0	0	0	0	0	0	0	1
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	0	0	0	1	0	0	0	0
Perry	0	0	0	0	0	1	0	0
Pilgrim	0	0	0	0	0	0	0	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	1	0
Prairie Island 2	0	0	0	0	0	0	1	0
Quad Cities 1	0	0	0	0	0	1	0	0
Quad Cities 2	0	0	0	0	0	0	0	0

Table A-1.15 Cause Codes — Miscellaneous — Quarterly PI Data

Plant Name	Calendar Year-Quarter 94-4	95-1	95-2	95-3	95-4	96-1	96-2	96-3
River Bend	0	0	0	0	0	0	0	0
Robinson 2	0	0	0	0	2	0	0	0
Salem 1	0	0	0	0	0	0	0	1
Salem 2	0	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	0	1	0	0
San Onofre 3	0	0	0	0	0	1	0	0
Seabrook	0	0	0	0	0	1	0	0
Sequoyah 1	0	0	0	0	0	0	0	0
Sequoyah 2	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	1	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	0	0	0	0	0	0	0
Surry 2	0	0	1	0	0	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	0
Susquehanna 2	0	0	0	0	0	1	0	0
Three Mile Isl 1	0	0	0	0	1	0	0	0
Turkey Point 3	0	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	1	0	0	1	0
Vogtle 2	0	0	0	1	0	0	0	0
Wash. Nuclear 2	0	1	0	0	0	0	0	0
Waterford 3	0	0	0	0	0	0	0	1
Watts Bar 1	NYL	NYL	NYL	NYL	0	0	0	0
Wolf Creek	0	0	0	0	0	0	0	0
Zion 1	0	0	0	0	0	1	0	1
Zion 2	0	0	0	0	0	0	0	1
Total	5	9	9	14	10	18	14	16

NYL means the plant was not yet licensed for low power operation.

APPENDIX A-2

Other Operational Experience Data

CONTENTS

Tables

Other Plant Operational Experience Data

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

Tables A-2.1 through A-2.5 present reactor scram data. Tables A-2.6 through A-2.9 contain data on engineered safety features actuations. Tables A-2.10 through A-2.12 provide annual unit operating factors and outage data. Table A-2.13 summarizes data on allegations at commercial nuclear plant sites.

Note that in Tables A-2.2 through A-2.5, because of round-off in some individual entries in the "Scrams/1000 Critical Hours" columns, the sum of the individual entries may not equal the total shown for that column.

Table A-2.1 Automatic and Manual Reactor Scrams While Critical and Reactor Scrams/1000 Critical Hours

Plant Name	Rx Type	CY92 Total Scrams	CY92 Auto	CY92 Man	CY92 Total Scram Rate	CY93 Total Scrams	CY93 Auto	CY93 Man	CY93 Total Scram Rate	CY94 Total Scrams	CY94 Auto	CY94 Man	CY94 Total Scram Rate	CY95 Total Scrams	CY95 Auto	CY95 Man	CY95 Total Scram Rate	1996 Total Scrams	1996 Auto	1996 Man	1996 Total Scram Rate
Arkansas 1	PWR	0	0	0	0.00	1	1	0	0.13	1	1	0	0.12	3	2	1	0.40	2	2	0	0.24
Arkansas 2	PWR	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	2	2	0	0.29	0	0	0	0.00
Beaver Valley 1	PWR	1	1	0	0.12	1	1	0	0.17	2	2	0	0.28	0	0	0	0.00	1	1	0	0.14
Beaver Valley 2	PWR	0	0	0	0.00	1	1	0	0.15	1	1	0	0.12	1	1	0	0.13	1	0	1	0.12
Big Rock Point	BWR	3	2	1	0.63	0	0	0	0.00	2	0	2	0.30	1	1	0	0.12	2	2	0	0.31
Braidwood 1	PWR	0	0	0	0.00	1	1	0	0.12	1	1	0	0.14	1	1	0	0.16	0	0	0	0.00
Braidwood 2	PWR	4	4	0	0.48	1	1	0	0.14	2	2	0	0.31	0	0	0	0.00	0	0	0	0.00
Browns Ferry 1	BWR	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Browns Ferry 2	BWR	2	2	0	0.24	1	0	1	0.17	3	3	0	0.41	3	3	0	0.35	1	1	0	0.13
Browns Ferry 3	BWR	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	3	3	0	0.41
Brunswick 1	BWR	2	2	0	0.79	0	0	0	0.00	0	0	0	0.00	3	2	1	0.40	1	0	1	0.12
Brunswick 2	BWR	1	1	0	0.42	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Byron 1	PWR	1	1	0	0.11	0	0	0	0.00	1	1	0	0.14	0	0	0	0.00	2	1	1	0.39
Byron 2	PWR	1	0	1	0.14	2	2	0	0.27	1	1	0	0.11	0	0	0	0.00	1	0	1	0.14
Callaway	PWR	3	3	0	0.41	0	0	0	0.00	0	0	0	0.00	3	2	1	0.40	1	0	1	0.12
Calvert Cliffs 1	PWR	1	1	0	0.20	2	1	1	0.23	3	3	0	0.51	3	0	3	0.35	2	0	2	0.35
Calvert Cliffs 2	PWR	5	1	4	0.63	1	1	0	0.16	4	4	0	0.50	3	2	1	0.42	1	1	0	0.12
Catawba 1	PWR	0	0	0	0.00	2	2	0	0.29	1	1	0	0.11	0	0	0	0.00	0	0	0	0.00
Catawba 2	PWR	2	2	0	0.24	1	1	0	0.14	4	3	1	0.57	3	2	1	0.42	1	1	0	0.14
Clinton 1	BWR	3	2	1	0.50	2	0	2	0.29	0	0	0	0.00	2	0	2	0.27	2	1	1	0.25
Comanche Peak 1	PWR	5	2	3	0.70	4	3	1	0.57	2	2	0	0.23	3	2	1	0.40	5	3	2	0.60
Comanche Peak 2	PWR	NYC	–	–	–	3	2	1	0.58	4	1	3	0.69	2	2	0	0.24	3	2	1	0.43
Cook 1	PWR	1	1	0	0.17	0	0	0	0.00	0	0	0	0.00	2	1	1	0.33	3	2	1	0.39
Cook 2	PWR	1	1	0	0.32	2	2	0	0.24	3	3	0	0.58	4	4	0	0.48	1	1	0	0.13
Cooper Station	BWR	0	0	0	0.00	1	1	0	0.19	1	1	0	0.33	0	0	0	0.00	0	0	0	0.00
Crystal River 3	PWR	2	2	0	0.30	1	1	0	0.13	0	0	0	0.00	0	0	0	0.00	1	1	0	0.18
Davis-Besse	PWR	1	1	0	0.11	2	2	0	0.27	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Diablo Canyon 1	PWR	2	2	0	0.27	1	1	0	0.12	1	1	0	0.14	3	1	2	0.41	4	1	3	0.56
Diablo Canyon 2	PWR	0	0	0	0.00	1	1	0	0.14	2	1	1	0.26	1	0	1	0.12	1	1	0	0.13
Dresden 2	BWR	0	0	0	0.00	0	0	0	0.00	1	0	1	0.17	1	0	1	0.33	1	0	1	0.55
Dresden 3	BWR	2	2	0	0.35	3	2	1	0.42	1	1	0	0.32	4	3	1	0.70	2	1	1	0.36
Duane Arnold	BWR	2	2	0	0.28	1	1	0	0.14	1	0	1	0.12	1	1	0	0.14	0	0	0	0.00
Farley 1	PWR	1	1	0	0.14	0	0	0	0.00	0	0	0	0.00	2	2	0	0.27	0	0	0	0.00
Farley 2	PWR	7	4	3	0.98	1	1	0	0.14	3	2	1	0.34	4	2	2	0.55	1	1	0	0.11
Fermi 2	BWR	3	0	3	0.42	5	4	1	0.61	0	0	0	0.00	2	2	0	0.26	0	0	0	0.00

NYC means the plant was not yet critical.

Table A-2.1 Automatic and Manual Reactor Scrams While Critical and Reactor Scrams/1000 Critical Hours

		CY92				CY93				CY94				CY95				1996			
Plant Name	Rx Type	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate
FitzPatrick	BWR	0	0	0	0.00	4	3	1	0.56	0	0	0	0.00	1	1	0	0.15	2	1	1	0.24
Fort Calhoun	PWR	3	3	0	0.52	2	2	0	0.28	1	1	0	0.11	3	1	2	0.41	1	0	1	0.12
Ginna	PWR	2	2	0	0.26	2	2	0	0.26	1	1	0	0.14	1	0	1	0.13	2	1	1	0.30
Grand Gulf	BWR	3	3	0	0.41	1	1	0	0.14	2	1	1	0.24	5	5	0	0.71	1	0	1	0.11
Haddam Neck	PWR	1	1	0	0.14	2	1	1	0.28	2	1	1	0.29	1	0	1	0.15	0	0	0	0.00
Harris	PWR	3	2	1	0.46	0	0	0	0.00	0	0	0	0.00	2	2	0	0.27	4	3	1	0.52
Hatch 1	BWR	4	3	1	0.47	5	3	2	0.70	2	2	0	0.26	0	0	0	0.00	4	3	1	0.51
Hatch 2	BWR	3	2	1	0.43	1	0	1	0.13	1	1	0	0.13	2	1	1	0.28	0	0	0	0.00
Hope Creek	BWR	1	0	1	0.14	2	1	1	0.23	5	5	0	0.70	1	0	1	0.14	0	0	0	0.00
Indian Point 2	PWR	3	3	0	0.35	0	0	0	0.00	0	0	0	0.00	2	2	0	0.34	4	4	0	0.48
Indian Point 3	PWR	2	2	0	0.37	0	0	0	0.00	0	0	0	0.00	2	0	2	1.07	1	0	1	0.23
Kewaunee	PWR	2	1	1	0.26	2	2	0	0.26	0	0	0	0.00	2	2	0	0.26	2	2	0	0.23
LaSalle 1	BWR	1	1	0	0.15	3	1	2	0.41	3	3	0	0.56	2	1	1	0.24	2	1	1	0.33
LaSalle 2	BWR	3	2	1	0.49	0	0	0	0.00	4	4	0	0.48	0	0	0	0.00	1	0	1	0.13
Limerick 1	BWR	0	0	0	0.00	1	1	0	0.12	1	0	1	0.13	3	1	2	0.37	2	2	0	0.25
Limerick 2	BWR	1	0	1	0.12	2	2	0	0.27	1	1	0	0.11	3	3	0	0.37	1	1	0	0.11
Maine Yankee	PWR	1	0	1	0.14	0	0	0	0.00	2	1	1	0.25	1	0	1	3.12	1	1	0	0.19
McGuire 1	PWR	2	2	0	0.29	1	1	0	0.19	1	1	0	0.16	3	1	2	0.37	2	1	1	0.26
McGuire 2	PWR	5	5	0	0.80	4	1	3	0.62	0	0	0	0.00	0	0	0	0.00	1	1	0	0.15
Millstone 1	BWR	1	1	0	0.17	1	1	0	0.12	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Millstone 2	PWR	0	0	0	0.00	5	5	0	0.65	1	0	1	0.23	1	0	1	0.29	0	0	0	0.00
Millstone 3	PWR	3	2	1	0.46	1	1	0	0.16	1	0	1	0.12	0	0	0	0.00	0	0	0	0.00
Monticello	BWR	0	0	0	0.00	3	2	1	0.41	2	2	0	0.26	0	0	0	0.00	2	2	0	0.26
Nine Mile Pt. 1	BWR	4	4	0	0.77	2	2	0	0.27	4	4	0	0.47	1	1	0	0.13	1	1	0	0.12
Nine Mile Pt. 2	BWR	2	2	0	0.35	1	1	0	0.14	2	1	1	0.24	4	0	4	0.57	0	0	0	0.00
North Anna 1	PWR	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	1	1	0	0.11	1	1	0	0.12
North Anna 2	PWR	2	2	0	0.27	2	1	1	0.27	1	1	0	0.12	1	1	0	0.14	1	1	0	0.12
Oconee 1	PWR	3	3	0	0.40	2	1	1	0.25	1	1	0	0.14	0	0	0	0.00	1	1	0	0.13
Oconee 2	PWR	1	1	0	0.14	2	2	0	0.27	2	2	0	0.27	1	1	0	0.12	1	0	1	0.13
Oconee 3	PWR	4	4	0	0.59	1	1	0	0.12	3	3	0	0.44	1	1	0	0.13	1	1	0	0.12
Oyster Creek	BWR	4	4	0	0.53	0	0	0	0.00	2	2	0	0.32	1	1	0	0.12	3	3	0	0.38
Palisades	PWR	5	5	0	0.75	0	0	0	0.00	0	0	0	0.00	1	0	1	0.15	0	0	0	0.00
Palo Verde 1	PWR	2	2	0	0.33	1	0	1	0.15	0	0	0	0.00	3	3	0	0.41	3	3	0	0.38
Palo Verde 2	PWR	3	3	0	0.35	2	1	1	0.42	2	2	0	0.33	1	1	0	0.13	1	1	0	0.13
Palo Verde 3	PWR	1	1	0	0.14	2	1	1	0.25	2	2	0	0.33	0	0	0	0.00	1	1	0	0.13

Table A-2.1 Automatic and Manual Reactor Scrams While Critical and Reactor Scrams/1000 Critical Hours

Plant Name	Rx Type	CY92 Total Scrams	CY92 Auto	CY92 Man	CY92 Total Scram Rate	CY93 Total Scrams	CY93 Auto	CY93 Man	CY93 Total Scram Rate	CY94 Total Scrams	CY94 Auto	CY94 Man	CY94 Total Scram Rate	CY95 Total Scrams	CY95 Auto	CY95 Man	CY95 Total Scram Rate	1996 Total Scrams	1996 Auto	1996 Man	1996 Total Scram Rate
Peach Bottom 2	BWR	4	3	1	0.65	1	1	0	0.13	1	1	0	0.13	0	0	0	0.00	0	0	0	0.00
Peach Bottom 3	BWR	4	3	1	0.52	2	1	1	0.30	2	1	1	0.23	3	2	1	0.37	1	1	0	0.12
Perry	BWR	1	1	0	0.15	2	0	2	0.47	1	0	1	0.23	3	3	0	0.36	1	1	0	0.15
Pilgrim	BWR	2	2	0	0.27	3	3	0	0.42	1	1	0	0.16	1	0	1	0.14	1	1	0	0.12
Point Beach 1	PWR	1	1	0	0.13	0	0	0	0.00	0	0	0	0.00	2	1	1	0.26	0	0	0	0.00
Point Beach 2	PWR	0	0	0	0.00	1	1	0	0.13	0	0	0	0.00	1	1	0	0.14	1	1	0	0.13
Prairie Island 1	PWR	0	0	0	0.00	1	1	0	0.12	0	0	0	0.00	0	0	0	0.00	1	1	0	0.14
Prairie Island 2	PWR	0	0	0	0.00	0	0	0	0.00	1	1	0	0.11	0	0	0	0.00	3	3	0	0.35
Quad Cities 1	BWR	1	1	0	0.16	1	1	0	0.14	1	0	1	0.38	0	0	0	0.00	1	0	1	0.28
Quad Cities 2	BWR	0	0	0	0.00	5	4	1	1.06	2	1	1	0.34	1	1	0	0.23	0	0	0	0.00
River Bend	BWR	3	3	0	0.86	2	2	0	0.32	3	2	1	0.53	1	0	1	0.11	3	0	3	0.40
Robinson 2	PWR	1	1	0	0.17	0	0	0	0.00	2	0	2	0.29	1	1	0	0.13	1	0	1	0.12
Salem 1	PWR	0	0	0	0.00	4	3	1	0.67	5	4	1	0.76	0	0	0	0.00	0	0	0	0.00
Salem 2	PWR	3	3	0	0.58	2	1	1	0.36	2	1	1	0.32	1	1	0	0.41	0	0	0	0.00
San Onofre 1	PWR	0	0	0	0.00	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–
San Onofre 2	PWR	2	2	0	0.24	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
San Onofre 3	PWR	1	1	0	0.15	2	2	0	0.30	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Seabrook	PWR	3	2	1	0.42	5	3	2	0.61	1	1	0	0.18	1	0	1	0.13	1	1	0	0.13
Sequoyah 1	PWR	4	3	1	0.51	2	1	1	1.56	3	2	1	0.50	5	2	3	0.73	3	1	2	0.41
Sequoyah 2	PWR	4	4	0	0.56	2	1	1	0.79	0	0	0	0.00	4	3	1	0.49	2	0	2	0.26
South Texas 1	PWR	1	1	0	0.16	0	0	0	0.00	2	1	1	0.28	3	3	0	0.39	1	1	0	0.12
South Texas 2	PWR	3	1	2	0.35	2	2	0	2.70	1	1	0	0.19	2	2	0	0.25	1	1	0	0.13
St. Lucie 1	PWR	1	1	0	0.12	3	0	3	0.44	5	4	1	0.64	2	1	1	0.30	3	0	3	0.47
St. Lucie 2	PWR	4	1	3	0.59	2	0	2	0.30	1	1	0	0.14	1	1	0	0.15	2	0	2	0.30
Summer	PWR	2	2	0	0.23	1	1	0	0.14	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Surry 1	PWR	2	1	1	0.28	2	2	0	0.24	1	0	1	0.15	2	1	1	0.26	0	0	0	0.00
Surry 2	PWR	0	0	0	0.00	5	5	0	0.78	0	0	0	0.00	4	2	2	0.56	3	2	1	0.40
Susquehanna 1	BWR	1	1	0	0.15	1	1	0	0.19	0	0	0	0.00	0	0	0	0.00	1	1	0	0.13
Susquehanna 2	BWR	1	0	1	0.14	0	0	0	0.00	1	1	0	0.15	1	1	0	0.13	1	0	1	0.13
Three Mile Isl 1	PWR	1	1	0	0.11	1	1	0	0.13	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Trojan	PWR	4	3	1	0.83	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–
Turkey Point 3	PWR	0	0	0	0.00	0	0	0	0.00	1	1	0	0.13	1	0	1	0.13	3	1	2	0.36
Turkey Point 4	PWR	1	1	0	0.14	2	2	0	0.27	2	2	0	0.26	0	0	0	0.00	1	0	1	0.13
Vermont Yankee	BWR	1	1	0	0.13	0	0	0	0.00	1	1	0	0.12	1	1	0	0.13	1	1	0	0.12
Vogtle 1	PWR	1	0	1	0.12	2	2	0	0.26	2	2	0	0.25	1	1	0	0.11	1	0	1	0.14

PSD means the plant was permanently shut down.

Table A-2.1 Automatic and Manual Reactor Scrams While Critical and Reactor Scrams/1000 Critical Hours

		CY92				CY93				CY94				CY95				1996			
Plant Name	Rx Type	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate
Vogtle 2	PWR	2	2	0	0.28	2	1	1	0.26	2	2	0	0.25	1	1	0	0.13	0	0	0	0.00
Wash. Nuclear 2	BWR	3	0	3	0.52	4	3	1	0.57	1	0	1	0.15	3	3	0	0.43	1	0	1	0.16
Waterford 3	PWR	0	0	0	0.00	2	2	0	0.23	1	1	0	0.13	1	1	0	0.14	1	1	0	0.13
Watts Bar 1	PWR	NYC	–	–	–	NYC	–	–	–	NYC	–	–	–	NYC	–	–	–	4	2	2	0.83
Wolf Creek	PWR	2	2	0	0.26	0	0	0	0.00	0	0	0	0.00	1	1	0	0.12	2	1	1	0.28
Yankee-Rowe	PWR	0	0	0	0.00	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–	PSD	–	–	–
Zion 1	PWR	0	0	0	0.00	1	1	0	0.14	2	2	0	0.47	0	0	0	0.00	3	3	0	0.53
Zion 2	PWR	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00	0	0	0	0.00
Total All Plants		195	154	41	0.27	163	121	42	0.22	142	111	31	0.19	155	102	53	0.20	138	86	52	0.18
Number of All Plants		111				109				109				109				110			
Total All BWR		66	50	16	0.30	60	42	18	0.26	52	39	13	0.22	54	37	17	0.21	42	27	15	0.16
Number of BWR Plants		37				37				37				37				37			
Total All PWR		129	104	25	0.25	103	79	24	0.21	90	72	18	0.17	101	65	36	0.19	96	59	37	0.18
Number of PWR Plants		74				72				72				72				73			

NYC means the plant was not yet critical. PSD means the plant was permanently shutdown.

Table A-2.2 Reactor Scram Initiating Systems

BWR Plants

	Total Scrams					Scrams/1000 Critical Hours[1]				
System	**CY92**	**CY93**	**CY94**	**CY95**	**1996**	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Turbine	14	12	13	10	13	0.06	0.05	0.06	0.04	0.05
Feedwater	9	8	4	9	10	0.04	0.03	0.02	0.03	0.04
Main Generator	4	5	7	7	5	0.02	0.02	0.03	0.03	0.02
Electrical	8	5	6	8	4	0.04	0.02	0.03	0.03	0.02
Support	5	7	5	8	4	0.02	0.03	0.02	0.03	0.02
Condensate	5	5	3	4	2	0.02	0.02	0.01	0.02	0.01
Main Steam	8	4	3	1	2	0.04	0.02	0.01	0.00	0.01
RPS	9	9	5	2	1	0.04	0.04	0.02	0.01	0.00
RCS	4	5	4	4	1	0.02	0.02	0.02	0.02	0.00
Control Rod Drive	0	0	2	1	0	0.00	0.00	0.01	0.00	0.00
Total	66	60	52	54	42	0.30	0.26	0.22	0.21	0.16
Number of BWR Plants	37	37	37	37	37					

[1]Critical hours: 1992 = 221,641.0; 1993 = 234,735.5; 1994 = 233,389.0; 1995 = 259,566.2; and 1996 = 256,807.8

PWR Plants

	Total Scrams					Scrams/1000 Critical Hours[2]				
System	**CY92**	**CY93**	**CY94**	**CY95**	**1996**	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Feedwater	40	28	20	24	26	0.08	0.06	0.04	0.05	0.05
Electrical	17	6	9	16	17	0.03	0.01	0.02	0.03	0.03
Turbine	17	12	11	9	12	0.03	0.02	0.02	0.02	0.02
Support	12	6	10	8	11	0.02	0.01	0.02	0.02	0.02
Main Generator	10	15	19	15	10	0.02	0.03	0.04	0.03	0.02
RPS	9	16	8	5	6	0.02	0.03	0.02	0.01	0.01
Control Rod Drive	5	5	4	12	5	0.01	0.01	0.01	0.02	0.01
Condensate	10	1	2	1	4	0.02	0.00	0.00	0.00	0.01
RCS	5	7	2	5	3	0.01	0.01	0.00	0.01	0.01
Main Steam	4	7	5	6	2	0.01	0.01	0.01	0.01	0.00
Total	129	103	90	101	96	0.25	0.21	0.17	0.19	0.18
Number of PWR Plants	74	72	72	72	73					

[2]Critical hours: 1992 = 512,763.6; 1993 = 491,488.6; 1994 = 518,224.2; 1995 = 518,681.0; and 1996 = 524,569.2

Table A-2.3 Activities at Time of Reactor Scram

BWR Plants

	Total Scrams					Scrams/1000 Critical Hours[1]				
Activity	**CY92**	**CY93**	**CY94**	**CY95**	**1996**	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Normal Operation	31	28	25	29	20	0.14	0.12	0.11	0.11	0.08
Power Change	14	8	9	7	9	0.06	0.03	0.04	0.03	0.04
Maintenance	6	9	8	6	8	0.03	0.04	0.03	0.02	0.03
Testing	15	15	10	12	5	0.07	0.06	0.04	0.05	0.02
Total	66	60	52	54	42	0.30	0.26	0.22	0.21	0.16
Number of BWR Plants	37	37	37	37	37					

[1]Critical hours: 1992 = 221,641.0; 1993 = 234,735.5; 1994 = 233,389.0; 1995 = 259,566.2; and 1996 = 256,807.8

PWR Plants

	Total Scrams					Scrams/1000 Critical Hours[2]				
Activity	**CY92**	**CY93**	**CY94**	**CY95**	**CY96**	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Normal Operation	65	59	52	64	54	0.13	0.12	0.10	0.12	0.10
Testing	21	25	10	14	18	0.04	0.05	0.02	0.03	0.03
Power Change	17	8	14	7	15	0.03	0.02	0.03	0.01	0.03
Maintenance	26	11	14	16	9	0.05	0.02	0.03	0.03	0.01
Total	129	103	90	101	96	0.25	0.21	0.17	0.19	0.18
Number of PWR Plants	74	72	72	72	73					

[2]Critical hours: 1992 = 512,763.6; 1993 = 491,488.6; 1994 = 518,224.2; 1995 = 518,681.0; and 1996 = 524,569.2

Table A-2.4 Reactor Scram Causes

BWR Plants

Cause	Total Scrams CY92	CY93	CY94	CY95	1996	Scrams/1000 Critical Hours[1] CY92	CY93	CY94	CY95	1996
Equipment	47	37	37	40	29	0.21	0.16	0.16	0.15	0.11
Personnel Error	12	14	9	8	8	0.05	0.06	0.04	0.03	0.03
Other	7	9	6	6	5	0.03	0.04	0.03	0.03	0.02
Total	66	60	52	54	42	0.30	0.26	0.22	0.21	0.16
Number of BWR Plants	37	37	37	37	37					

[1]Critical hours: 1992 = 221,641.0; 1993 = 234,735.5; 1994 = 233,389.0; 1995 = 259,566.2; and 1996 = 256,807.8

PWR Plants

Cause	Total Scrams CY92	CY93	CY94	CY95	1996	Scrams/1000 Critical Hours[2] CY92	CY93	CY94	CY95	1996
Equipment	80	72	63	74	72	0.16	0.15	0.12	0.14	0.14
Personnel Error	30	20	14	15	6	0.06	0.04	0.03	0.03	0.01
Other	19	11	13	12	18	0.04	0.02	0.03	0.01	0.03
Total	129	103	90	101	96	0.25	0.21	0.17	0.19	0.18
Number of PWR Plants	74	72	72	72	73					

[2]Critical hours: 1992 = 512,763.6; 1993 = 491,488.6; 1994 = 518,224.2; 1995 = 518,681.0; and 1996 = 524,569.2

Table A-2.5 Reactor Scram Signals

BWR Plants

Signals	Total Scrams					Scrams/1000 Critical Hours[1]				
	CY92	CY93	CY94	CY95	1996	CY92	CY93	CY94	CY95	1996
Manual	16	18	13	17	15	0.07	0.08	0.06	0.07	0.06
Turbine Trip	19	12	13	20	10	0.09	0.05	0.06	0.08	0.04
Low Reactor Water Level	11	12	5	6	5	0.05	0.05	0.02	0.02	0.02
Other	20	18	21	11	12	0.09	0.08	0.09	0.04	0.04
Total	66	60	52	54	42	0.30	0.26	0.22	0.21	0.16
Number of BWR Plants	37	37	37	37	37					

[1]Critical hours: 1992 = 221,641.0; 1993 = 234,735.5; 1994 = 233,389.0; 1995 = 259,566.2; and 1996 = 256,807.8

PWR Plants

Signals	Total Scrams					Scrams/1000 Critical Hours[2]				
	CY92	CY93	CY94	CY95	1996	CY92	CY93	CY94	CY95	1996
Manual	25	23	18	38	37	0.05	0.05	0.03	0.07	0.07
Turbine Trip	35	29	27	23	22	0.07	0.06	0.05	0.04	0.04
Low SG Level	23	17	16	12	9	0.04	0.03	0.03	0.02	0.02
Other	46	34	29	28	27	0.09	0.07	0.06	0.04	0.03
Total	129	103	90	101	96	0.25	0.21	0.17	0.19	0.18
Number of PWR Plants	74	72	72	72	73					

[2]Critical hours: 1992 = 512,763.6; 1993 = 491,488.6; 1994 = 518,224.2; 1995 = 518,681.0; and 1996 = 524,569.2

Table A-2.6 Engineered Safety Feature Actuations

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
Arkansas 1	PWR	2	2	1	0	1
Arkansas 2	PWR	1	0	0	1	0
Beaver Valley 1	PWR	6	4	3	5	5
Beaver Valley 2	PWR	6	7	4	1	1
Big Rock Point	BWR	2	1	2	0	0
Braidwood 1	PWR	4	0	2	0	0
Braidwood 2	PWR	2	1	1	1	3
Browns Ferry 1	BWR	5	2	1	1	1
Browns Ferry 2	BWR	6	6	5	3	3
Browns Ferry 3	BWR	2	3	2	6	6
Brunswick 1	BWR	13	7	10	16	1
Brunswick 2	BWR	7	6	5	1	0
Byron 1	PWR	1	1	1	0	1
Byron 2	PWR	1	2	2	0	1
Callaway	PWR	4	1	0	1	1
Calvert Cliffs 1	PWR	0	1	3	1	2
Calvert Cliffs 2	PWR	0	2	3	1	1
Catawba 1	PWR	3	2	0	0	1
Catawba 2	PWR	1	8	1	1	1
Clinton 1	BWR	3	3	1	2	3
Comanche Peak 1	PWR	5	1	0	0	2
Comanche Peak 2	PWR	NYL	5	6	0	4
Cook 1	PWR	0	0	0	1	0
Cook 2	PWR	0	1	1	1	3
Cooper Station	BWR	7	6	7	5	5
Crystal River 3	PWR	3	2	0	1	1
Davis-Besse	PWR	0	0	0	0	0
Diablo Canyon 1	PWR	4	1	6	4	5
Diablo Canyon 2	PWR	1	0	3	1	1
Dresden 2	BWR	17	9	9	1	2
Dresden 3	BWR	15	8	10	7	7
Duane Arnold	BWR	7	8	5	9	2
Farley 1	PWR	3	0	0	4	3
Farley 2	PWR	1	1	0	2	1
Fermi 2	BWR	6	7	4	4	4
FitzPatrick	BWR	10	5	3	3	3
Fort Calhoun	PWR	7	4	1	3	2
Ginna	PWR	5	1	4	5	6
Grand Gulf	BWR	7	12	1	7	1
Haddam Neck	PWR	0	3	0	1	0

NYL means the plant was not yet licensed for low power operation.

Table A-2.6 Engineered Safety Feature Actuations

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
Harris	PWR	1	1	1	5	6
Hatch 1	BWR	17	13	10	3	5
Hatch 2	BWR	13	6	6	6	6
Hope Creek	BWR	11	4	8	6	6
Indian Point 2	PWR	25	3	2	12	7
Indian Point 3	PWR	1	0	0	2	3
Kewaunee	PWR	5	7	2	1	2
LaSalle 1	BWR	6	6	6	3	3
LaSalle 2	BWR	12	4	3	6	5
Limerick 1	BWR	7	10	11	7	9
Limerick 2	BWR	4	9	5	10	4
Maine Yankee	PWR	1	0	0	0	0
McGuire 1	PWR	3	1	2	0	1
McGuire 2	PWR	3	4	0	1	2
Millstone 1	BWR	3	1	2	1	1
Millstone 2	PWR	1	0	6	5	0
Millstone 3	PWR	2	0	0	3	2
Monticello	BWR	2	3	11	1	3
Nine Mile Pt. 1	BWR	2	1	1	0	0
Nine Mile Pt. 2	BWR	18	5	3	4	2
North Anna 1	PWR	0	3	1	0	0
North Anna 2	PWR	3	0	0	0	0
Oconee 1	PWR	0	0	0	0	0
Oconee 2	PWR	1	0	2	0	0
Oconee 3	PWR	0	0	1	0	1
Oyster Creek	BWR	5	2	6	0	0
Palisades	PWR	9	1	2	3	0
Palo Verde 1	PWR	2	1	0	2	3
Palo Verde 2	PWR	5	2	0	2	2
Palo Verde 3	PWR	4	1	2	0	0
Peach Bottom 2	BWR	8	1	3	3	4
Peach Bottom 3	BWR	5	2	3	5	3
Perry	BWR	7	4	8	5	2
Pilgrim	BWR	13	12	4	4	2
Point Beach 1	PWR	3	2	0	2	1
Point Beach 2	PWR	2	0	2	2	1
Prairie Island 1	PWR	3	1	6	0	4
Prairie Island 2	PWR	3	1	2	1	2
Quad Cities 1	BWR	6	5	0	0	1
Quad Cities 2	BWR	9	5	2	4	1

Table A-2.6 Engineered Safety Feature Actuations

Plant Name	Rx Type	CY92	CY93	CY94	CY95	1996
River Bend	BWR	8	7	11	3	6
Robinson 2	PWR	1	1	0	1	0
Salem 1	PWR	16	9	2	0	0
Salem 2	PWR	18	4	3	2	0
San Onofre 1	PWR	0	PSD	PSD	PSD	PSD
San Onofre 2	PWR	1	0	0	0	0
San Onofre 3	PWR	4	0	1	0	1
Seabrook	PWR	4	3	1	1	0
Sequoyah 1	PWR	6	7	6	2	2
Sequoyah 2	PWR	6	3	2	0	4
South Texas 1	PWR	9	2	3	2	1
South Texas 2	PWR	4	4	7	4	0
St. Lucie 1	PWR	0	3	4	2	3
St. Lucie 2	PWR	1	2	0	0	0
Summer	PWR	3	1	1	1	1
Surry 1	PWR	2	2	0	1	0
Surry 2	PWR	2	0	0	5	1
Susquehanna 1	BWR	11	4	3	2	2
Susquehanna 2	BWR	4	3	3	5	5
Three Mile Isl 1	PWR	1	0	1	1	1
Trojan	PWR	6	PSD	PSD	PSD	PSD
Turkey Point 3	PWR	4	0	2	0	2
Turkey Point 4	PWR	4	0	0	0	1
Vermont Yankee	BWR	5	3	4	4	1
Vogtle 1	PWR	3	4	2	0	2
Vogtle 2	PWR	8	0	0	2	0
Wash. Nuclear 2	BWR	8	5	1	1	1
Waterford 3	PWR	4	0	1	1	0
Watts Bar 1	PWR	NYL	NYL	NYL	2	5
Wolf Creek	PWR	3	2	4	2	2
Yankee-Rowe	PWR	0	PSD	PSD	PSD	PSD
Zion 1	PWR	10	4	2	1	2
Zion 2	PWR	4	0	1	0	1
Total All Plants		552	327	295	254	222
Number of All Plants		111	109	109	110	110
Total BWR Plants		291	198	179	148	110
Number Of BWR Plants		37	37	37	37	37
Total PWR Plants		261	129	116	106	112
Number Of PWR Plants		74	72	72	73	73

NYL means the plant was not yet licensed for low power operations. PSD means the plant was permanently shutdown.

Table A-2.7 Engineered Safety Feature Actuations of Selected Systems

BWR Plants

System	CY92	CY93	CY94	CY95	1996
HVAC	154	109	65	82	52
RWCU	96	53	46	48	28
Emergency Power	30	37	18	16	14
ECCS	30	25	18	19	9
Total	310	224	147	165	103
Number of BWR Plants	37	37	37	37	37

PWR Plants

System	CY92	CY93	CY94	CY95	1996
Emergency Power	48	45	44	37	37
HVAC	103	31	20	19	23
ECCS	25	21	15	11	9
Total	176	97	79	67	69
Number of PWR Plants	74	72	72	73	73

ECCS - systems include: BWR - high pressure coolant injection, high pressure core spray, isolation condensers, low pressure core spray, and low pressure coolant injection.

PWR - high pressure safety injection, accumulators, and low pressure safety injection.

Emergency Power - includes all unplanned emergency diesel generator starts, including high pressure core spray diesel.

RWCU - BWR reactor water cleanup system.

HVAC - systems include: standby gas treatment, containment fan cooling, containment combustible gas control, containment purge, reactor building environmental control, drywell environmental control, shield annulus return and exhaust, access corridors environmental control, auxiliary building environmental control, fuel building environmental control, radwaste building environmental control, control building environmental control, emergency onsite power supply building environmental control, turbine building environmental control, and plant exhaust.

Table A-2.8 Engineered Safety Feature Actuation Activities

BWR Plants					
Activity	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Normal Operation	188	115	85	84	68
Testing	71	55	57	43	21
Maintenance	32	27	34	20	20
Other	0	1	3	1	1
Total	291	198	179	148	110
Number of BWR Plants	37	37	37	37	37

PWR Plants					
Activity	**CY92**	**CY93**	**CY94**	**CY95**	**1996**
Normal Operation	163	55	62	52	54
Testing	70	52	30	32	40
Maintenance	28	21	23	21	18
Other	0	1	1	1	0
Total	261	129	116	106	112
Number of PWR Plants	74	72	72	73	73

Table A-2.9 Engineered Safety Feature Actuation Causes

BWR Plants

Activity	CY92	CY93	CY94	CY95	1996
Equipment	138	92	72	67	55
Personnel Error	89	60	71	44	37
Procedure	32	20	14	24	10
Other	32	26	22	13	8
Total	291	198	179	148	110
Number of BWR Plants	37	37	37	37	37

PWR Plants

Activity	CY92	CY93	CY94	CY95	1996
Equipment	125	64	61	37	51
Personnel Error	75	43	26	32	31
Procedure	29	19	18	14	13
Other	32	3	11	23	17
Total	261	129	116	106	112
Number of PWR Plants	74	72	72	73	73

Table A-2.10 Critical, On-Line, Outage, and Availability Data for 1996

Plant Name	Reactor Type	Reactor Critical Hours	Generator On-Line Hours	Forced Outage Hours	Scheduled Outage Hours	Unit Availability Factor
Arkansas 1	PWR	8218.2	8198.3	188.1	397.6	93.3
Arkansas 2	PWR	7592.1	7347.6	289.6	1146.8	83.6
Beaver Valley 1	PWR	7036.8	6966.1	621.2	1196.7	79.3
Beaver Valley 2	PWR	8036.8	8013.9	21.5	748.6	91.2
Big Rock Point	BWR	6353.0	6274.1	307.9	2202.0	71.4
Braidwood 1	PWR	6636.7	6502.2	499.9	1781.9	74.0
Braidwood 2	PWR	7367.7	7350.0	0.0	1434.0	83.7
Browns Ferry 1	BWR	0.0	0.0	0.0	0.0	0.0
Browns Ferry 2	BWR	7981.0	7916.0	100.0	768.0	90.1
Browns Ferry 3	BWR	7238.3	7015.8	238.5	340.7	92.4
Brunswick 1	BWR	8328.1	8251.7	246.0	286.3	93.9
Brunswick 2	BWR	7475.9	7279.3	211.5	1293.2	82.9
Byron 1	PWR	5149.0	5057.8	38.9	3687.3	57.6
Byron 2	PWR	7340.4	7335.8	416.2	1032.0	83.5
Callaway	PWR	8538.2	8512.7	257.0	14.3	96.9
Calvert Cliffs 1	PWR	5760.4	5607.4	1070.2	2106.4	63.8
Calvert Cliffs 2	PWR	8657.3	8629.2	154.8	0.0	98.2
Catawba 1	PWR	6021.7	5982.8	159.5	2641.7	68.1
Catawba 2	PWR	7070.1	6976.8	833.4	973.8	79.4
Clinton 1	BWR	7907.1	7872.4	745.2	166.4	89.6
Comanche Peak 1	PWR	8288.0	8237.3	546.7	0.0	93.8
Comanche Peak 2	PWR	6908.0	6861.0	645.0	1278.0	78.1
Cook 1	PWR	7786.4	7517.8	527.6	738.6	85.6
Cook 2	PWR	7687.6	7641.9	10.6	1131.5	87.0
Cooper Station	BWR	6810.5	6683.3	0.0	2100.7	76.1
Crystal River 3	PWR	5516.3	5291.9	1259.4	2232.7	60.2
Davis-Besse	PWR	7490.1	7452.6	0.0	1331.4	84.8
Diablo Canyon 1	PWR	7096.9	6999.6	333.1	1451.3	79.7
Diablo Canyon 2	PWR	7505.6	7319.5	304.6	1159.9	83.3
Dresden 2	BWR	1821.0	1522.0	2281.0	4981.0	17.3
Dresden 3	BWR	5599.7	5297.0	3487.0	0.0	60.3
Duane Arnold	BWR	8784.0	8784.0	0.0	0.0	100.0
Farley 1	PWR	8007.3	7817.1	102.2	864.7	89.0
Farley 2	PWR	8722.7	8692.4	91.6	0.0	99.0
Fermi 2	BWR	8122.4	8068.7	634.8	80.5	91.9
FitzPatrick	BWR	8402.4	8274.0	510.0	0.0	94.2
Fort Calhoun	PWR	8211.8	8170.9	387.3	225.8	93.0
Ginna	PWR	6710.4	6648.9	422.8	1712.3	75.7
Grand Gulf	BWR	8732.2	8710.9	73.1	0.0	99.2
Haddam Neck	PWR	7104.3	7102.5	601.5	1080.0	80.9

Footnotes at end of table.

Table A-2.10 Critical, On-Line, Outage, and Availability Data for 1996

Plant Name	Reactor Type	Reactor Critical Hours	Generator On-Line Hours	Forced Outage Hours	Scheduled Outage Hours	Unit Availability Factor
Harris	PWR	7699.8	7583.5	715.4	485.1	86.3
Hatch 1	BWR	7832.3	7667.3	194.1	922.6	87.3
Hatch 2	BWR	7507.4	7407.4	64.7	1311.9	84.3
Hope Creek	BWR	5699.9	5542.0	0.0	3242.0	63.1
Indian Point 2	PWR	8300.3	8170.6	298.1	315.3	93.0
Indian Point 3	PWR	4360.1	4194.2	4589.8	0.0	47.7
Kewaunee	PWR	8514.4	8510.6	32.3	241.1	96.9
LaSalle 1	BWR	5986.5	5560.2	759.1	2464.7	63.3
LaSalle 2	BWR	7964.2	7860.0	661.0	263.0	89.5
Limerick 1	BWR	7903.6	7760.1	266.6	757.3	88.3
Limerick 2	BWR	8761.0	8684.5	99.5	0.0	98.9
Maine Yankee	PWR	5356.3	5063.6	1126.1	2594.3	57.6
McGuire 1	PWR	7752.1	7643.3	99.8	1040.9	87.0
McGuire 2	PWR	6684.9	6643.1	1194.4	946.5	75.6
Millstone 1	BWR	831.5	826.0	7031.0	927.0	9.4
Millstone 2	PWR	3319.4	3294.0	5258.0	232.0	37.5
Millstone 3	PWR	4023.3	3999.7	4418.3	366.0	45.5
Monticello	BWR	7697.4	7545.0	212.7	1026.3	85.9
Nine Mile Pt. 1	BWR	8348.2	8292.2	373.3	118.5	94.4
Nine Mile Pt. 2	BWR	8714.3	8708.8	0.0	75.2	99.1
North Anna 1	PWR	8051.4	8015.0	40.7	728.3	91.2
North Anna 2	PWR	8208.2	8202.3	30.5	551.2	93.4
Oconee 1	PWR	7881.1	7825.7	40.9	917.4	89.1
Oconee 2	PWR	7558.8	7514.1	331.6	938.3	85.5
Oconee 3	PWR	8556.1	8553.1	230.9	0.0	97.4
Oyster Creek	BWR	7793.7	7717.2	490.8	576.0	87.9
Palisades	PWR	8425.0	8367.1	384.9	32.0	95.3
Palo Verde 1	PWR	7962.7	7851.2	326.1	606.7	89.4
Palo Verde 2	PWR	7598.6	7548.7	53.0	1182.3	85.9
Palo Verde 3	PWR	7682.8	7570.0	83.5	1130.5	86.2
Peach Bottom 2	BWR	8246.4	8212.0	0.0	572.0	93.5
Peach Bottom 3	BWR	8270.3	8143.0	256.0	385.0	92.7
Perry	BWR	6654.3	6499.5	497.5	1787.0	74.0
Pilgrim	BWR	8380.8	8346.1	304.8	133.1	95.0
Point Beach 1	PWR	8219.5	8173.6	0.0	610.4	93.1
Point Beach 2	PWR	7429.6	7391.3	19.1	1373.6	84.1
Prairie Island 1	PWR	7381.4	7328.7	61.2	1394.1	83.4
Prairie Island 2	PWR	8684.2	8654.8	129.2	0.0	98.5
Quad Cities 1	BWR	3547.9	3232.0	654.7	4897.3	36.8
Quad Cities 2	BWR	5759.2	5642.0	3142.0	0.0	64.2

Footnotes at end of table.

Table A-2.10 Critical, On-Line, Outage, and Availability Data for 1996

Plant Name	Reactor Type	Reactor Critical Hours	Generator On-Line Hours	Forced Outage Hours	Scheduled Outage Hours	Unit Availability Factor
River Bend	BWR	7460.3	7338.2	498.9	946.9	83.5
Robinson 2	PWR	8231.2	8231.2	0.0	552.8	93.7
Salem 1	PWR	0.0	0.0	3672.0	5112.0	0.0
Salem 2	PWR	0.0	0.0	6624.0	2160.0	0.0
San Onofre 2	PWR	8757.6	8750.6	33.4	0.0	99.6
San Onofre 3	PWR	8620.0	8592.3	164.3	27.4	97.8
Seabrook	PWR	7898.3	7789.7	93.3	901.0	88.7
Sequoyah 1	PWR	7317.6	7073.6	400.2	1310.2	80.5
Sequoyah 2	PWR	7579.2	7469.1	81.6	1233.3	85.0
South Texas 1	PWR	8167.9	8127.9	87.1	569.0	92.5
South Texas 2	PWR	7815.4	7670.2	125.4	988.4	87.3
St. Lucie 1	PWR	6327.8	6147.6	530.8	2105.6	70.0
St. Lucie 2	PWR	6558.9	6428.6	240.4	2115.0	73.2
Summer	PWR	7928.9	7830.4	0.0	953.6	89.1
Surry 1	PWR	8333.0	8293.1	0.0	490.9	94.4
Surry 2	PWR	7464.0	7435.4	488.2	860.4	84.6
Susquehanna 1	BWR	7557.3	7507.0	102.9	1174.1	85.5
Susquehanna 2	BWR	7949.0	7838.1	437.7	508.2	89.2
Three Mile Isl 1	PWR	8509.5	8483.2	0.0	300.8	96.6
Turkey Point 3	PWR	8395.9	8266.3	318.0	199.7	94.1
Turkey Point 4	PWR	7968.1	7846.5	26.6	910.9	89.3
Vermont Yankee	BWR	8149.2	8131.4	68.9	583.7	92.6
Vogtle 1	PWR	7265.3	7190.2	411.2	1182.6	81.9
Vogtle 2	PWR	8233.0	8232.5	0.0	551.5	93.7
Wash. Nuclear 2	BWR	6236.8	5999.6	123.3	2661.1	79.7
Waterford 3	PWR	7459.7	7378.4	545.7	859.9	84.0
Watts Bar 1	PWR	2980.5	2971.1	9.4	67.5	97.5
Wolf Creek	PWR	7135.2	7080.2	159.5	1544.3	80.6
Zion 1	PWR	5658.1	5305.4	971.3	2507.3	60.4
Zion 2	PWR	8013.2	7825.6	146.8	811.6	89.1
Total All Plants		779536.2	769160.1	69450.2	111920.7	81.0
Total BWR Plants		256807.1	252408.8	25074.5	37551.7	80.4
Total PWR Plants		522729.1	516751.3	44375.7	74369.0	81.3

Reactor critical hours: Excludes pre-commercial hours. For 1996, this equals 1840.1 hours for Watts Bar 1.

Unit Availability Factor: $\frac{\text{(Generator On-Line Hours + Unit Reserve Shutdown Hours)} \times 100}{\text{Period Hours}}$

Unit Reserve Shutdown Hours: The hours the unit was removed from on-line operation for economic or other similar reasons when operation could have continued. For 1996, this equals 0 hours for all plants except Wash. Nuclear 2, which had 997.4 hours.

Period Hours: The gross hours from the beginning of the year or commercial operation, whichever comes last, to the end of the year or permanent shutdown, whichever comes first. For 1996, this equals 8784 hours for all plants except Browns Ferry 1, which had 0 hours, Browns Ferry 3, which had 7595 hours, and Watts Bar 1, which had 3048 hours.

Table A-2.11 Capacity Factors for 1996

Plant Name	Reactor Type	Net Electrical Energy (GWH)	MDC (Net MWe)	DER (Net MWe)	Capacity Factor (MDC Net)	Capacity Factor (DER Net)
Arkansas 1	PWR	6801.7	836.0	850.0	92.6	91.1
Arkansas 2	PWR	6391.0	858.0	912.0	84.8	79.8
Beaver Valley 1	PWR	5518.9	810.0	835.0	77.6	75.2
Beaver Valley 2	PWR	6357.8	820.0	836.0	88.3	86.6
Big Rock Point	BWR	379.3	67.0	72.0	64.4	60.0
Braidwood 1	PWR	6943.8	1120.0	1120.0	70.6	70.6
Braidwood 2	PWR	8006.4	1120.0	1120.0	81.4	81.4
Browns Ferry 1	BWR	0.0	0.0	1065.0	0.0	0.0
Browns Ferry 2	BWR	8191.3	1065.0	1065.0	87.6	87.6
Browns Ferry 3	BWR	7226.3	1065.0	1065.0	89.3	89.3
Brunswick 1	BWR	6350.5	767.0	821.0	94.3	88.1
Brunswick 2	BWR	4870.0	754.0	821.0	73.5	67.5
Byron 1	PWR	5138.2	1105.0	1120.0	52.9	52.2
Byron 2	PWR	8066.1	1105.0	1120.0	83.1	82.0
Callaway	PWR	9735.9	1125.0	1171.0	98.5	94.7
Calvert Cliffs 1	PWR	4554.6	835.0	845.0	62.1	61.4
Calvert Cliffs 2	PWR	7310.9	840.0	845.0	99.1	98.5
Catawba 1	PWR	6697.2	1129.0	1145.0	67.5	66.6
Catawba 2	PWR	7859.2	1129.0	1145.0	79.2	78.1
Clinton 1	BWR	7155.4	933.0	933.0	87.6	87.3
Comanche Peak 1	PWR	8795.2	1150.0	1150.0	87.1	87.1
Comanche Peak 2	PWR	7287.8	1150.0	1150.0	72.1	72.1
Cook 1	PWR	7203.2	1000.0	1020.0	82.0	80.4
Cook 2	PWR	8019.0	1060.0	1090.0	86.1	83.8
Cooper Station	BWR	4865.8	764.0	778.0	72.5	71.2
Crystal River 3	PWR	4232.5	818.0	825.0	58.9	58.4
Davis-Besse	PWR	6440.0	873.0	906.0	84.1	80.9
Diablo Canyon 1	PWR	7181.3	1073.0	1086.0	76.2	75.3
Diablo Canyon 2	PWR	7648.3	1087.0	1119.0	80.1	77.8
Dresden 2	BWR	609.8	772.0	794.0	9.0	8.7
Dresden 3	BWR	3649.5	773.0	794.0	53.7	52.3
Duane Arnold	BWR	4480.9	520.0	538.0	98.3	94.8
Farley 1	PWR	6291.2	812.0	829.0	88.2	86.4
Farley 2	PWR	7031.3	822.0	829.0	97.4	96.6
Fermi 2	BWR	6641.6	876.0	1116.0	86.3	67.8
FitzPatrick	BWR	6374.7	766.0	816.0	94.3	88.9
Fort Calhoun	PWR	3831.8	478.0	478.0	91.3	91.3
Ginna	PWR	3113.8	470.0	470.0	75.4	75.4
Grand Gulf	BWR	10586.4	1179.0	1250.0	102.7	96.4
Haddam Neck	PWR	4052.7	560.0	582.0	82.4	79.3

Footnotes at end of table

Table A-2.11 Capacity Factors for 1996

Plant Name	Reactor Type	Net Electrical Energy (GWH)	MDC (Net MWe)	DER (Net MWe)	Capacity Factor (MDC Net)	Capacity Factor (DER Net)
Harris	PWR	6353.7	860.0	900.0	84.1	80.4
Hatch 1	BWR	5579.3	805.0	822.0	82.2	80.3
Hatch 2	BWR	5919.6	809.0	784.0	84.1	86.0
Hope Creek	BWR	5524.5	1031.0	1067.0	61.0	58.9
Indian Point 2	PWR	7695.7	931.0	986.0	93.1	88.9
Indian Point 3	PWR	3916.1	965.0	965.0	46.2	46.2
Kewaunee	PWR	4308.4	511.0	535.0	96.0	91.7
LaSalle 1	BWR	5374.6	1036.0	1078.0	59.1	56.8
LaSalle 2	BWR	8055.9	1036.0	1078.0	88.5	85.1
Limerick 1	BWR	8040.7	1105.0	1105.0	84.4	84.4
Limerick 2	BWR	9640.1	1115.0	1115.0	98.4	98.4
Maine Yankee	PWR	3887.9	860.0	870.0	51.5	50.9
McGuire 1	PWR	8428.0	1129.0	1180.0	85.0	81.3
McGuire 2	PWR	7335.2	1129.0	1180.0	74.0	70.8
Millstone 1	BWR	501.6	641.0	660.0	8.9	8.7
Millstone 2	PWR	2803.3	871.0	870.0	36.6	36.7
Millstone 3	PWR	4534.9	1137.0	1154.0	45.4	44.7
Monticello	BWR	3915.5	544.0	553.0	82.6	81.2
Nine Mile Pt. 1	BWR	4786.6	565.0	613.0	96.4	88.9
Nine Mile Pt. 2	BWR	9788.5	1105.0	1143.0	100.6	97.5
North Anna 1	PWR	6951.5	893.0	907.0	88.6	87.3
North Anna 2	PWR	7368.8	897.0	907.0	93.5	92.5
Oconee 1	PWR	6585.6	846.0	886.0	88.6	84.6
Oconee 2	PWR	6304.9	846.0	886.0	84.8	81.0
Oconee 3	PWR	7276.6	846.0	886.0	97.9	93.5
Oyster Creek	BWR	4701.2	619.0	650.0	86.5	82.3
Palisades	PWR	6354.2	730.0	805.0	99.1	89.9
Palo Verde 1	PWR	9407.5	1227.0	1249.0	87.3	85.7
Palo Verde 2	PWR	9312.6	1227.0	1249.0	86.4	84.9
Palo Verde 3	PWR	9299.6	1230.0	1253.0	86.1	84.5
Peach Bottom 2	BWR	7709.0	1093.0	1119.0	80.3	78.4
Peach Bottom 3	BWR	8780.5	1093.0	1119.0	91.5	89.3
Perry	BWR	7097.4	1160.0	1191.0	69.3	67.8
Pilgrim	BWR	5353.5	670.0	655.0	91.0	93.0
Point Beach 1	PWR	4017.2	485.0	497.0	94.3	92.0
Point Beach 2	PWR	3261.5	485.0	497.0	76.6	74.7
Prairie Island 1	PWR	3707.3	513.0	530.0	82.3	79.6
Prairie Island 2	PWR	4472.4	512.0	530.0	99.4	96.1
Quad Cities 1	BWR	2273.9	769.0	789.0	33.7	32.8
Quad Cities 2	BWR	4115.3	769.0	789.0	60.9	59.4

Footnotes at end of table.

Table A-2.11 Capacity Factors for 1996

Plant Name	Reactor Type	Net Electrical Energy (GWH)	MDC (Net MWe)	DER (Net MWe)	Capacity Factor (MDC Net)	Capacity Factor (DER Net)
River Bend	BWR	6654.4	936.0	936.0	80.9	80.9
Robinson 2	PWR	5847.5	683.0	700.0	97.5	95.1
Salem 1	PWR	-28.2	1106.0	1115.0	0.0	0.0
Salem 2	PWR	-40.2	1106.0	1115.0	0.0	0.0
San Onofre 2	PWR	9360.0	1070.0	1070.0	99.6	99.6
San Onofre 3	PWR	9188.3	1080.0	1080.0	96.9	96.9
Seabrook	PWR	8674.0	1158.0	1148.0	85.4	86.0
Sequoyah 1	PWR	7690.8	1111.0	1148.0	78.8	76.3
Sequoyah 2	PWR	8356.6	1106.0	1148.0	86.0	82.9
South Texas 1	PWR	10071.8	1251.0	1251.0	91.7	91.7
South Texas 2	PWR	9489.9	1251.0	1251.0	86.4	86.4
St. Lucie 1	PWR	4883.5	839.0	830.0	66.3	67.0
St. Lucie 2	PWR	5251.1	839.0	830.0	71.3	72.0
Summer	PWR	7066.3	945.0	954.0	88.6	87.2
Surry 1	PWR	6677.5	801.0	788.0	95.1	96.5
Surry 2	PWR	5988.8	801.0	788.0	85.1	86.5
Susquehanna 1	BWR	7975.1	1090.0	1100.0	83.3	82.5
Susquehanna 2	BWR	8478.5	1094.0	1100.0	88.2	87.7
Three Mile Isl 1	PWR	6832.3	786.0	819.0	99.0	95.0
Turkey Point 3	PWR	5478.1	666.0	693.0	93.6	90.0
Turkey Point 4	PWR	5171.4	666.0	693.0	88.4	85.0
Vermont Yankee	BWR	4193.2	510.0	522.0	93.6	91.4
Vogtle 1	PWR	8184.3	1162.0	1169.0	80.2	79.7
Vogtle 2	PWR	9511.5	1162.0	1169.0	93.2	92.6
Wash. Nuclear 2	BWR	5329.7	1107.0	1153.0	54.9	52.6
Waterford 3	PWR	7995.8	1075.0	1104.0	84.7	82.5
Watts Bar 1	PWR	3203.6	1095.0	1160.0	96.0	90.6
Wolf Creek	PWR	8209.4	1163.0	1170.0	80.1	79.9
Zion 1	PWR	5320.0	1040.0	1040.0	58.2	58.2
Zion 2	PWR	7525.8	1040.0	1040.0	82.4	82.4
Total All Plants		681172.6	909.3	930.0	78.8	77.1
Total BWR Plants		211169.9	858.9	887.0	78.0	75.6
Total PWR Plants		470002.7	934.2	951.2	79.2	77.7

MDC and DER	September 1996 values.
Total All and BWR Plants	Excludes the Browns Ferry Units' administrative hold periods.
Capacity Factor	$\frac{\text{Net Electrical Energy x 100,000}}{\text{Period Hours x MDC Net}}$ or $\frac{\text{Net Electrical Energy x 100,000}}{\text{Period Hours x DER Net}}$
Period Hours	The gross hours from the beginning of the year or commercial operation, whichever comes last, to the end of the year or permanent shutdown, whichever comes first. For 1996, this equals 8784 hours for all plants except Browns Ferry 1, which had 0 hours, Browns Ferry 3, which had 7595 hours, and Watts Bar 1, which had 3048 hours.

Table A-2.12 Industry Critical, On-Line, Outage, Availability, and Capacity Data

Industry Data	CY92	CY93	CY94	CY95	1996
Period Hours	946645.4	932176.0	937320.0	938340.0	950531.0
Reactor Critical Hours	734404.5	724323.0	751614.2	778247.3	779536.2
Generator On-Line Hours	720477.6	713214.3	741181.1	766413.7	769160.1
Unit Reserve Shutdown	1764.2	4.0	12.9	578.1	997.4
Forced Outage Hours	57559.9	66907.7	74847.2	48276.9	69450.2
Scheduled Outage Hours	168607.9	152054.0	121291.7	123649.4	111920.7
Net Electrical Energy (GWH)	619888.6	610686.6	641725.7	674087.4	681172.6
Average MDC (Net MWe)	898.1	902.5	903.4	905.6	909.3
Average DER (Net MWe)	918.3	922.4	924.4	926.4	930.0
Availability Factor	76.3	76.5	79.1	81.7	81.0
Capacity Factor (MDC Net)	72.9	72.6	75.8	79.3	78.8
Capacity Factor (DER Net)	71.3	71.0	74.1	77.5	77.1

Industry Data	Excludes the Browns Ferry Units' administrative hold periods.
Period Hours	The gross hours from the beginning of the year or commercial operation, whichever comes last, to the end of the year or permanent shutdown, whichever comes first.
Unit Reserve Shutdown Hours	The hours the unit was removed from on-line operation for economic or other similar reasons when operation could have continued.
Net Electrical Energy (GWH)	Gross electrical output of the unit measured at the output terminals of the turbine generator during the reporting period, minus the normal station service electrical energy utilization. Negative quantities should not be used. The unit of measurement for this table is gigawatt-hours.
Maximum Dependable Capacity (MDC Net) (Net MWe)	Dependable main-unit gross capacity, winter or summer, whichever is smaller, less the normal station service loads. The dependable capacity varies because the unit efficiency varies during the year due to cooling water temperature variations. It is the gross electrical output as measured at the output terminals of the turbine generator during the most restrictive seasonal conditions, less the normal station service loads. The unit of measurement for this table is megawatts.
Design Electrical Rating plant (DER Net) (Net MWe)	The nominal net electrical output of the unit specified by the utility and used for the purpose of design. The unit of measurement for this table is megawatts.
Availability Factor	$\frac{\text{(Generator On-Line Hours + Unit Reserve Shutdown Hours) x 100}}{\text{Period Hours}}$
Capacity Factor (MDC Net)	$\frac{\text{Net Electrical Energy x 100,000}}{\text{Period Hours x MDC Net}}$
Capacity Factor (DER Net)	$\frac{\text{Net Electrical Energy x 100,000}}{\text{Period Hours x DER Net}}$

Table A-2.13. Allegations at Commercial Nuclear Plant Sites for CY 1992 Through CY 1996

Site	1992				1993				1994				1995				1996			
	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I
Arkansas 1, 2	10	0	8	2	5	0	5	0	4	0	1	1	6	0	3	2	3	0	3	2
Beaver Valley 1, 2	5	0	2	1	12	0	2	1	6	0	3	0	8	0	3	4	4	2	0	1
Big Rock Point	2	0	0	0	2	0	2	1	1	0	0	0	1	0	1	0	0	0	0	0
Braidwood 1, 2	5	0	2	1	3	0	1	0	11	3	3	5	2	0	3	1	7	1	3	1
Browns Ferry 1, 2, 3	20	0	6	8	37	0	11	7	25	1	9	5	9	0	8	2	20	6	9	5
Brunswick 1, 2	14	0	1	1	23	1	6	7	15	0	6	0	10	1	9	2	5	4	6	2
Byron 1, 2	4	0	3	3	3	0	0	0	6	0	0	1	6	0	2	0	3	1	2	1
Callaway	1	0	0	0	5	0	2	1	2	0	1	1	3	0	1	1	6	3	4	1
Calvert Cliffs 1, 2	7	0	5	2	7	0	3	1	5	0	2	0	8	0	1	3	10	1	0	3
Catawba 1, 2	1	0	0	0	2	2	0	0	8	0	1	0	2	0	3	0	5	3	1	0
Clinton 1	10	0	4	3	5	0	1	0	22	0	5	1	12	0	10	1	13	12	3	2
Comanche Peak 1, 2	25	0	12	7	26	0	15	1	5	0	3	1	10	0	4	1	16	5	4	2
Cook 1, 2	3	0	1	1	4	0	0	1	11	0	7	3	4	0	2	1	5	2	3	1
Cooper Station	6	0	4	2	3	0	1	0	11	0	3	2	10	1	4	3	17	7	9	1
Crystal River 3	14	0	2	3	6	0	3	4	3	0	0	1	4	2	1	2	19	12	7	5
Davis-Besse	9	0	2	1	9	0	2	2	7	0	4	0	5	0	2	1	12	2	5	2
Diablo Canyon 1, 2	5	0	3	2	13	0	2	2	9	0	3	2	18	2	11	2	23	10	5	4
Dresden 2, 3	8	0	1	1	13	0	9	4	12	0	4	0	14	1	4	1	8	6	6	2
Duane Arnold	5	0	1	2	3	0	1	0	1	0	3	0	3	0	1	1	1	1	0	1
Farley 1, 2	1	0	0	0	1	0	0	0	2	0	0	0	1	0	0	0	5	0	0	0
Fermi 2	8	0	3	0	10	0	4	4	14	0	0	2	12	2	6	4	12	5	4	4
FitzPatrick	9	0	2	0	7	0	5	0	3	0	0	0	8	0	3	2	5	3	2	1
Fort Calhoun	4	0	3	1	6	0	0	1	6	0	2	1	5	0	0	2	3	2	3	1
Ginna	0	0	1	0	0	0	0	0	1	0	0	0	2	0	2	0	4	2	1	0
Grand Gulf	2	0	1	0	3	0	2	0	3	0	0	1	2	0	3	0	8	3	1	3
Haddam Neck	2	0	1	1	10	0	3	3	12	0	1	3	9	0	1	1	21	15	1	6
Harris	3	0	1	0	1	0	0	0	5	0	0	1	2	0	1	0	3	0	1	0
Hatch 1, 2	7	0	0	0	7	0	2	1	4	0	1	0	8	0	2	0	4	1	4	0
Hope Creek	5	0	1	1	4	0	4	0	8	1	0	2	9	0	3	3	25	12	4	2
Indian Point 2	3	0	0	0	8	0	1	0	1	0	1	0	3	0	2	2	6	4	0	1
Indian Point 3	9	0	5	0	14	0	3	1	25	0	10	5	11	1	3	5	5	3	2	1
Kewaunee	0	0	0	0	1	0	1	0	0	0	0	0	1	0	0	0	0	0	0	0
LaSalle 1, 2	7	0	0	2	6	0	5	0	14	0	4	2	10	0	5	0	10	4	4	3
Limerick 1, 2	7	0	1	1	17	0	1	0	16	0	10	2	5	0	3	0	5	3	2	0
Maine Yankee	4	0	0	0	1	0	1	0	9	1	1	0	6	0	1	2	18	9	0	2

Footnotes at end of table

Table A-2.13. Allegations at Commercial Nuclear Plant Sites for CY 1992 Through CY 1996

Site	1992				1993				1994				1995				1996			
	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I
McGuire 1, 2	3	0	0	0	0	0	0	0	8	0	0	4	2	0	0	0	5	1	2	0
Millstone 1, 2, 3	51	1	59	17	35	1	8	7	42	3	22	13	48	4	22	15	76	50	10	27
Monticello	1	0	0	0	3	0	1	0	1	0	1	0	0	0	0	0	6	2	0	0
Nine Mile Pt. 1, 2	18	0	0	0	11	0	6	2	8	1	3	2	9	1	0	4	16	5	3	1
North Anna 1, 2	6	0	0	2	12	0	3	1	2	0	1	0	7	0	0	0	6	2	3	1
Oconee 1, 2, 3	1	0	1	0	1	0	0	0	4	0	1	0	1	0	0	1	7	3	2	1
Oyster Creek	3	0	0	0	9	0	2	1	2	0	0	0	5	0	2	1	5	3	3	2
Palisades	1	0	0	0	7	0	2	1	4	0	3	0	10	0	2	1	8	1	1	0
Palo Verde 1, 2, 3	58	1	10	19	52	0	5	21	23	0	11	11	22	3	15	5	20	9	10	4
Peach Bottom 2, 3	7	0	1	0	7	0	2	1	7	0	3	2	3	0	2	0	4	3	1	0
Perry	6	0	1	0	8	0	3	0	9	0	2	4	2	0	2	0	9	5	3	4
Pilgrim	22	0	5	0	16	0	4	2	14	0	1	2	9	2	2	3	4	1	0	0
Point Beach 1, 2	2	0	0	0	3	0	0	1	1	0	2	0	4	0	2	0	3	0	2	1
Prairie Island 1, 2	7	0	3	0	4	0	2	1	5	0	3	0	3	0	1	1	5	1	0	0
Quad Cities 1, 2	11	0	1	1	6	0	2	1	10	0	6	1	6	0	2	0	12	7	6	2
River Bend	19	0	5	3	21	0	14	6	17	0	11	1	13	2	6	10	22	4	11	7
Robinson 2	5	0	1	0	4	0	2	3	1	0	0	0	5	1	1	1	1	0	3	0
Salem 1, 2	5	0	2	2	11	0	4	2	20	3	1	5	13	2	5	5	38	20	7	7
San Onofre 2, 3	6	0	1	2	12	0	0	3	14	1	2	6	28	2	14	3	32	14	10	6
Seabrook	16	0	2	0	6	0	0	0	4	0	4	1	4	0	0	0	2	1	0	0
Sequoyah 1, 2	30	0	15	6	24	1	6	7	16	1	4	6	20	2	5	3	27	12	11	7
South Texas 1, 2	14	0	9	8	43	0	16	13	24	1	13	9	16	2	5	7	20	5	8	6
St. Lucie 1, 2	7	0	2	1	3	0	0	0	7	0	0	1	14	0	5	2	48	14	22	5
Summer	3	0	1	0	2	0	1	0	2	0	0	0	3	0	1	0	8	1	6	1
Surry 1, 2	7	0	2	2	3	0	2	1	6	0	2	0	3	0	1	0	4	1	3	0

Footnotes at end of table

Table A-2.13. Allegations at Commercial Nuclear Plant Sites for CY 1992 Through CY 1996

	1992				1993				1994				1995				1996			
Site	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I	Rc'd	Open	Sub	H&I
Susquehanna 1, 2	9	0	2	1	8	0	5	2	7	0	0	0	5	0	4	1	34	17	9	1
Three Mile Isl 1	6	0	0	0	7	0	2	0	3	0	2	0	2	0	1	0	2	2	0	1
Turkey Point 3, 4	14	0	4	1	14	0	5	5	7	0	4	0	9	0	2	0	19	5	11	2
Vermont Yankee	8	0	1	1	6	0	3	0	1	0	5	0	5	0	2	1	10	8	1	1
Vogtle 1, 2	10	0	2	2	7	0	3	5	11	3	1	0	8	0	4	0	7	0	5	0
Wash. Nuclear 2	5	0	2	0	12	0	1	6	0	0	2	0	18	3	5	3	13	6	7	4
Waterford 3	4	0	2	2	5	0	3	2	3	0	2	0	9	0	2	3	25	9	15	8
Watts Bar 1	65	0	38	27	67	0	50	19	73	0	29	19	47	4	42	16	38	10	27	7
Wolf Creek	5	0	3	1	6	0	3	0	10	0	5	0	5	0	5	1	12	5	2	0
Zion 1, 2	11	0	4	4	3	0	3	0	17	0	2	3	10	0	4	2	12	5	5	0

Rc'd: The total number of allegations received during the year. Each allegation may contain more than one concern.
Open: The number of allegations received during the year with one or more concerns remaining open.
Sub: The number of allegations fully or partially substantiated for that year. Partially substantiated means that not all the concerns were substantiated.
H&I: The number of allegations that include harassment and intimidation issues without regard to whether they are substantiated.
The data are current as of September 30, 1996.

APPENDIX B

Summary of 1996 Abnormal Occurrences

CONTENTS

NUREG-0090, Vol. 19

Report No. 96-1 Plant Trip With Multiple Complications at Wolf Creek Nuclear Generating Station

On January 30, 1996, at Wolf Creek Nuclear Generating Station, one train of the essential service water system (ESWS) became inoperable when frazil ice blocked the suction bay trash racks, and the second train was degraded.

At approximately 2:00 a.m., operators received alarms indicating that the traveling screens for the circulating water (CW) system were becoming blocked. The site watch reported that the traveling screens for Bays 1 and 3 were frozen and that water levels in these bays were approximately 8 feet below normal. The ESWS was started with the intent to separate it from the service water (SW) system. However, the ESWS was incorrectly aligned, which reduced warming flow to the ESWS suction bays. At approximately 3:30 a.m., operators received a service water low pressure alarm because the bays were 12 feet below normal, and an electric fire pump started. The shift supervisor then directed a manual reactor/turbine trip. Following the scram, five control rods failed to fully insert. The event was further complicated because the turbine-driven auxiliary feedwater pump developed a packing leak and was declared inoperable. The loss of CW bay level was subsequently determined to be caused by ice blockage of the traveling screens, which was caused by freezing water from the spray wash system.

The ESWS Train A pump was tripped and declared inoperable at 7:47 a.m. due to low discharge pressure and high strainer differential pressure. At about 5:45 p.m. the operators declared Train A operable based on an engineering evaluation. However, the A pump was stopped again at approximately 7:30 p.m. when the pump exhibited further oscillations in flow and pressure. At approximately 8:00 p.m. operators noted that the ESWS Train B bay level was 15 feet below normal and decreasing slowly. Operators placed additional heat loads on Train B and the bay level subsequently recovered. At 10:14 p.m. the operators again started the Train A ESWS pump but secured it at 10:27 p.m. due to decreasing flow and pressure. At about 9:00 a.m. on January 31, 1996, divers inspected the suction bay of Train A and noted complete blockage of the trash racks by frazil ice. The condition of the Train B trash racks was not determined because the pump was running. The ice blockage was cleared later that day using heating and air sparging of the trash racks.

This event was caused by deficiencies in the ESWS warming line design, which was exacerbated by the initial incorrect alignment of the ESWS. A 1976 design calculation specified a warming line flow rate of 4000 g.p.m. to prevent frazil ice. This calculation assumed a warming line temperature of 3 °F above freezing, but the assumption was never validated. The warming line temperature during the event was approximately 1 °F above freezing. Additionally, due to the elevations and configuration of the warming line, portions of the line operated with partial pipe flows. Flow through the lines was estimated to have been 2500 g.p.m. and warming flow was estimated to be 1700 g.p.m., which was less than half the design specification. To prevent recurrence, the licensee changed the hydraulics of the ESWS discharge to the ultimate heat sink, and the warming line to the ESWS pumphouse to establish and distribute the proper amount of flow to the ESWS warming line.

The NRC conducted an Augmented Inspection Team inspection and issued a civil penalty of $300,000.

Report No. 96-2 Containment-Bypass Leakage via Disconnected Hydrogen-Monitor Lines at Braidwood Units 1 and 2

On November 9, 1994, the licensee for Braidwood Unit 2 completed a containment integrated leak rate test (ILRT). For this test, the 0.25-inch containment penetration hydrogen sensing lines for trains "A" and "B" were disconnected and a balloon placed on the end of each line to identify any leakage. The procedure did not specify whether to disconnect the sensing line inside the hydrogen monitor cabinet or outside. The operators who lined up the test disconnected the lines inside the cabinet. The licensee's investigation concluded that when other operators restored the system from the test, they looked at the exterior sensing lines and assumed that the lines were reconnected. Therefore, the sensing lines remained disconnected inside the cabinet.

On January 31, 1995, the operations department wrote a problem identification report on the growing difference between the hydrogen readings on the "A" and "B" trains which are taken during each shift. During troubleshooting on February 15, 1995, the "A" train lines were found to be disconnected. Surveillance tests performed on December 11, 1994, and January 25, 1995, provided missed opportunities to detect the deficiency with the "A" train. It could not be conclusively determined when the "B" train was restored. Two maintenance workers had a recollection of discovering balloons on the sensing lines in a hydrogen monitoring cabinet in late 1994. Maintenance records indicate these individuals worked on the "B" train on December 20, 1994. However, computer and operator logs for the "B" train appear to have been accurately reading containment hydrogen following the ILRT.

The hydrogen monitors are normally isolated. However, during a loss of coolant accident, the Emergency Operating Procedures direct the operators to put them into service to monitor containment hydrogen concentration. This would create an unfiltered release path from the containment to the auxiliary building. The licensee calculated that regulatory dose limits could be exceeded within approximately 3 hours. NRC review found the licensees calculations to be conservative. There are area radiation monitors near the hydrogen monitors. These area radiation monitors alarm in the control room and the alarm response procedures call for notification of Radiation Protection personnel to survey the area. Additionally, there are radiation monitors in the auxiliary building exhaust that would assist the operators in identifying the leak. The containment bypass flow path could be isolated remotely from the control room and it appears credible that the leak could be isolated prior to exceeding regulatory limits.

The cause of this event was a procedural deficiency in that the ILRT procedure did not provide adequate guidance on where the containment penetration hydrogen sensing lines should be disconnected. Additionally, the operator tasked with reconnecting the containment penetration hydrogen sensing lines, after the ILRT was completed, did not display a questioning attitude when he found that the lines appeared to be reconnected. To prevent recurrence, the licensee revised its ILRT line up and restoration sheets to provide adequate guidance on where disconnections and connections are to be performed. Additionally, a General Information Notice was issued to all site personnel highlighting the human performance problems identified from this event.

The NRC exercised escalated enforcement action and the licensee was assessed a $100,000 civil penalty. Information Notice 96-13, "Potential Containment Leak Paths Through Hydrogen Analyzers," was also issued to alert other licensees to this event.

APPENDIX C

Reports Issued in 1996

Table C-1 Reports Issued in 1996

Date	Title	No.	Author
	Special Studies		
12/95	Precursors to Potential Severe Core Damage Accidents: 1994 A Status Report	NUREG/CR-4674 Vols. 21 and 22	
01/96	Performance Indicators for Operating Commercial Nuclear Power Reactors Data: Through September 1995		
01/96	Evidence of Aging Effects on Certain Safety-Related Components	NUREG/CR-6442	
08/96	Office for Analysis and Evaluation of Operational Data – Annual Report, 1994-FY 95	NUREG-1272, Vol. 9, No. 1	
08/96	Isolation Condenser System Reliability, 1987–1993	S96-01 (INEL-95/0478)	
09/96	Assessment of Spent Fuel Cooling	S96-02	J. Ibarra W. Jones G. Lanik H. Ornstein S. Pullani
02/96	Emergency Diesel Generator Power System Reliability 1987–1993	S96-03 (INEL-95/0035)	
	Engineering Evaluations		
03/96	Motor-Operated Valve Key Failures	E96-01	C. Hsu
04/96	Analysis of Allegation Data	E96-02	S. Israel
06/96	Analysis of Allegation Data	E96-02 Supplement 1	S. Israel
04/96	Steam Generator Tube Failures	NUREG/CR-6365 (E96-03)	

Table C-1 Reports Issued in 1996 (cont.)

Date	Title	No.	Author
	Technical Reviews		
10/95	Potential Damage to Low-Pressure Injection Valves During Surveillance Testing	T95-02	E. Brown
10/95	Review of the National Transportation Safety Board's Safety Study NTSB/SS-94/01, "A Review of Flightcrew-Involved, Major Accidents of U.S. Carriers,1978 Through 1990"	T95-03	J. Kauffman
03/96	Technical Review Report – AEOD Technical Reports by Category	T96-01	S. Israel
03/96	Technical Review Report – AEOD Technical Reports by Category	T96-01 Revision 1	S. Israel
04/96	Technical Review Report – Target Rock Two-Stage SRV Performance Update	T96-02	M. Wegner
08/96	Technical Review Report – Response of Babcock & Wilcox Company Plants Following a Loss of Nonemergency AC Power	T96-03	W. Raughley

APPENDIX D

Reports Issued From CY 1980 Through CY 1995

Table D-1 Reports Issued in CY 1995

Date	Title	No.	Author
	Special Studies		
02/95	High-Pressure Coolant Injection (HPCI) System Performance, 1987–1993 Final	S95-02	
03/95	Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994	S95-01	J. Kauffman S. Israel
	Engineering Evaluations		
07/95	Operating Events With Inappropriate Bypass or Defeat of Engineered Safety Features	E95-01	J. Kauffman
	Technical Reviews		
03/95	Major Disturbances on the Western Grid and Related Events	T95-01	M. Wegner

Table D-2 Reports Issued in CY 1994

Date	Title	No.	Author
	Special Studies		
11/94	Office for Analysis and Evaluation of Operational Data 1993 Annual Report,	NUREG-1272, Vol. 8, No. 1 Power Reactors	
10/94	Operating Experience Feedback Report – Reliability of Safety-Related Steam Turbine-Driven Standby Pumps	NUREG-1275, Vol. 10 (S94-01)	J. Boardman
09/94	Operating Experience Feedback Report – Turbine-Generator Overspeed Protection Systems	NUREG-1275, Vol. 11 (S94-02)	H. Ornstein
	Technical Reviews		
03/94	The Electrical Transient Which Followed the Los Angeles Earthquake – January 17, 1994	T94-01	M. Wegner
05/94	Review of Mispositioned Equipment Events	T94-02	S. Israel
07/94	Computer-Based Digital System Failures	T94-03	E. Lee
12/94	Potential for Boiling Water Reactor Emergency Core Cooling System Strainer Blockage Due to Loss-of-Coolant Accident Generated Debris	T94-04	J. Boardman

Table D-3 Reports Issued in CY 1993

Date	Title	No.	Author
	Special Studies		
07/93	Office for Analysis and Evaluation of Operational Data 1992 Annual Report, Power Reactors	NUREG-1272, Vol. 7, No. 1	
04/93	Review of Auxiliary Feedwater System Reliability	S93-01	J. Houghton D. Rasmuson J. Boardman
	Not issued	S93-02	
04/93	Operating Experience Feedback – Service Water System Failures and Degradations	S93-03	J. Houghton
	Not issued	S93-04	
04/93	Operational Data Analysis of Shutdown and Low Power Licensee Event Reports	S93-05	R. Prato
12/93	Potter & Brumfield Model MDR Rotary Relay Failures	S93-06	R. Spence
	Engineering Evaluations		
02/93	Insights from Common-Mode Failure Events	E92-02 Supplement 1	S. Israel
02/93	Human Factors Aspects of Boiling Water Reactor Reactivity Management Events During Power Operations	E93-01	J. Kauffman
03/93	Evaluation of Loss of Offsite Power Due to Plant-Centered Events	E93-02	S. Mazumdar
12/93	Electrical Inverter Operating Experience 1985-1992	E93-03	J. Ibarra
	Technical Reviews		
06/93	Primary System Integrity, Pressurized Water Reactor Coolant System Leaks	T93-01	J. Kauffman J. Stuller
08/93	Tardy Licensee Actions	T93-02	S. Israel
12/93	Loss of Annunciator and Computer System Events	T93-03	J. Ibarra
12/93	U.S. Nuclear Regulatory Commission Review of Operating Experience	T93-04	H. Ornstein

Table D-4 Reports Issued in CY 1992

Date	Title	No.	Author
	Case Studies		
12/92	Operating Experience Feedback Report – Human Performance in Operating Events	NUREG-1275, Vol. 8 (C92-01)	J. Kauffman G. Lanik R. Spence E. Trager
	Special Studies		
07/92	Office for Analysis and Evaluation of Operational Data 1991 Annual Report, Power Reactors	NUREG-1272, Vol. 6, No. 1	
	Not issued	S92-01	
04/92	Safety and Safety/Relief Valve Reliability	S92-02	M. Wegner
06/92	Review of Operational Experience with Molded Case Circuit Breakers in U.S. Commercial Nuclear Power Plants	S92-03	J. Houghton W. Leschek P. O'Reilly D. Rasmuson
	Not issued	S92-04	
	Not issued	S92-05	
	Not issued	S92-06	
09/92	Operating Experience Feedback Report – Experience with Pump Seals Installed in Reactor Coolant Pumps Manufactured by Byron Jackson	NUREG-1275, Vol. 7	
12/92	Operating Experience Feedback Report – Pressure Locking and Thermal Binding of Gate Valves	NUREG-1275, Vol. 9 (S92-07)	C. Hsu
	Engineering Evaluations		
05/92	Inadequate Management Control of Snubber Surveillance	E92-01	C. Hsu
06/92	Insights From Common-Mode Failure Events	E92-02	S. Israel

Table D-4 Reports Issued in CY 1992 (cont.)

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

Table D-5 Reports Issued in CY 1991

Date	Title	No.	Author
	Special Studies		
07/91	Office for Analysis and Evaluation of Operational Data 1990 Annual Report, Power Reactors	NUREG-1272, Vol. 5, No. 1	
09/91	Performance of Emergency Diesel Generators in Restoring Power to Their Associated Safety Buses–A Review of Events Occurring at Power	S91-01	T. Cintula
	Engineering Evaluations		
02/91	A Review of Water Hammer Events After 1985	E91-01	E. Brown
	Technical Reviews		
02/91	Causes of Incorrect System Flows	T91-01	S. Israel
02/91	Incorrect Rotation of PDP	T91-02	T. Cintula
03/91	Overloaded Emergency Buses	T91-03	S. Israel
04/91	Turbine Overspeed Trip Due to Steam Valve Leakage and Condensate	T91-04	C. Hsu
05/91	Setpoint Testing of Pressurizer Safety Valves With Water-Filled Loop Seals	T91-05	M. Wegner
06/91	Deficiencies in External Flood Protection	T91-06	S. Israel
07/91	Evaluation of Partial Loss of Station Power Events at Prairie Island Unit No. 2 on December 21 and December 26, 1989	T91-07	F. Manning

Table D-6 Reports Issued in CY 1990

Date	Title	No.	Author
	Case Studies		
10/90	Operating Experience Feedback Report – Solenoid-Operated Valve Problems at U.S. Light Water Reactors	NUREG-1275, Vol. 6 (C90-01)	H. Ornstein
	Special Studies		
07/90	Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Power Reactors	NUREG-1272, Vol. 4, No. 1	
03/90	Review of Thermal Stratification Operating Experience	S902	T. Su
08/90	Recurrence of Important Safety Issues Reported in LERs	S90-01	S. Israel
	Engineering Evaluations		
02/90	Failures of Electrical Supply and Power Generation Equipment Which Disrupted Plant Function at Nuclear Power Plants	E90-01	M. Wegner
02/90	Crosby Low Pressure Relief Valves	E90-02	S. Israel
05/90	Overpressurization of Auxiliary Feedwater Systems	E90-03	C. Hsu
04/90	Swelling and Cracking in Hafnium Control Rods	E90-04	M. Wegner
05/90	Operational Experience on Bus Transfer	E90-05	S. Mazumdar
07/90	Potential for Residual Heat Removal System Pump Damage	E90-06	C. Hsu
07/90	Effects of Internal Flooding of Nuclear Power Plants on Safety Equipment	E90-07	T. Su
09/90	Low Temperature Overpressure Protection: Testing PORVs With the Alternate Pneumatic Supply	E90-08	S. Israel S. Salah
10/90	Additional Factors Affecting the Lift Setpoint of Pressurizer Safety Valves	E90-09	L. Padovan
12/90	Evaluation of Boiling Water Reactor Mode Switch Events	E90-10	W. Jones

Table D-6 Reports Issued in CY 1990 (cont.)

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

Table D-7 Reports Issued in CY 1989

Date	Title	No.	Author
	Special Studies		
06/89	Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Power Reactors	NUREG-1272, Vol. 3, No. 1	
03/89	Operating Experience Feedback Report – Technical Specifications	NUREG-1275, Vol. 4	P. O'Reilly G. Plumlee
03/89	Operating Experience Feedback Report – Progress in Scram Reduction	NUREG-1275, Vol. 5	L. Bell P. O'Reilly
08/89	Operating Experience Feedback Report – Progress in Scram Reduction	NUREG-1275, Vol. 5 Addendum	L. Bell
01/89	Application of the NPRDS for Effectiveness Monitoring (Appendices A and B are Proprietary)	S804B	P. O'Reilly T. Wolf P. Cross-Prather
02/89	Maintenance Programs at Nuclear Power Plants (Table 2 is Proprietary)	S901 Revision 1	M. Chiramal S. Israel M. Wegner S. Stern
	Engineering Evaluations		
02/89	Problems With Oils, Greases, Solvents and Other Chemical Materials	E901	S. Israel
03/89	Fire and Explosive Mixtures Resulted From Introduction of Hydrogen Into Plant Air Systems	E902	H. Ornstein
	Not issued	E903	
04/89	On Demand Malfunctions of HPCI and RCIC	E904	T. Cintula
06/89	Electrical Bus Bar Failures	E905	M. Padovan
08/89	Failure of Steam Generator Isolation Check Valve	E906	T. Cintula
09/89	Diversion of Seal Cooler Flow for RHR Pumps	E907	S. Israel
10/89	Excessive Valve Body Erosion at Brunswick	E908	E. Brown
12/89	Operator Actions During Operational Events	E909	S. Israel

Table D-7 Reports Issued in CY 1989 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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 Reviews		
01/89	Millstone Unit 1-Safety/Relief Valve Discharge Line Vacuum Breakers Failed Open	T901	T. Su
02/89	Inadvertent Reactor Trips Due to RCS Flow Instrumentation Maintenance Activities	T902	M. Padovan
03/89	Generic Implication of Browns Ferry Fire on November 2, 1987	T903	T. Su
04/89	Design Deficiency of Safety Injection Block Switch	T904	S. Mazumdar
04/89	Failure of 4160V GE Magneblast Breaker to Trip Open	T905	S. Mazumdar
04/89	Broken Lifting Beam Bolts in HPCI Terry Turbine	T906	T. Cintula
04/89	Component Degradation Due to Indiscriminate Painting	T907	M. Padovan
	A nonreactor report (see NUREG-1272, Vol.4, No. 2)	T908	
05/89	Operating Events Involving Dampers	T909	S. Israel
06/89	Investigation of Cracked Control Rod Drive Seal Housings at Palisades	T910	W. Jones
06/89	Evaluation of Individually Reported Safety System LERs for Their Combined Significance	T911	F. Manning
06/89	Selected Maintenance Rework	T912	S. Israel
07/89	Comparison of the Proposed Maintenance Effectiveness (ME) Indicator With Catawba and Farley Nuclear Plants Regarding Inspections (Proprietary)	T913	N. Thomasson T. Wolf M. Harper
09/89	Overview of Design/Installation Fabrication Errors in 1988	T914	S. Israel

Table D-7 Reports Issued in CY 1989 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
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-8 Reports Issued in CY 1988

Date	Title	No.	Author
	Case Studies		
08/88	Operating Experience Feedback Report – Service Water System Failures and Degradations in Light Water Reactors	NUREG-1275, Vol. 3	P. Lam E. Leeds
	Special Studies		
03/88	Significant Events That Involved Procedures	S801	E. Trager
03/88	Operational Experience Feedback Evaluation Rancho Seco Nuclear Generating Station, Restart	S802	G. Plumlee
06/88	AEOD Concerns Regarding the Power Oscillation Event at LaSalle 2 (BWR-5)	S803	J. Kauffman
08/88	Preliminary Results of the Trial Program for Maintenance Performance Indicators	S804A	
09/88	Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data-1987 Power Reactors	NUREG-1272 Vol. 2, No. 1 (S804)	
	Not issued	S805	
	Not issued	S806	
	A nonreactor report (see NUREG-1272, Vol. 3, No. 2)	S807	
	Engineering Evaluations		
04/88	BWR Overfill Events Resulting in Steam Line Flooding	E801	J. Kauffman
05/88	Design and Operating Deficiencies in Control Room Emergency Ventilation Systems	E802	S. Israel
08/88	Inadequate NPSH in High Pressure Safety Injection Systems in PWRs	E803	S. Israel
08/88	Reliability of Recirculation Pump Breaker During an ATWS	E804	T. Su
09/88	Potential LOCA Due to Energized Uncovered Pressurizer Heaters	E805	T. Cintula
10/88	Loss of Decay Heat Removal Due to Rapid Refueling Cavity Pumpdown	E806	M. Padovan

Table D-8 Reports Issued in CY 1988 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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 Reviews		
01/88	Perry Nuclear Power Plant Unit 1-Unexpected MSIVs Closure and Reopening	T801	T. Su
	Not issued	T802	
05/88	Summary of Early Operational Experience of Foreign Commercial Nuclear Reactors (Proprietary)	T803	P. O'Reilly
05/88	"Precursor" Operational Events That Occurred From November 1, 1987, Through March 1988	T804	F. Manning
05/88	Insights From Significant Events in 1987	T805	S. Israel
05/88	Recent Operational Experience Trends at Fermi 2	T806	T. Wolf
06/88	Recent Operational Experience Trends at Indian Point 2	T807	T. Wolf
06/88	A Technical Basis for Granting Test Frequency Relief	T808	G. Plumlee
06/88	Blocked Thimble Tubes/Stuck Incore Detector	T809	M. Wegner
07/88	An Analysis of NPRDS Data for Hatch Plant (Proprietary)	T810	T. Wolf P. Cross-Prather
11/88	Degradation of Ice Condenser Containment Functional Capability	T811	F. Manning
	Incident Investigation Program Reports		
02/88	Incident Investigation Manual	NUREG-1303	

Table D-9 Reports Issued in CY 1987

Date	Title	No.	Author
	Case Studies		
03/87	Operating Experience Feedback Report – Air Systems Problems at U.S. Light Water Reactors	NUREG-1275, Vol. 2 (C701)	H. Ornstein
	Special Studies		
05/87	Report to the U.S. Regulatory Commission on Analysis and Evaluation of Operational Data-1986	NUREG-1272 (S701)	
05/87	Loss of Decay Heat Removal Function at Pressurized Water Reactors With Partially Drained Reactor Coolant Systems	S702	H. Ornstein
	A nonreactor report (see NUREG-1272, Vol. 2, No. 2)	S703	
	Engineering Evaluations		
01/87	Potential Containment Airlock Window Failure Due to Radiation	E701	S. Israel
03/87	MOV Failure Due to Hydraulic Lockup From Excessive Grease in Spring Pack	E702	E. Brown
03/87	Loss of Offsite Power Due to Unneeded Actuation of Startup Transformer Protection Differential Relay	E703	F. Ashe
03/87	Discharge of Primary Coolant Outside of Containment at PWRs While on RHR Cooling	E704	S. Israel
03/87	RWCU System Automatic Isolation and Safety Considerations	E705	N. Thomasson
03/87	Inadequate Mechanical Blocking of Valves	E706	T. Cintula
03/87	Design and Construction Problems at Operating Nuclear Plants	E707	C. Hsu
08/87	Depressurization of Reactor Coolant Systems at PWRs	E708	S. Israel
08/87	Auxiliary Feedwater Pump Trips Due to Low Suction Pressure	E709	C. Hsu

Table D-9 Reports Issued in CY 1987 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
10/87	Inadequate NPSH in Low-Pressure Safety Systems in PWRs	E710	S. Israel
	Program Support Reports		
07/87	Operational Experiences at Newly Licensed Nuclear Power Plants	NUREG-1275, Vol. 1	R. Dennig
09/87	Trends and Patterns Program Report-Operational Experience Feedback on Main Feedwater Flow Control and Main Feedwater Flow Bypass Valves and Valve Operators	P701	G. Plumlee
	Technical Reviews		
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
	A nonreactor report (see NUREG-1272, Vol. 2, No. 2)	T711	

Table D-9 Reports Issued in CY 1987 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
11/87	Unplanned Criticality Events at U.S. Power Reactors Similar to That at Oskarshamm Unit 3 on 07/30/87	T712	T. Wolf
12/87	Mispositioning of "Reverse Acting" Valve Controllers	T713	J. Stewart
	A nonreactor report (see NUREG-1272, Vol. 3, No. 2)	T714	

Table D-10 Reports Issued in CY 1986

Date	Title	No.	Author
	Case Studies		
	A nonreactor report (see NUREG-1272, May 1987)	C601	
08/86	Operational Experience Involving Turbine Overspeed Trips	C602	C. Hsu
12/86	A Review of Motor-Operated Valve Performance	C603	E. Brown
12/86	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	C604	M. Chiramal
12/86	Operational Experience Involving Losses of Electrical Inverters	C605	F. Ashe
	Special Studies		
04/86	AEOD Annual Report for 1985	S601	J. Heltemes
05/86	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	S602	J. Crooks
06/86	Adequacy of the Scope of IE Bulletin 86-01	S603	E. Leeds
	Engineering Evaluations		
05/86	Core Damage Precursor Event at Trojan	E514 Revision 1	D. Zukor
01/86	Deficient Operator Actions Following Dual Function Valve Failures	E601	E. Leeds
01/86	Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures	E602	E. Leeds
02/86	Delayed Access to Safety Related Areas During Plant Operation	E603	T. Cintula
03/86	Spurious System Isolations Due to the Panalarm Model 86 Thermocouple Monitor	E604	E. Leeds
04/86	Lightning Events at Nuclear Power Plants	E605	M. Chiramal
05/86	Loss of Safety Injection Capability at Indian Point Unit 2	E606	R. Tripathi

Table D-10 Reports Issued in CY 1986 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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
	Program Support Reports		
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
	Technical Reviews		
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

Table D-10 Reports Issued in CY 1986 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
05/86	Events Resulting From Deficiencies in Labeling and Identification Systems	T604	E. Trager
06/86	Failure of Main Steam Safety Valves to Properly Reseat	T605	R. Freeman
08/86	Inadvertent Recirculation Actuation Signals at Combustion Engineering Plants	T606	T. Cintula
09/86	Occurrence of Events Involving Wrong Units/Wrong Train/Wrong Component-Update Through June 1986	T607	E. Trager
11/86	Hydrogen Fire and Failure of Detection System	T608	M. Chiramal
12/86	Foreign Material and Debris in Safety-Related Fluid Systems	T609	E. Leeds
12/86	ADS/RCIC System Interaction Events at River Bend Unit 1	T610	E. Leeds
12/86	Denied Access Due to Negative Room Pressure	T611	T. Cintula
12/86	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	T612	R. Tripathi
	Incident Investigation Program Reports		
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-3).

Table D-11 Reports Issued in CY 1985

Date	Title	No.	Author
	Case Studies		
09/85	Licensee Event Report System, Evaluation of First Year Results and Recommendations for Improvements	NUREG-1022, Supplement 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
	A nonreactor report (see AEOD Annual Report for 1985[S601])	C505	
	Special Studies		
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		
01/85	Motor-Operated Valve Failures Due to Hammering Problem	E501	M. Chiramal
01/85	Failure of Residual Heat Removal Suppression Pool Cooling Valve to Operate	E502	C. Hsu
03/85	Partial Failures of Control Rod Systems to Scram	E503	M. Chiramal
03/85	Loss or Actuation of Various Safety-Related Equipment Due to Removal of Fuses or Opening of Circuit Breakers	E504	F. Ashe
03/85	Service Water System Air Release Valve Failures	E505	S. Salah
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

Table D-11 Reports Issued in CY 1985 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
5/85	Nuclear Plant Operating Experience Involving Safety System 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
	Program Support Reports		
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
	Technical Reviews		
01/85	Failure of Automatic Protection for Boron Dilution Event at Callaway Unit 1	T501	R. Freeman
03/85	Comparative Analysis of Recent Feedline Water Hammer Events at Maine Yankee, Calvert Cliffs, Palisades, and Salem	T502	E. Leeds
05/85	Pressurizer Level Instrumentation of Combustion Engineering Reactor Units	T503	M. Chiramal
05/85	Loss of Instrument Air and Subsequent Pressure Transient at Callaway Unit 1	T504	R. Freeman

Table D-11 Reports Issued in CY 1985 (cont.)

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

Table D-12 Reports Issued in CY 1984

Date	Title	No.	Author
	Case Studies		
02/84	Licensee Event Report System, Description of System and Guidelines for Reporting	NUREG-1022 Supplement 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
	A nonreactor report (see AEOD Semiannual Report, Sept. 1984)	C405	
	Special Studies		
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 1984	S404	D. Zukor
09/84	Semiannual Report on AEOD Activities	S405	J. Heltemes
10/84	Application of Risk Perspectives: A Procedures Guide	S406	P. Lam
	Engineering Evaluations		
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

Table D-12 Reports Issued in CY 1984 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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 Salem Unit 2 on January 14, 1982	E323 Revision 1	N. Trehan
04/84	Reversed Differential Pressure Instrument Sensing Lines	E408	S. Rubin
05/84	Operating Experience Involving Air in Instrument Sensing Lines	E409	S. Salah
05/84	Operational Experiences Involving Standby Gas Treatment Systems That Illustrate Potential Common-Cause Failure or Degradation Mechanisms	E410	F. Ashe
05/84	Failure of Anti-Cavitation Device in Residual Heat Removal Service Water Heat Exchanger Outlet Valve	E411	C. Hsu
05/84	Adverse System Interaction With Domestic Water Systems	E412	T. Cintula
05/84	Natural Circulation in Pressurized Water Reactors	E413	W. Lanning
05/84	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	E414	P. Lam
06/84	Overcooling Transient	E415	E. Imbro
06/84	Erosion in Nuclear Power Plants	E416	E. Brown
07/84	Loosening of Flange Bolts on Residual Heat Removal Heat Exchanger Leading to Primary to Secondary Side Leakage	E417	C. Hsu
07/84	Feedwater Transients During Startup at Westinghouse Plants	E418	D. Zukor
07/84	Failures of Fischer-Porter Transmitters Used in Safety Related Systems	E419	M. Chiramal
08/84	Operational Experiences Involving Shorted Lamp Sockets of Indication Lights	E420	M. Chiramal
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

Table D-12 Reports Issued in CY 1984 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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
	Program Support Reports		
02/84	Operating History Overview for Diesel Generators in Nuclear Service	P401	R. Dennig M. Chiramal
03/84	AEOD Trends and Patterns Program Plan	P402	R. Dennig
05/84	AEOD Trends and Patterns Evaluation Report, Preliminary Assessment of LER Reporting Under 10 CFR 50.73	P403	F. Hebdon
03/84	LER Data on Personnel Errors	P404	F. Hebdon
11/84	Draft Trends and Patterns Analysis of Feedwater Transients at Westinghouse PWRs	P405	M. Harper
11/84	Trends and Patterns Analysis of Reactor Scrams (Pilot Study)	P406	L. Bell
	Technical Reviews		
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

Table D-12 Reports Issued in CY 1984 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
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 Cooldown Rate Event at LaSalle Unit 1	T417	S. Salah
08/84	Events Involving Fires or Other Related Abnormalities in Motor Control Centers with Aluminum Bus Bars	T418	F. Ashe
08/84	Contamination of Snubber Bleed Screw and Lockup Poppet Valve	T419	C. Hsu
08/84	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	T420	P. Lam

Table D-12 Reports Issued in CY 1984 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
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-13 Reports Issued in CY 1983

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

Table D-13 Reports Issued in CY 1983 (cont.)

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

Table D-13 Reports Issued in CY 1983 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
08/83	Human Factors Contributions to Accident Sequence Precursor Events	N305	E. Trager
	Program Support Reports		
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
	Technical Reviews		
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
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

Table D-13 Reports Issued in CY 1983 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
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
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

Table D-13 Reports Issued in CY 1983 (cont.)

Date	Title	No.	Author
	Technical Reviews (cont.)		
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-14 Reports Issued in CY 1982

Date	Title	No.	Author
	Case Studies		
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		
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 Accidentsin LWRs	E205	M. El-Zeftawy
02/82	Load Reduction Transient on January 14, 1982, at Salem Unit 2	E206	N. Trehan
02/82	LER 50-336/81-26: Investigation of the Relative Frequency of Valve Overtravel Anomalies That Could Result in a Potential Centrifugal Pump Runout Exceeding Net Positive Suction Head	E207	E. Imbro
02/82	An Observed Difference in Lift Setpoint for Steam Generator and Pressurizer Safety Valves	E208	T. Cintula
02/82	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback Unit 1 on April 13, 1979)	E209	M. Chiramal
02/82	Inadequate Switchgear Cooling at Beaver Valley Unit 1	E210	W. Lanning

Table D-14 Reports Issued in CY 1982 (cont.)

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

Table D-14 Reports Issued in CY 1982 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
06/82	Repetitive Overspeed Trips of the Steam Driven Emergency Feedwater Pump on Initial Start at Arkansas Nuclear One, Unit 2	E228	E. Imbro
06/82	Potential for Flooding in Control Room at San Onofre Units 2 and 3	E229	T. Cintula
07/82	Water in the Fuel Oil Tank at Surry Power Station, Unit 2-Additional Information	E230	N. Trehan
07/82	Millstone Unit 2 Loss of Shutdown Cooling Due to Trip of Low Pressure Safety Injection Pump	E231	M. Chiramal
07/82	Potential Deficiency in the Sigma Lumigraph Indicators Model Number 9270	E232	F. Ashe
07/82	Carbon Dioxide Systems Used for Fire Protection in or Adjacent to Critical Areas	E233	M. Chiramal
08/82	Failure in a Section of 4 kV Bus Cable Manufactured by Okonite	E234	F. Ashe
08/82	Wiring Error in Handswitch for Solenoid Control Valves Associated With High-Pressure Coolant Injection System Steam Condensing Mode Pressure Control Valve at Duane Arnold	E235	S. Rubin
08/82	Brunswick Steam Electric Plant Unit 2 Loss of Residual Heat Removal Service Water on January 16, 1982	E236	T. Wolf
08/82	Power-Operated Relief Valve Failure at Robinson	E237	E. Brown
08/82	Water in the Lube Oil in Safety Injection Pump IA-A at Sequoyah-LER 81-076	E238	N. Trehan
09/82	Main Steam Isolation Valve Closures and Pressurizer Safety Valve Actuations at St. Lucie Unit 1 on December 19, 1981	E239	T. Cintula
09/82	Preliminary Account of Events Associated With a Reactor Trip at Hatch Unit 2 on August 25, 1982	E240	S. Rubin
10/82	Emergency Diesel Generator System Problems at Fitzpatrick	E241	M. Chiramal
10/82	Fuel Assembly Degradation While in the Spent Fuel Storage Pool	E242	E. Brown

Table D-14 Reports Issued in CY 1982 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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
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-15 Reports Issued in CY 1981

Date	Title	No.	Author
	Case Studies		
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	NUREG-0785 (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		
01/81	Degradation of Internal Appurtenances in LWR Piping	E101	E. Brown
01/81	Sequoyah Unit 1 Loss of Annunciation	E102	M. Chiramal
02/81	Davis-Besse Nuclear Power Station, Unit 1-Engineered Safety Features Actuation System (ESFAS)	E103	M. Chiramal
03/81	Engineering Evaluation of Feedwater Transient and System Pipe Break at Turkey Point 3	E104	S. Sands
03/81	Water Hammer During Restart of Residual Heat Removal Pumps	E105	J. Huang
03/81	Water Hammer in the Steam Condensing Mode of the Residual Heat Removal System Operation	E106	J. Huang
04/81	Peach Bottom Unit 3 Occurrence on February 25, 1981	E107	F. Ashe
04/81	Hatch Units 1 and 2-Alternate Offsite Source Interlock With Emergency Diesel Generators	E108	M. Chiramal
04/81	Potential Common-Mode Failure of Diesel Generators	E109	M. Chiramal
04/81	Requirements of the Preferred or Offsite Power System	E110	F. Ashe
05/81	Evaluation of High Pressure Safety Injection Pump Operability Without Service Water	E111	E. Imbro
06/81	Inoperability of Instrumentation Due to Extreme Cold Weather	E112	M. Chiramal

Table D-15 Reports Issued in CY 1981 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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
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

Table D-15 Reports Issued in CY 1981 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
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-16 Reports Issued in CY 1980

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

Table D-16 Reports Issued in CY 1980 (cont.)

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

APPENDIX E

AEOD Technical Reports by Category

E-1 PWR Plant Systems

E-2 BWR Plant Systems

E-3 Activity/Human Factor Deficiency

E-4 Topics

INTRODUCTION

Background

Appendices C and D list approximately 500 operating experience reports published by AEOD since 1980. These reports cover a broad spectrum of operating experience data. Some of them have also been published as NUREGs, including the NUREG-1275 series of Operating Experience Feedback Reports. AEOD reports have been broadly disseminated throughout the nuclear community and to the public. Most reports can be found in the NRC's Public Document Room, the local public document rooms, and the Nuclear Documents (NUDOCS) database under the *Task Identifier* AE, followed by the report number.

This appendix has been prepared as an aid to more effectively communicate the lessons of operating experience, and to help ensure that those lessons are not forgotten. It contains tables of AEOD report numbers sorted by topic. A report may be listed in more than one topical area, depending upon its scope. To find the title for any report, refer to Appendix C or D, as appropriate.

Tables E-1 and E-2 use system descriptions which contain material copyrighted by *Nuclear Power Experience*. This material is reproduced by permission of Hagler Bailly Consulting, Inc.

Definitions

AEOD reports are designated by an alphanumeric sequence. The first character is a letter prefix which denotes the type of report. The remaining characters comprise the report number, which indicate the year of publication and the sequence number of the report. These designators are described below.

Case Studies. Case studies are designated by a C prefix and involve substantive, in-depth analyses of significant safety issues that are identified through the review of operating experience. Case studies document the bases for AEOD recommendations for regulatory or industry actions. Before being published, each case study report goes through a rigorous peer review process to ensure technical adequacy.

Special Studies. Special studies are designated by an S prefix and document accelerated investigations and suggest or recommend regulatory actions that are to be completed expeditiously.

Engineering Evaluations. Engineering evaluations are designated by an E prefix. They document assessments of significant operating events and suggest remedial actions, if appropriate.

Technical Reviews. AEOD technical reviews are designated by a T prefix and document studies in which the staff concludes there is little safety significance. These studies typically conclude that the licensees' or industry's planned or scheduled corrective actions are adequate.

Program Support Reports. Program support reports are designated by a P prefix and document studies of trends and patterns in a variety of systems, components, events, and programs.

Report Number. For reports issued in the 1980s, the report number is a three digit number. The first digit is the last digit of the year of publication (i.e., a "0" indicates 1980 and a "1" indicates 1981). The remaining two digits are the sequence number, representing the sequential order of publication in that year. For reports issued in the 1990s, the report number consists of four digits with a hyphen in the middle. The first two digits (before the hyphen) are the last two digits of the year of publication (i.e., "90" indicates 1990 and "95" indicates 1995). The last two digits are the sequential number of publication in that year.

APPENDIX E-1

PWR Plant Systems

CONTENTS

PWR Plant Systems[1]

I. Fuel - E205, E242, E301, E313, E326
Includes uranium fuel pellets and cladding, fuel assemblies, holdout springs, guide tubes

II. Reactor Internals - T809
Includes upper guide structure, thermal shield, core barrel, supports for core instrumentation

III. Reactor Vessel - E114
Includes reactor pressure vessel (RPV), nozzles, head bolts, seals

IV. Control Rods and Drives
A. *Control Rods* - T324, E613, T712, E90-04 Includes absorber and poison rods, rod control cluster assemblies, control element assemblies

B. *Drives* - E206, T311, E323 Rev. 1, S503, T511, T910, T91-07 Includes magnetic jack control rod drive mechanisms, housings, drive shafts, motors, clutches, latches, grippers

V. Reactor Coolant System (RCS)
A. *Pumps* - E003, E127, E326, E415, T707, NUREG-1275 Vol. 7 Includes main reactor coolant pumps (RCPs), casings, flanges, shafts, bearings, seals, impellers, speed controls

B. *Piping* - E101, T322, T701 Includes main coolant lines, welds, fittings

C. *Relief and Safety Valves* - T314, T321, C401, E426, T91-05, S92-02, T92-01, T93-01, T96-02 Includes safety/relief valves (SRVs), including pressurizer SRV and power-operated relief valves (PORV)

D. *Steam Generators (SG)* - C003, C005, C101, E101, E224, T330, E413, E423, T512, E708, E906, E909, E96-03 Includes SG shell, internal tubing, support plates, nozzles, manways, blowdown lines

E. *Pressurizer* - C102, E208, E237, E239, E320, E421, T506, E708, T704, E805, E909, S902, E90-08, E90-09 Includes pressurizer shell, internal heaters, manway, nozzles, pressurizer relief tank, PORV block valves, resistance temperature detectors (RTDs), manifold valves

F. Miscellaneous - **None** Includes additional RCS loop valves not associated with above categories

VI. Turbine Cycle Systems
A. *Turbine* - None Includes main turbine, high pressure (HP) and low pressure (LP) cylinders, including casings, rotors, shafts, blades, bearings, stop and control valves, drain and crossover lines, lube oil system

B. *Generator* - E209 Main generator system includes rotor, stator, exciter, brushes, bearings, coils, voltage regulator, armature, commutator, windings, generator cooling, seal oil systems

C. *Condensers* – None Main condenser includes tubes, baffles, vacuum pump, air ejector, hotwell, shells, water boxes

D. *Steam* – C005, E017, E122, E128, E208, E239, E415, E502, E514, T90-04, T90-08, S92-02 Includes turbine bypass and atmospheric steam dump valves, SRV, main steam isolation valves (MSIV), moisture separator reheaters (MSRs), main steam line (MSL) piping

E. *Condensate and Feedwater* – C003, C005, E013, E017, E020, E104, E115, E117, E206, E211, E213, E228, E248, E255, E323, E325, T302, T319, T327, T337, C404, P405, E323

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Rev. 1, E415, E418, T402, T416, P501, E502, E514, T502, NUREG-1154, C602, T603, E709, P701, T703, E906, T927, E90-03, T91-04, S93-01, NUREG-1275 Vol. 10 Includes condensate, booster, feedwater (FW) and auxiliary feedwater (AFW) pumps, condensate storage tank (CST), demineralizer system, LP and HP heaters, associated valves and piping

F. ***Circulating Water*** **– E016, C204, E243, E311, T318, T333, T804, T90-06, T90-16** Includes intake structures, screens, cooling towers, discharge gates and canals, associated pumps and valves, saltwater system

G. ***Electrohydraulic Control (EHC) System*** **– C102, E247, E249** Includes EHC fluids, auto-stop oil, interface valves, valve operators for HP and LP turbine, pumps and associated controls

H. ***Miscellaneous*** **– E416** Includes heater drain system, extraction steam

VII. Safety Systems

A. ***Emergency Core Cooling System*** **(ECCS) – E111, E207, E238, E317, T310, T319, T328, T335, E404, T403, T422, S603, E606, T708, E803, T90-14, NUREG-1275 Vol. 9, T93-01, T95-02** Includes safety injection (SI), upper-head injection systems, accumulators, boron injection tank

B. ***Containment Pressure Suppression*** **– E316, T317, T338, T513, E710, T811, T926, T90-17** Includes containment spray, ice condensers, recirculation spray, chemical addition tanks

C. ***Containment Atmosphere Cooling*** **– E012, E203, E221** Includes containment fans, containment air recirculation

D. ***Containment Isolation*** **– E009, E011, E221** Includes containment isolation valves (CIV), containment

E. ***Miscellaneous*** **– E204, E229, E230, E233, T331, T339, T424, T608** Includes fire systems, containment hydrogen venting, purge and recombiners, security systems, respirators

VIII. Auxiliary Systems

A. ***Coolant Volume, Purification, Chemical Sampling*** **– C102, E308, E314, T415, E512, T501, T504, E607, T702, E910, T91-02, NUREG-1275 Vol. 9, T93-01, T95-02** Includes chemical and volume control system (CVCS), charging pumps, post-accident sampling system, boric acid storage tank, boron recycle system, letdown lines and valves

B. ***Auxiliary Cooling*** **– E001, E005, E006, E012, C105, E111, E115, E132, C202, E223, E231, E256, E257, E302, E303, E304, E310, E311, E315, E321, E202, T303, T305, T341, E411, T403, T415, C503, E502, E506, T505, T514, T602, E704, E710, S702, NUREG-1275 Vol. 3, E806, E807, T804, E907, T919, T926, E90-02, E90-06, T90-11, T90-13, T91-01, T91-03, E91-01, S93-03** Includes residual heat removal (RHR), component cooling water (CCW), service water (SW) and essential raw cooling water systems, pumps, heat exchangers (HX), shutdown cooling, associated valves and piping

C. ***Miscellaneous*** **– None** Includes RCS drains, containment sump valves

IX. Instrumentation and Control (I&C)

A. ***Nuclear Instrumentation*** **- T92-05** I&C for incore neutron flux monitoring, including source and intermediate range monitors (SRMs and IRMs), power range monitors, related amplifiers and indicators

B. ***Reactor Protection System (RPS)*** **– E103, E014, E133, E245, P301, T316, T320, P406, E323 Rev. 1, E419, E421, P504, T503, T90-07, 94-03** I&C for manual or auto reactor trip channel actuation, including RTD, reactor trip breakers, pressurizer pressure and level transmitters, SG level transmitters and FW flow transmitters. Includes anticipated transient without scram (ATWS) backfits

C. ***Reactor Control*** **– C004, E323, E507, NUREG-1195** Includes the integrated control system (ICS), axial flux monitors, control rod positioning, and other rod and core performance monitoring and control I&C

D. ***Turbine Cycle*** **– E017, E228, T706, NUREG-1275 Vol. 11** I&C for manual or automatic turbine trip channel actuation and turbine generator and FW control, including EHC, vibration and wear probes, governors, FW and AFW flow

E. ***Safety Systems*** **- E019, E102, E103, E112, E114, E121, E226, E227, E321, T308, T310, T313, T320, T335, C402, E404, E409, E419, T405, P503, E508, E515, C604, E605, T606, T612, T904, NUREG-1275 Vol. 8, T93-03** Includes I&C for actuation of ECCS, engineered safety features (ESFs), solid state protection system, fire systems, containment pressure suppression and isolation, and main steam isolation, refueling water storage tank level, toxic gas isolation system (TGIS), TGIS butane monitor, containment sump level, borated water storage tank level, steam line differential pressure, RPV level, SG level and flow, auxiliary alarm annunciator

F. ***Process Systems*** **– C004, E314, T331** I&C for process computer, RCP pressure seal sensing, CVCS tank level, heat tracing controls, accumulator level, containment fan coil unit SW flow, and acidity or alkalinity instruments

G. ***Reactor Coolant Control*** **– T902** I&C for RCS flow, subcooling monitors, pressurizer level (B&W)

H. ***Miscellaneous*** **– E123, C204, E302, T333, T401, T504, T506, T804, T92-06** I&C for containment sampling and monitoring, general area radiation monitoring, I&C air, incore thermocouples (T/Cs), gaseous nitrogen system valves and loose parts monitor

X. Fuel Handling Facilities and Systems - E242, E313, S96-02

Includes reactor cavity, refueling canal, fuel transfer system, spent fuel pool, and racks, new fuel storage, cranes and lifting devices, tools and fixtures, and associated I&C

XI. Electrical Systems

A. ***Emergency Power*** **– E008, E126, E220, E302, E251, E253, E254, E302, E307, E318, E324, P401, E424, E427, E510, E514, E612, T914, T925, S91-01, T92-08, T92-10, E93-03, S96-03** Includes batteries, diesel generators (DG), battery chargers, motor generator (MG) sets, and associated I&C

B. ***Other Electrical*** **– C003, E004, E008, E010, C104, E102, E110, E116, E131, E102, E110, E116, E131, E210, E212, E222, E234, E246, E251, NUREG/CR-3122, C301, E320, T301, T311, T315, E401, E412, T404, T418, T424, E504, C605, NUREG-1190, E703, E905, T919, E90-01, E90-05, T90-03, T91-07, S91-01, S92-03, S93-06, E93-02, T96-03** Electrical distribution systems include buses, breakers, inverters, transformers, motor control centers (MCC), switchgear, on- and offsite distribution lines, and associated I&C

XII. Liquid Radwaste System - None Includes liquid and solid radwaste tanks, evaporators, filters, valves, chemical drains, piping, associated I&C

XIII. Gaseous Radwaste System - E120, E327, T411 Includes waste gas processing, auxiliary building gas treatment, waste gas decay tank, compressor, gaseous hydrogen recombiner, filters, stack monitors, associated I&C

XIV. Buildings and Containment

A. ***Penetrations*** **– E701, E808, T804, T916** Includes airlocks, hatches, manways, electrical and piping penetrations, fire doors, seals, gaskets to containment and among plant buildings

B. ***Rooms*** **– E229, E306, E611, T611, E802, T909, E90-07** Control room, remote shutdown panel, control room ventilation, auxiliary building, turbine building

C. ***Miscellaneous*** **– E124, E304, T413, T710, T909** Includes heating, ventilation and air conditioning (HVAC), fire dampers, charcoal absorbers, containment purge butterfly valves and purge isolation valves

XV. Miscellaneous Systems - E219, E406, E412, E423, C501, S503, E501, C603, E702, NUREG-1275 Vol. 2, NUREG-1275 Vol. 6, E902, T914, T928, E90-08, T90-04, E92-01, T92-03, T92-04, E96-01 Includes plant air systems, snubbers, pipe and building supports, nonradioactive waste neutralizing systems, general valve operator problems, rupture discs and rescue breathing apparatus, auxiliary systems

XVI. Operational Problems
A. ***Inservice inspection*** **– T327, E612, T805, E906, E910, T95-02** Includes operational problems arising from scheduled inservice inspections (ISIs)

B. ***Refueling*** **– E205, E806, S96-02** Includes operational errors occurring during initial fuel load, refueling or spent fuel handling

C. ***Miscellaneous*** **– E411, E704, NUREG-1275 Vol. 8, S96-02, T95-03** Includes operator errors and procedural problems relating to the full range of plant systems, especially those involving radiation exposure or contamination

APPENDIX E-2

BWR Plant Systems

CONTENTS

BWR Plant Systems[2]

I. Fuel - E205
Includes uranium fuel pellets and cladding, fuel assemblies spacers, tie plates and channels

II. Reactor Internals - E002
Includes jet pumps, FW and core spray spargers, steam dryer assembly, core support and guide, core shroud

III. Reactor Vessel - E114
Includes RPV, lines and nozzles - FW and core spray, control rod drive (CRD) return, recirculation

IV. Control Rods and Drives
A. ***Control Rods* – T340, T510, T712, S803**
Includes rods, sheaths, blades

B. ***Drives* – C001, C002, E007, E015, C103, E225, E240, T306, C403, E403** Includes CRDs, hydraulic control units, hydraulic supply system, scram discharge header

V. Recirculation, Steam and Relief
A. ***Pumps* – E107** Includes recirculation pumps, drives, and seals, speed controls, recirculation manifold

B. ***Piping* – None** Includes main steam lines, suction and discharge risers, flow restrictors, bypass lines

C. ***Relief and Safety Valves* – E240, E312, E322, E502, T610, S92-02, T96-02** Includes SRV, Includes SRV, MSIVs, automatic depressurization system (ADS) valves

D. ***Miscellaneous* – None** Includes recirculation loop valves (drain valves, sample isolation valves, flow control valves)

VI. Turbine Cycle Systems
A. ***Turbine* – None** Includes rotor, shaft, bearings, blades, casing, valves (admission, stop, control, intercept), cross-over piping, lube oil system

B. ***Generator* – E209** Includes rotor, stator, exciter, bearings, voltage regulator, core monitor, generator cooling systems

C. ***Condensers* – None** Includes tubes, baffles, spargers, shell, water box, hotwell vacuum systems (air ejector, vacuum pump), expansion joint

D. ***Steam* – T323, T417, T605, E801, T801**
Includes turbine bypass system, reheaters, moisture separators

E. ***Condensate and Feedwater* – T90-09**
Includes pumps, LP and HP heaters, condensate demineralizer system, CST

F. ***Circulating Water* – E113, E214, E215, T323, T90-06** Includes intake structure, discharge canal, circulating water pumps, dilution pumps, cooling towers, cooling water pumps

G. ***Miscellaneous* – E416** Includes extraction steam pipes and valves, heater drain system

VII. Safety Systems
A. ***Reactor Core Isolation Cooling (RCIC)* – T420, T610, E904, NUREG-1275 Vol. 9, T93-01, NUREG-1275 Vol. 10, T95-02**
Includes RCIC pump, drive and speed controls, and associated piping and valves

B. ***Standby Liquid Control (SBLC)* – T507, T91-02, T92-07** Includes SBLC pumps, tank, explosive valves, piping

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C. ***Core Spray (CS)* – E511, E513, NUREG-1275 Vol. 9, T95-02** Includes HP and LP core spray pumps, valves, piping

D. ***Residual Heat Removal (RHR)* – E105, E106, E119, E125, E218, E236, E244, E309, T332, T334, E411, E414, E417, E502, T515, S603, E601, E608, E609, E908, NUREG-1275 Vol. 9, T95-02** Includes low pressure coolant injection, containment coolers and shutdown cooling systems (including HXs), associated valves and piping

E. ***High Pressure Coolant Injection (HPCI)* – E235, E402, E407, E422, E425, T410, T508, T516, E904, T906, T922, NUREG-1275 Vol. 9, NUREG-1275 Vol. 10, S95-02, T95-02** Includes HPCI turbine, pumps, drives and speed controls, associated valves and piping

F. ***Miscellaneous* – E204, E229, E233, E240, E319, T304, T331, T336, C502, E511, E601, T601, T608, T923, T90-16, NUREG-1275 Vol. 9, S96-01** Includes isolation condenser systems, CIVs, fire protection systems, containments, drywell

VIII. Auxiliary Systems

A. ***Reactor Water Cleanup (RWCU)* – T307, T329, E705** Includes regenerative and nonregenerative HXs, filter-demineralizer units, RWCU pumps

B. ***Reactor Building Closed Cooling Water* – None** Includes pumps, surge tank, coolers, HXs

C. ***Miscellaneous* – C202, E505, NUREG-1275 Vol. 3, E807, T90-13, T91-01, S93-03** Includes SW systems, steam line drains, sump drains

IX. Instrumentation and Control (I&C)

A. ***Nuclear Instrumentation* – S803** Incore neutron flux detection I&C, including traversing incore probes, SRMs, local power range monitors, IRMs, average power range monitors

B. ***Reactor Protection System (RPS)* – E110, T306, P406, E412, T403, P504, T905, T90-07, T94-03** Trip channel systems for manual or automatic control rod scramming, safe reactor shutdown, including ATWS backfits

C. ***Reactor Control* – E018** Includes rod sequence control system, manual rod control system, rod block monitor system, rod position indication system, I&C for core performance, power, mode changes

D. ***Turbine Cycle* – T417, NUREG-1275 Vol. 11** EHC system, including electric pressure regulators, mechanical pressure regulators, FW flow controllers, condenser hotwell and heater level controls

E. ***Safety Systems* – E109, E114, E118, C201, E226, E227, T325, T336, C402, E405, E407, E408, E409, E425, T403, P503, E508, C604, E604, E605, E610, T612, NUREG-1275 Vol. 8, T93-03** I&C for ECCS, ESF, and other safety system actuations, including rod worth minimizer, isolation condenser, RCIC, SBLC, CS, RHR, HPCI, standby gas treatment (SBGT), ADS, torus, main steam line, and fire protection systems

F. ***Process Systems* – T309, T331** I&C for process computer, RWCU, flow, level, and pressure detectors, transmitters, and recorders

G. ***Miscellaneous* – E232, T92-06** Includes I&C for containment sampling and monitoring, leak detection, data acquisition, seismic and sonic detection and instrument air systems

X. Fuel Handling Facilities and Systems

S96-02 Includes refueling bridge platform, grapple, spent fuel pool and racks, and associated I&C, ventilation

XI. Electrical Systems

A. ***Emergency power* – E108, E109, E126, E129, E241, E307, E324, T336, P401, E401, E427, T408, E510, E612, T914, S91-01, T92-08, T92-10, E93-03, S96-03** Includes DG, gas turbine generators, alternating current (ac) uninterruptible power supply UPS, direct current (dc) backup, MG sets, safety buses, batteries, and battery charger current (dc) backup, MG sets, safety buses, batteries, and battery charger

B. ***Other electrical*** – **E107, E108, E246, E250, NUREG/CR-3122, C301, T312, T336, E420, T407, T414, E504, T509, E605, E804, T903, T915, T921, E90-01, E90-05, E90-10, T90-12, S91-01, S92-03, S93-06, E93-02, S96-03**
Includes main unit transformer, auxiliary transformer, safeguards inverters, MCCs, buses, breakers, relays, fuses, switchgear, on- and offsite distribution lines

XII. Liquid Radwaste System
None Includes concentrator, demineralizer, filters, collector tanks, drain tanks, sample tanks, surge tank, CST, spent resin tank, solid radwaste separators, centrifuges, and hopper, and associated I&C

XIII. Gaseous Radwaste System
None Includes stack gas and offgas charcoal absorbers, cryogenic distillate systems, sample pumps, recombiners, high-efficiency particulate air filters, monitors, analyzers, and other I&C

XIV. Buildings and Containment
A. ***Penetrations*** – **C103, T412, E808** includes airlock, manway, hatch, electrical and tubing penetrations, seals, and gaskets to containment and among plant buildings

B. ***Rooms*** – **E229, T406, E603, E611, T909, E90-07** Control rooms, remote shutdown panel, control room ventilation, auxiliary building, turbine building

C. ***Miscellaneous*** – **E322, T307, E410, T421, T710, T713, E802, T903, T909**
Includes HVAC systems, suppression chamber (torus) pressure suppression systems, containment atmosphere dilution systems, SBGT systems, vacuum breakers, gaseous nitrogen systems, cranes

XV. Miscellaneous Systems
E007, E219, E406, E412, E414, T419, C501, S503, E501, C603, E702, NUREG-1275 Vol. 2, NUREG-1275 Vol. 6, T914, E92-01, T92-03, T92-04, E96-01 Includes plant air systems, auxiliary boilers, seismic and component restraints (hangers, snubbers, etc.), general valve operator problems

XVI. Operational Problems
A. ***Inservice Inspection*** – **T805, T95-02**
Includes operational problems arising from scheduled ISIs

B. ***Refueling*** – **E205, E612, S96-02** Includes chiefly errors arising from mishandling of equipment during periods of removal of RPV head for initial fuel loading, refueling and spent fuel handling

C. ***Miscellaneous*** – **C92-01, S96-02, T95-03** Includes operator and personnel errors, procedural problems relating to the full range of plant systems, particularly those concerning exposure to radiation or radioactive contamination

APPENDIX E-3

Activity/Human Factor Deficiency

CONTENTS

Activity/human Factor Deficiency

I. Administrative/Procedures
C002, E004, E008, E010, E223, E306, E323, T306, T313, T320, T321, T328, E425, E426, T416, C503, T502, T509, T510, T512, C602, C603, E602, E608, T612, E706, T710, T713, S801, S803, E801, T806, E90-02, E90-07, T90-03, T90-12, NUREG-1275 Vol. 8, E92-01, T92-07, S93-05, E93-01, T93-01, S95-01, E95-01, S96-02

II. Construction
E707, T91-06

III. Design
E012, E013, E017, E018, E213, E225, E235, E308, T301, T302, T303, T308, T319, T325, T329, T336, E407, E408, E410, T408, T421, E502, E511, C602, C604, C605, E604, E607, E611, E707, E708, E709, E710, T703, T708, T710, S803, E802, E803, T805, T904, T909, T914, E90-07, T91-01, NUREG-1275 Vol. 9, T94-02, S95-01, E96-01, E96-03, S96-02

IV. Fabrication, Part 21, Quality Assurance
S401, E403, T410, T805, T914, E96-03

V. Installation
E408, E424, E611, T701, T704, T805, T914, T90-03, E96-01, E96-03

VI. Maintenance
C204, E237, S401, E401, E403, E410, E414, C503, E504, T511, C605, E607, E608, T612, E707, E708, T701, T704, S804A, S804B, E802, T809, NUREG-1275 Vol. 6, S901 Rev. 1, E901, T902, T912, T913, E90-03, E90-07, S91-01, S92-02, NUREG-1275 Vol. 9, E92-01, T92-01, T92-04, T92-09, S93-05, T93-01, T94-02, T94-04, S95-01, E96-01, E96-03, T96-02

VII. Operation
E221, E223, T328, T340, E602, T708, T712, S803, E801, E802, E803, E901, E909, T909, NUREG-1275 Vol. 8, S93-05, E93-01, S95-01, E95-01, T95-03, S96-02

VIII. Radiation Protection
None

IX. Test and Calibration
E129, E318, E320, T304, T305, T313, E410, E414, E420, E421, E425, T410, T424, C503, E512, E515, T510, C605, E90-03, E90-07, E90-08, S92-02, E92-01, T92-01, T92-04, T92-05, T92-07, NUREG 1275 Vol. 11, T95-02, E96-03, T96-02

X. Licensee Program
E96-02, E96-02, Supplement 1

APPENDIX E-4

Topics

CONTENTS

CONTENTS (cont.)

Topics

I. Anticipated Transient Without Scram (ATWS)
C001, C002, C103, E133, P301, E323 Rev. 1, E503, S803, E804

II. Stress Corrosion Cracking and Variations
E242, E313, T402, E506, E613, T906, T910, E96-03

III. Loss-of-Offsite Power (LOOP)
C003, E253, E302, E401, E413, E605, E610, NUREG-1190, E703, T915, T925, E90-01, E90-05, T90-12, T91-03, T91-07, S91-01, T92-08, E93-02, T94-01, T95-01, T96-03

IV. Unplanned Criticality
T712, S803

V. Foreign Reactor
T712, E706, E805, T803

VI. Weather Related
C003, E112, E226, E401, T405, E605, T90-09

VII. Natural Circulation
C003, C101, E413, S96-01

VIII. Transient
C004, E014, E104, E114, C205, E206, E221, E238, E240, E246, E247, E249, E306, E323, E326, C403, P405, E323 Rev. 1, E413, E415, E418, T417, P501, E509, E514, T605, NUREG-1195, E708, E801, T801, E904, E905, E909, E90-09, T90-08, NUREG-1275 Vol. 8, T92-02, T96-03

IX. Loss of Coolant Accident (LOCA)
C004, E112, E223, E253, E302, E322, T301, T318, C403, E417, C502, T506, E704, E705, E710, T707, E805, S902, T91-03, T94-04, S95-01

X. Flooding
E221, E225, E229, T514, E705, E90-07, T91-06, T92-06

XI. Water Hammer
C005, E104, E105, E106, E309, T327, T329, T337, E402, T502, T516, NUREG-1190, E91-01

XII. Flooding of Steam Lines and Steam Generator Overfill
C005, E013, E017, E303, E801

XIII. Single Failure, Common Cause, Common Mode
E004, E109, E116, E125, C204, E219, E230, E302, E304, E311, E325, T302, T304, T313, T336, T339, C404, E403, E405, E408, E410, E426, T410, T418, T421, E503, T505, T507, T509, T513, T515, C604, S603, E702, E709, T703, T708, NUREG-1275 Vol. 3, E802, NUREG-1275 Vol. 6, E907, E910, T909, T919, E90-05, E90-07, T90-11, NUREG-1275 Vol. 9, E92-02, T92-05, T94-04, T95-02, E96-01

XIV. Paralleling
E008, E010, T921

XV. Valves
E009, E012, C102, C103, E104, E120, E122, E124, E128, E132, E208, E211, E225, E235, E237, E239, E248, E304, E305, E311, E312, E315, E320, E325, T303, T314, T318, T319, T321, T326, C404, S402, E403, E414, T410, T420, C502, E502, E505, E506, E511, E513, T507, T513, T514, C603, E601, E702,

XV. Valves (cont.)
E705, E706, T801, NUREG-1275 Vol. 6, E906, E908, E909, T918, T927, T928, E90-02, E90-06, E90-09, T90-04, T91-04, T91-05, S92-02, NUREG-1275 Vol. 9, T92-01, T92-03, T92-04, T92-09, NUREG-1275 Vol. 10, NUREG-1275 Vol. 11, T95-02, E96-01, T96-02

XVI. Clams, Bivalves, and Debris
E016, E111, E119, E123, C202, C204, E202, E215, E219, E220, E318, T305, T307, T402, T419, T422, E512, T513, T609, NUREG-1275 Vol. 3, E905, T916, T923, T90-16, S93-03

XVII. Mode Switch
E018, E90-10

XVIII. Shared Systems
E507, E510

XIX. Recirculation Actuation Signal
E019, T335, T606, E710, E803, T916

XX. Blowdown
C103, E218, E239, E706, E909, E90-02, S95-01

XXI. TMI Precursor Event
E115, E117, E120, E216, E326, P402, E909

XXII. Sabotage
E113, T322, T903

XXIII. Fire
E116, E120, T404, T418, T608, E902, E905, T903, T915, E90-01

XXIV. Corrosion and Erosion
E130, E312, T318, T341, E411, E416, E908, T90-13, S93-03, E96-03

XXV. Steam Generator Tube Rupture
E224, T330, E708, E909, E96-03

XXVI. NET Positive Suction Head (NPSH) and Pump Runout
E213, E214, E218, E256, E257, E302, E314, E323, E325, E326, C404, E323 Rev. 1, E411, T515, E606, T603, E709, E710, T703, S702, E803, E806, E807, E910, T916, T927, E90-06

XXVII. Overpressure
E248, C401, E90-03, E90-09

XXVIII. Piping
E255, E308, T314, T322, T337, T341, E612, E705, E902, S902, T90-16, E92-01

XXIX. Stratification
E256, E415, S902

XXX. Safety Injection Actuation Signal Bypassed or Blocked
E326, T310, E909, NUREG-1275 Vol. 8, E95-01

XXXI. Explosion
E327, E902

XXXII. Harsh Environment
T302, T92-06

XXXIII. Final Safety Analysis Report (FSAR)
T319, T323, E423, E612, E90-08

XXXIV. Fastener
E424, T906

XXXV. Operational Experience – General

NUREG-1275 Vol. 1, NUREG-1275 Vol. 4, NUREG-1275 Vol. 5, NUREG/CR-4674 Vol. 1-22, S96-02

XXXVI. Risk Assessment

NUREG/CR-4674 Vol. 1-22, S96-02

XXXVII. Component Aging

NUREG/CR-6442

XXXVIII. System Reliability

S95-02, S96-01, S96-03

APPENDIX F

Status of AEOD Recommendations

Status of AEOD Recommendations

This appendix summarizes the year-end status of all AEOD recommendations that are either new or still outstanding since the last report. During 1996, all outstanding recommendations were resolved, and no new recommendations were added. Therefore, as of September 30, 1996, no AEOD recommendations were outstanding.

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

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 currently outstanding or has been resolved in the past year, including a description and status for each, follows.

AEOD Recommendations Tracking System

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
NRR/DRPM/PECB	D. L. Skeen	High

Status: Resolved

The case study was issued in December 1990, and issued as NUREG-1275, Vol. 6, "Operating Experience Feedback Report – Solenoid-Operated Valve Problems," in February 1991. Generic Letter 91-15, "Operating Experience Feedback Report, Solenoid-Operated Valve Problems," was issued in September 1991 to alert licensees to the issues presented in the case study. AEOD worked with the Electric Power Research Institute/Nuclear Maintenance Assistance Center (EPRI/NMAC) and developed an SOV maintenance guide. After the guide was issued, EPRI/NMAC held several SOV workshops in which AEOD participated. AEOD has also been working with the Institute of Electrical and Electronic Engineers and the American Society of Mechanical Engineers (ASME) to

AEOD Recommendations Tracking System

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

formulate industry consensus documents on SOVs. AEOD has participated in many meetings of the Air-Operated Valve Users Group (AUG) and has presented updates on operating experience relating to SOVs and AOVs. Similarly, AEOD has made numerous presentations on SOV operating experience at meetings of the American Nuclear Society and ASME, and the International Conference on Nuclear Energy.Many plants have examined their SOVs in light of the generic letter and the case study and have implemented the recommendations in the study to varying degrees. Still there have been recent occasions in which plants have had SOV problems which have been traced back to their failure to implement the generic letter and case study recommendations.

AEOD Recommendations Tracking System

Recommendation Source: Special Study AEOD/S92-02 (NUREG-1275, Vol. 9)

Responsible AEOD Engineer: E.J. Brown

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

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

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

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

Status: Resolved

The special study was issued in December 1992, and subsequently issued as NUREG-1275 Vol. 9, "Pressure Locking and Thermal Binding of Gate Valves," in March 1993. NRR has conducted several workshops for NRC inspectors with AEOD participation. The subject was also presented in a public workshop on Generic Letter (GL) 89-10 at the 1993 Motor-Operated Valve (MOV) Users Group meeting in February 1993, in which AEOD staff participated. These issues were incorporated into Supplement 6 to GL 89-10. NRR inspector guidance was developed and provided as Temporary Instruction 2515/109 for GL 89-10 Part 2 inspections. The Part 2 inspections indicated that most licensees had completed little in identifying and correcting the valve locking problem. AEOD staff had discussions through mid-1993 with licensees representing 31 operating plants to obtain information regarding licensees' evaluations on this subject. The discussions found that, despite licensees' efforts, most failed to either identify the gate valves susceptible to the problem or to implement corrective action. NRR and AEOD conducted a public workshop in February 1994 to discuss gate valve pressure locking and thermal binding technical issues. The workshop proceedings were issued as NUREG/CP-0146 in July 1995. NRC Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," was issued August 17, 1995.

APPENDIX G

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

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

In accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," dated August 12, 1992, upon receipt of an Incident Investigation Team (IIT) report, the Executive Director for Operations (EDO) shall identify and assign NRC office responsibility for potentially industry-generic and plant-specific actions resulting from the investigation that are safety significant and warrant additional attention or action. Office Directors designated by the EDO as having responsibility for the resolution of issues or concerns are responsible for providing written status reports on the disposition of assigned actions. Follow-up actions associated with the IIT report do not necessarily include all licensee actions, nor do they cover NRC staff activities associated with normal event follow-up, 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.

AEOD is responsible for monitoring the status of the assigned staff actions, evaluating the adequacy of the actions taken by the responsible office(s) to confirm that pertinent aspects of each IIT finding are addressed in the implemented resolution, and documenting the resolution of all staff actions. Actions whose resolution are reviewed and approved by the Commission are not subject to independent review by AEOD. The independent assessment should be completed by the end of the calendar year following the year in which the staff action was reported as resolved by the responsible office(s). The EDO resolves any conflicts between AEOD and the responsible office(s) regarding the adequacy of the actions taken by the staff.

This appendix summarizes the status of each of the action items that the EDO assigned to various NRC offices as a result of completed IITs at reactor facilities that were not documented as resolved in the 1994-FY95 AEOD Annual Report, NUREG-1272, Vol. 9, No.1.

AEOD IIT Tracking System

Action Source: **IIT Report on Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on 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 the following 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)

Status: Ongoing

The staff published a proposed rule for public comment (59 FR 52707-52714) on October 19, 1994 (Reference 3). Comments were received that documented significant impact upon the conduct of outages, inaccuracies in the regulatory analysis, and the need for an improved regulatory guide. Accordingly, the staff has redrafted the rule and is completing a new regulatory guide and regulatory analysis. Several meetings have been held with industry and more will be conducted to better assure that the potential impacts of the rule are not overlooked. The redrafted rule contains major revisions to the draft that was published in 1994. Based on the extent of these revisions, it has been decided to publish the rule a second time in draft form for public comment. It is anticipated that the second publication for comment will be before the end of 1997.

Though this action was documented as ***Resolved*** in the 1993 AEOD Annual Report based on publication of NUREG-1449 (Reference 4), it has been re-categorized as ***Ongoing*** pending publication of the final rule.

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: Resolved

Item 1: Adequacy of Shutdown Risk Management (cont.)

RES completed its review of alternate cooling methods as documented in NUREG/CR-5855, "Thermal-Hydraulic Processes During Reduced Inventory Operation with Loss of Residual Heat Removal." RES, in a memorandum dated August 15, 1991, provided recommendations to NRR regarding generic communications for guidance to licensees in planning options in the event of a loss of RHR. These recommendations were considered and the information was

AEOD IIT Tracking System

Item 1: Adequacy of Shutdown Risk Management (cont.)

incorporated as appropriate into NUREG-1449, "Shutdown and Low-Power Operations at Commercial Nuclear Power Plants in the United States," Final Report, September 1993.

RES has completed the final Phase 2 of the low power and shutdown risk project, which included computational models of mid-loop operational risk. Documentation of this shutdown risk study for Surry was completed with the publication of NUREG/CR-6144, Volume 1, in October 1995 (see References 7 and 8).

Risk assessment studies for low power and shutdown operations had already been initiated in support of the NRC's response to the 1986 Chernobyl accident and in part from the Diablo Canyon event of April 10, 1987. As a result, an analysis of the accident sequences leading to core damage for all plant operational states during low power and shutdown operations was performed for two specific plants, Surry and Grand Gulf. The study was limited to those plants because they were NUREG 1150 plants, had completed level 3 PRAs, and these licensees had also agreed to allow the staff access to plant data and to various design and procedural documentation. Phase 1 was an abridged risk study and was completed in May of 1992. Phase 2 was a more detailed study in which the accident frequency analysis was combined with the accident progression and consequence analysis to calculate risk. Completion of the Phase 2 portion of the study for Surry and Grand Gulf is documented as NUREG/CR 6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf Unit 1," and as NUREG/CR 6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit 1."

The conclusions reached in the Phase 2 study for both Surry and Grand Gulf were that, during mid-loop operation, risk consequences are high compared to those risks at full power, despite the much lower level of the core radionuclide inventory and decay heat level. This finding is in agreement with the general view held by the staff that risks during mid-loop operation are significant. These findings and conclusions were used as source material for the regulatory analysis of the draft shutdown rule, which is scheduled for a Commission briefing in July 1997. RES has also been reviewing plant IPEs to ensure that licensees have included some level of shutdown risk analyses. AEOD performed an independent review and found that the objective of establishing shutdown risk methodology has been completed. Alternate cooling methods for loss of RHR were documented in NUREG-1449 and in NUREG/CR 5769, "Natural Circulation Cooling in US PWRs," which was issued in January of 1992. This item is considered resolved.

AEOD IIT Tracking System

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

Status: Ongoing

As discussed in Item 1(a) above, the staff plans to continue development of a new rule. Publication of a draft rule for a second round of public comment is planned for 1997. The rule will be consistent with any changes in technical specifications and improved safety, while allowing more operational flexibility. Guidance regarding response to events is also to be included.

Although this action was documented as ***Resolved*** in the 1993 AEOD Annual Report based on publication of NUREG-1449 (Reference 4), it has remained open and re-categorized as ***Ongoing*** pending publication of the final rule.

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-offsite power event. Consider the priorities and requirements for notifications to offsite authorities. Follow up as appropriate. (Responsible Office: NRR)

Status: Ongoing

The staff will coordinate with the Nuclear Energy Institute to develop emergency classification guidelines for shutdown and low power operations as part of its follow-up work to NUREG-1449 (Reference 4). These guidelines will expand upon the current guidance already established for classification of emergencies. Accountability of personnel during outages is addressed in Section 6.12.2 of NUREG-1449. Licensees are expected to have plans and procedures in place to address the evacuation and accountability of the large numbers of personnel on-site during plant shutdowns or refuelings. The staff is continuing to assess the availability of notification systems and alternates during a loss-of-offsite power event and the priorities and requirements for notification of offsite authorities.

References:

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

AEOD IIT Tracking System

2. Memorandum from D.L. Morrison, Director, Office of Research to D.F. Ross, Acting Director, Office for Analysis and Evaluation of Operational Data, "Resolution of Vogtle Action Item 1B," dated January 22, 1997.

3. "Shutdown and Low-Power Operations for Nuclear Power Reactors," Federal Register (59 FR 52707), October 19, 1994.

4. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, Final Report," September 1993.

5. NUREG/CR-5855, "Thermal-Hydraulic Processes During Reduced Inventory Operation with Loss of Residual Heat Removal," April 1991.

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

7. NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf Unit 1," Volumes 2 through 5, June and July 1994.

8. NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit 1," Volumes 2 through 5, June and July 1994.

AEOD IIT Tracking System

Action Source: **IIT Report on the Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 on February 7, 1993 (Reference 1).**

Item 1: Adequacy of Regulations and Guidance for Protected Area Barriers, Entry Modes, and Design Basis Threat

Action (b): Evaluate the need for guidance for response to unauthorized forced entry into the protected area. Issue new guidance as appropriate. (Responsible Office: NRR/RES)

Status: Ongoing

In NUREG 1272, Volume 9, Number 1, it was reported that the responsible program office (NRR) had completed its review of this action item. During AEOD's independent review, issues were brought to the attention of the responsible program office. Accordingly, this action item will remain ***Ongoing***.

References: 1. NUREG-1485, "Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 on February 7, 1993," dated April 1993.

APPENDIX H

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

In accordance with Management Directive 8.7, "NRC Diagnostic Evaluation Program [DEP]," dated June 7, 1991, upon receipt of a Diagnostic Evaluation Team (DET) report, the EDO assigns NRC office responsibility for generic and plant-specific staff actions resulting from the Diagnostic Evaluation (DE). Office Directors designated by the EDO as having responsibility for resolving issues or concerns are responsible for providing written status reports on the disposition of assigned actions. The AEOD Director will maintain the status of the staff actions involving generic issues and will report them in the AEOD Annual Report.

This appendix summarizes the status of each of the open generic action items that the EDO assigned to various NRC offices as a result of completed DEs that were not documented as resolved in the 1994-FY95 AEOD Annual Report, NUREG-1272, Vol. 9, No. 1. As part of the NRC's streamlining effort, AEOD oversight and administration of the DEP ended in 1996. AEOD will, however, continue to track open generic actions from completed DEs until they are closed.

AEOD DET Action Tracking System

Action Source: **Memorandum from J. Taylor to Office Directors and Region III Administrator, "Staff Actions Resulting from the Diagnostic Evaluation at Quad Cities Nuclear Power Station," dated December 23, 1993 (Reference 1).**

Item 10: Staff Identification and Timely Resolution of Potentially Significant Licensee Safety Issues and Performance Problems

The DET observed that the licensee and NRC staff failed to recognize and/or appropriately evaluate degraded and/or nonconforming conditions. A number of potential operability issues existed, including those identified in the licensee's Vulnerability Assessment Team (VAT) report. However, neither the licensee nor the staff aggressively pursued resolution of the issues. Although the licensee had undertaken a number of improvement efforts in response to NRC-identified concerns, including issues identified during the Dresden and Zion diagnostic evaluations, many of these improvement initiatives were not completed, and corrective actions for many of the lessons learned from the Dresden and Zion evaluations were not implemented. Further, although staff inspections and oversight reviews of Quad Cities conducted just prior to the DET identified significant performance issues, inspection activities and reviews conducted earlier in the twelve month period prior to the DET did not fully convey the broad performance problems and weaknesses identified by the DET. The STP and Fitzpatrick DETs also provided an integration of plant performance which were significantly more negative than that provided by the previous respective inspection and review activities.

Action (a): Evaluate the need to provide additional training and/or guidance to the staff on actions to be taken when information on safety issues potentially impacting equipment operability is received by the staff. (Responsible Office: NRR/AEOD)

Status: Resolved

NRC Inspection Manual, Part 9900: "Technical Guidance," consists of the following two inserts, "Degraded and Non-Conforming Equipment" and "Operability." This guidance was developed after issuance of guidance for public comment in 1990 and was followed up with workshops in 1992 and 1993. It is presently being evaluated for potential revisions stemming from lessons learned from more recent plant events. This issue will be tracked by the ADP Process Improvement Plan, Item 30b. This issue is closed as a Quad Cities DET issue.

The DET finding also addressed the need for improved integration of inspection findings. Recent initiatives to improve the NRC's assessment capabilities have been made to strengthen integration of findings within the assessment processes. These initiatives include the issuance of a new Inspection Manual Chapter that provides

AEOD DET Action Tracking System

Item 10: (cont.)

guidance for accomplishing plant performance reviews (PPRs); establishing programmatic requirements for the plant issues matrix (PIM) such that each site will have a PIM; and using the PPR results for input to the NRR prebriefs and senior management meetings. This will provide senior management consistent data for their assessment processes.

References:

1. Memorandum from James M. Taylor to Office Directors and Region III Administrator, "Staff Actions Resulting from the Diagnostic Evaluation at Quad Cities Nuclear Power Station," dated December 23, 1993.

2. Memorandum from Frank J. Miraglia to James M. Taylor, "NRR Staff Actions Resulting from the Diagnostic Evaluation at Quad Cities, Units 1 and 2," November 4, 1996

AEOD DET Action Tracking System

Action Source: **Memorandum from J. Taylor to Office Directors and Region IV Administrator, "Staff Actions Resulting From The Special Inspection of Cooper Nuclear Station," dated December 22, 1994 (Reference 1).**

Item 3: NRC Headquarters Personnel Radiation Dosimetry

Regional representatives of the SET used NRC-issued dosimetry in addition to that supplied by the licensee. The SET team members from other offices did not have NRC issued dosimetry. NRC Manual Chapter 0524, "Standards for Protection Against Ionizing Radiation," provides general guidance for NRC staff and NRR Office Letter No. 1303, Revision 1, "Radiation Protection Procedures for NRR Personnel," provides specific guidance to NRR staff. Additionally, regional instructions provide for issuance and use of NRC-supplied dosimetry for personnel who travel to licensee facilities. However, the guidance for issuance, use, and monitoring of dosimetry by headquarters personnel does not appear to be generally known. This issue could be critical for individuals visiting plants outside the U.S. who are not subject to the monitoring standards of the Code of Federal Regulations. Currently the Office of Personnel has lead responsibility for the development of a Management Directive to establish an agency-wide personnel dosimetry program.

Action: Assess the level of compliance with NRC Manual Chapter 0524 and other Headquarters guidance regarding the issuance, use, and monitoring of personnel dosimetry. Evaluate the need to develop and issue additional guidance and procedures and provide training to ensure a consistent policy is generally known and complied with. (Responsible Office: OP/NRR/NMSS/AEOD)

Status: Resolved

This action was assigned on December 22, 1994, in a memorandum from J. Taylor to W. Russell, L. Callan, E. Jordan, R. Bernero, and P. Bird (Reference 1). NMSS responded that the level of compliance by NMSS employees is high and no additional guidance or procedures are required for NMSS staff (memorandum from R. Bernero to J. Taylor dated January 17, 1995 [Reference 2]). AEOD responded that additional guidance is needed at the staff level to inform AEOD staff of the requirements and procedures for obtaining and using NRC-supplied personnel monitoring dosimetry while at facilities that require it (memorandum from E. Jordan to J. Taylor dated March 20, 1995 [Reference 3]). AEOD is presently developing internal guidance and procedures for the issuance and collection of personnel monitoring dosimetry and tracking of personnel radiation exposures.

On April 14, 1995, J. Milhoan met with P. Bird, R. Bernero, M. Knapp, and W. Russell to discuss the NRC radiation protection program. It was decided that, before coming to

AEOD DET Action Tracking System

Item 3: (cont.)

final resolution on Issue 3, the staff would examine where efficiencies could be achieved in the radiation protection area. Per a memorandum from J. Milhoan to C. Paperiello, W. Russell, E. Jordan, and P. Bird dated May 4, 1995, (Reference 4) NMSS was assigned the lead for reviewing efficiencies to be obtained in the personnel monitoring area. This included taking the lead for updating and publishing Management Directive MD 10.131, "Protection of Employees Against Ionizing Radiation," formerly NRC Manual Chapter 0524. NRR was assigned the lead for reviewing efficiencies to be obtained in the environmental monitoring and laboratory programs.

On June 9, 1995, NMSS issued a memorandum from J. Paperiello to J. Milhoan (Reference 5) providing the results its review of NRC personnel monitoring. This review was coordinated with the offices of NRR, AEOD, RES, OP, and the Regions. The review concluded that the NRC should maintain the existing program as it provides a convenient, low cost method of recording the occupational doses of NRC employees, and allows office management the flexibility to use licensee dosimeters if they choose. The program also provides a defense against potential claims of damages incurred by exposure to radiation during employment at NRC. NMSS issued MD 10.131 on July 23, 1996.

Item 4: Use of temporary modifications in emergency operating procedures without verifying that the modifications could be installed given staffing and timing constraints.

While performing the Special Evaluation of the Cooper Nuclear Station, it was discovered that emergency operating procedures (EOPs) contained a total of 58 plant temporary modifications (PTMs) which would be implemented during execution of the EOPs. Most of the PTMs involved adding jumpers to or lifting leads from the control room instrument panel back-plane. Several weaknesses included the following: (1) some PTMs were never tested to verify that they would perform as designed, (2) the radiological evaluation did not consider potential doses to the operator from the TS-assumed design basis containment leak rate (or some reduced leak rate) into the reactor building, (3) 31 of PTMs would be installed outside the control room, and (4) no evaluation was made in the verification and validation of the EOP procedures to determine the time or staff needed to install the PTMs. NRR does not give credit for operator intervention to realign manual fluid systems during the first 20 minutes after the start of an event (e.g., start of drywell spray on a BWR). During the first 20 minutes following an ATWS event, possibly 10 PTMs would have to be installed outside the control room. Information obtained from Senior Resident Inspectors regarding the use of PTMs in EOPs at other stations showed that Susquehanna had

AEOD DET Action Tracking System

Item 4: (cont.)

approximately 155 per unit, Limerick had approximately 90 per unit, and Monticello had approximately 115.

Action: Evaluate (a) the significance and number of PTMs which could reasonably be installed in a plant during the early phases of an event which would require entry into EOPs and not degrade safety, and (b) the need to assess the proficiency of the operations crew to implement PTMs during operator license examinations. Provide guidance as necessary. (Responsible Office: NRR)

Status: Resolved

The staff recognized the necessity to perform a limited number of PTMs in accordance with plant EOPs during the initial hour of certain events. The adequacy of shift staffing and response time in regard to these essential PTMs varies due to the particular event, the plant-specific needs, and the plant-specific task allocation scheme used.

As part of the NRC Emergency Operating Procedure Inspection Program, the staff conducted a review of EOPs, EOP useability, and the EOP development process, paying particular attention to the validation and verification (V&V) activities at each operating nuclear power reactor facility. Region-led follow-up EOP inspections, conducted in accordance with Inspection Procedure 42001, "Emergency Operating Procedures," continue to evaluate EOPs and EOP programs, including the V&V activities.

The staff reviewed the results of the 22 EOP inspections conducted over a two year period. The review focussed on the use of PTMs and the V&V of procedural steps associated with them, particularly the staffing, timing, and environmental constraints. Although two inspection reports identified and addressed some plant-specific problems with emergency changes in plant configuration (e.g., installation of spool pieces) with respect to V&V, and staffing, timing, and environmental constraints, none of the inspections identified problems with the use of PTMs in implementing plant emergency operating procedures. The staff evaluated the results of this review, determined that the use of temporary modifications in plant operating procedures has not been a problem in general, and concluded that no further staff action is warranted in this area. This item is considered closed (see Reference 6).

Item 6: Safety-related equipment testing did not always assure operability. Significant weaknesses were recently identified in the licensee's testing and

AEOD DET Action Tracking System

Item 6: (cont.)

surveillance programs for safety-related systems and components. Deficiencies were found by the SET, regional inspectors, the licensee, and the DSA team. Identified weaknesses included pre-conditioning of equipment to assure passage of tests, and incomplete functional testing of safety-related system actuation logic. Additionally, surveillance procedures did not contain all required TS attributes, post-modification and post-maintenance testing was incomplete or not effectively planned, and preventive maintenance was ineffective in assuring equipment operability. Excessive testing resulted in plant challenges or degraded equipment while ineffective test result trending obscured declining equipment performance and the need for actions to correct problems before failure occurred. The SET report documents a number of testing weaknesses which substantially degraded the licensee's system operability assurance process. The SET results, together with previous DET findings for other facilities, indicate that licensee testing and surveillance programs vary significantly in their ability to detect or predict non-functionality or failures of systems and components. This situation appears to continue despite considerable operational experience feedback in the form of Information Notices, Bulletins, Generic Letters, and industry correspondence.

Action: Review the SET and previous DET reports to evaluate testing weaknesses in assuring operability. Identify any changes that could be made to improve the effectiveness of testing programs for assuring operational safety.(Responsible Office: AEOD)

Status: Ongoing

AEOD has completed a review of the 14 DET reports, the Cooper SET report, and the Maine Yankee Independent Safety Assessment (ISA) to identify common test program/implementation weaknesses. Numerous examples existed where these teams discovered significant testing program weaknesses that were not found or clearly understood through implementation of the routine NRC inspection program. Licensee implementation of inservice testing programs was a weakness in 9 of 15 DET/SET inspections. A major finding in the Maine Yankee ISA was the lack of a questioning attitude, which resulted in the use of poor surveillance procedures and the ineffective evaluation of surveillance test data to determine equipment operability.

AEOD has also substantially completed a testing study to assess testing effectiveness, which included a review of the 2295 LERs containing operating events from 1991 through 1993 on 69 individual plants. Major findings include the following:

- Of the 69 plants reviewed, 17 were responsible for approximately 50 percent of the events, many of which could have been detected/prevented through adequate testing, but were not.
- About 44 percent of the LERs involving system or equipment malfunctions were detected by testing.
- Human error during testing was responsible for 241 of the events.

AEOD DET Action Tracking System

- Test program weaknesses failed to prevent 393 of the events, of which 141 occurred because of untimely response to previous tests that had identified problems.
- Licensees were not effectively using test results to effect improvements.
- Several examples of incomplete control logic testing were noted.
- Emergency power supply testing was identified as a potential problem as the testing was not always complete.

Lessons learned from AEOD testing reviews support recommendations made in the Millstone Lessons Learned Task Group Report issued in September 1996 which recommends using a design-based team inspection approach. Design-based testing inspections would assist in determining the degree to which licensees demonstrate through testing that the plant existing condition meets appropriate design requirements and whether design margin has been reduced. In addition, staff actions resulting from the Maine Yankee ISA, which ask the staff to evaluate the need to improve the inspection process for testing and design areas, will further address this issue. When the AEOD report associated with these findings is issued, it will be the basis for closing this action item (Reference 7). Expected completion date is July 1, 1997.

References:

1. Memorandum from J. Taylor to W. Russell, L. Callan, E. Jordan, R. Bernero, and P. Bird. "Staff Actions Resulting From The Special Evaluation At Cooper Nuclear Station," dated December 22, 1994.

2. Memorandum from R. Bernero to J. Taylor, "Status of NMSS Staff Action Resulting From The Special Evaluation At The Cooper Nuclear Station," dated January 17,1995.

3. Memorandum from E. Jordan to J. Taylor, "Status of AEOD Staff Actions Resulting From The Special Evaluation At The Cooper Nuclear Station." dated March 20, 1995.

4. Memorandum from J. Milhoan to C. Paperiello, W. Russell, E. Jordan, and P. Bird, "Schedule For Radiation Protection Activity Review," dated May 4, 1995.

5. Memorandum from J. Paperiello to J. Milhoan dated June 9, 1995, providing the results of the review of NRC personnel monitoring.

6. Memorandum from W. Russell to J. Taylor, "NRR Staff Actions Resulting From The Special Evaluation At Cooper Nuclear Station," dated April 11, 1995.

7. Memorandum from E. Jordan to J. Taylor, "Status of AEOD Staff Actions Resulting From The Special Evaluation At The Cooper Nuclear Station," dated November 12, 1996.

APPENDIX I

Status of NRC Staff Actions Involving Potential Generic Issues Resulting From the NRC/INPO Team Review of the Effects of Hurricane Andrew on Turkey Point Units 3 and 4

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Action Source: **Memorandum from J. Taylor to Office Directors and Regional Administrators, "Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992," dated May 28, 1993 (Reference 1).**

Item 2: Adequacy of Licensee Offsite Communications for Natural Disasters Within the Plant Design Basis

Although diverse and redundant communications equipment existed, offsite communications were lost during the storm due to a common vulnerability to wind damage. Normal telephone service failed because the storm blew down the lines near the station. The dedicated commercial telephone lines servicing the telephones installed in the control room, the Technical Support Center, and the Emergency Operations Facility, used to give initial notification and status to the State in an emergency, also failed. The Federal Telecommunications System - 2000 lines used for the Emergency Notification System failed, cutting off normal communications with the NRC Operations Center. The cellular telephone systems also did not function because the storm damaged the on-site antennas and the offsite repeating stations. Except for the Security Department's one hand held radio for the company FM radio system, the licensee's radio systems did not function during and immediately following the storm. Overall, all offsite communications were lost for about four hours during the storm, and reliable communications were not restored for about 24 hours following the storm. The NRC's temporary satellite communications system considerably aided recovery efforts and would have been more beneficial if it had been onsite before the storm.

Action (a): Review the existing regulatory guidance and requirements related to normal and backup offsite communications system design capabilities for hurricanes. Based on this review, consider the adequacy of the guidance for other external events. Issue revised guidance or requirements as may be needed. (Responsible Office: NRR/AEOD)

Status: Ongoing

NRR has reviewed current regulations and regulatory guidance and identified rules and guidance that apply to offsite communications systems. In coordination with the Federal Emergency Management Agency and the NRC technical branches responsible for requirements on licensee communications, NRR reviewed the identified rules and guidance to determine whether they adequately account for external events.

NRR staff concluded that the requirements and guidance are sufficiently detailed to provide licensees with the staff's expectations of the capability of offsite communications to function during and following severe natural events. Notwithstanding this conclusion, however, there is insufficient information on the

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Item 2: (cont.)

existing offsite communications capabilities at nuclear power plant sites to conclude that the problems identified with the loss-of-offsite communications at Turkey Point are not pervasive in the industry. Therefore, to make a determination whether generic action is warranted to ensure compliance with the regulations, information was obtained on the offsite communication systems at a sampling of sites. Information was gathered during routine regional inspections scheduled between February and June 1996 using Temporary Instruction 2515/131, issued on January 18, 1996. NRR evaluated the inspection findings to determine whether guidance to the licensees in the form of a generic communication is necessary to ensure either survivability or rapid recoverability of these circuits from a severe natural event. On the basis of its review of the results of TI 2515/131, NRR concluded that additional guidance to the industry is warranted. Consequently, NRR will pursue issuance of an inforation notice to all reactor licensees informing them of the results of the TI. NRR will also pursue changes to the emergency preparedness inspection procedure for power reactor licensees to give guidance for review of licensees' offsite communication circuits as part of the core inspection program. The actions are scheduled for completion by January 1997 (Reference 2).

Item 4: Adequacy of NRC Guidance For Reviewing Licensee Preparation and Response to Natural Disasters and Industry Preplanned Support.

The Turkey Point Plant benefited greatly from the prior hurricane experience of the plant staff and the extensive preplanning done in preparing and implementing the licensee's Emergency Plan Implementing Procedure (EPIP) 20106 for "Natural Emergencies." The EPIP was also significantly expanded as a result of the insights gained, in part, from the Individual Plant Examination for Turkey Point. These additional procedures, which dealt with preparations for a Category 5 hurricane, contributed significantly to the licensee's preparations. In the aftermath of the hurricane the licensee had to take numerous extraordinary actions to establish a support services infrastructure which would allow the station staff to report to the plant each day. Such circumstances could potentially be more extreme following other external events (e.g., severe earthquake) for which there was no warning to permit advance preparations including the evacuation of families of plant personnel. The assistance provided by the St. Lucie plant in meeting Turkey Point's immediate and longer term needs such as personnel, spare parts and supplies, were helpful to the recovery.

Action (a): Consider the need for development of additional guidance for review of licensee preparations for a predicted hurricane. Develop and issue staff guidance as appropriate. (Responsible Office: NRR/Regions)

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Action (b): Consider the need for development of guidance for review of licensee preplanning for response to other external events. Develop and issue staff guidance as appropriate. (Responsible Office: NRR/Regions)

Status: Resolved

The staff has concluded that, from an emergency preparedness standpoint, sufficient guidance exists for reviewing licensee preparations in response to a hurricane or other external event. The staff issued Information Notice 93-53, Supplement 1, on April 29, 1994, to expand the scope of lessons learned to other external events and to discuss existing regulatory guidance for various external events. The action to provide guidance for inspectors to address any vulnerabilities that may develop from the review of Individual Plant Examinations of External Events (IPEEE) (Generic Letter 88-20, Supplement 4) has been incorporated into the Probabilistic Risk Assessment Implementation Plan (Reference 2).

Action (c): Coordinate with industry in consideration of preplanned measures to supplement individual utility resources to maintain adequate staffing and critical supplies immediately following a severe external event. (Responsible Office: AEOD/NRR)

Status: Resolved

The staff met with the Institute of Nuclear Power Operations (INPO) to discuss the plans and capabilities currently available under the Letters of Agreement between INPO and its member utilities. The meeting was conducted between the NRC, INPO, and the Nuclear Management and Resources Council on July 23, 1993. The meeting consisted of a presentation from INPO specifying the capabilities and procedures that were currently in place to coordinate and provide support to a nuclear facility during a time of emergency. The details of the program are included in INPO 86-032, Revision 6.

The staff has included communications with INPO into the goals and objectives which the NRC pursues during exercises with nuclear licensees. Communications with INPO was included in the 1994 Operations Center shakedown drills.

The staff originally planned to conduct a table top exercise, in conjunction with INPO, to test the resource brokering capabilities and procedures identified in INPO 86-032. The staff has subsequently determined that such an exercise is unnecessary for the following reasons.

First, licensees have successfully demonstrated during real events that they can obtain voluntary assistance from other utilities, either directly, or through a resource broker. This was illustrated during the event at the Wolf Creek Generating Station on January 30, 1996, involving the buildup of frazil ice on the trash racks of the essential service water intake bays. During this event, the licensee contacted the Utility Services Alliance to locate skid mounted diesel generators that could be brought to the

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site--one of which was located at the Cooper Nuclear Station. The licensee also contacted Fort Calhoun and other northern plants to seek advice on how to deal with icing problems. NRC also demonstrated its support capability by putting the licensee in contact with Steven F. Daly of the U.S. Army Corps of Engineers, the foremost authority on the frazil ice phenomenon.

Second, the NRC will continue to encourage the industry to maintain preplanned measures to supplement utility resources during an emergency, such as those outlined in the INPO Emergency Resources Manual, and will periodically discuss INPO's role at NRC/INPO management meetings. (see Reference 3).

References:

1. Memorandum from J. Taylor to Office Directors and Regional Administrators, "Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992," dated May 28, 1993.

2. Memorandum from F. Miraglia, Acting Director, to J. Taylor, "Fourth Annual Status Report - Office of Nuclear Reactor Regulation Generic Follow-On Actions to NUREG-1474, "Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station From August 20-30, 1992," dated November 4, 1996.

3. Memorandum from E. Jordan, Director, to J. Taylor, "Staff Actions in Response to the Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station--August 20-30, 1992," dated April 19, 1996.

APPENDIX J

Status of NRC Staff Actions Resulting From the Independent Safety Assessment of the Maine Yankee Atomic Power Station

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Action Source: **Memorandum from J. Taylor to Office Directors and the Region I Administrator, "Staff Actions Resulting From The Independent Safety Assessment of the Maine Yankee Atomic Power Station," dated November 27, 1996 (Reference 1).**

Item 1: Adequacy of Analytic Code Validation

The ISA team noted that the plant specific validation of RETRAN by Maine Yankee to known industry benchmarks for integral and separate effects test data was deficient. This validation is important to assure that the plant-specific application of the code effectively models known physical effects. The team found the NRC requirement for this validation to be vague. The single document which states NRC policy on this issue is Generic Letter 83-11, "Licensee Qualifications for Performing Safety Analysis in support of Licensing Actions," issued on February 8, 1993, which states, in part:

> ... some licensees planning to perform their own safety analyses may not intend to demonstrate their ability to use the code by performing their own code verification. Rather, they plan to rely on the code verification work previously performed by the code developer or others.
>
> NRR does not consider this acceptable and each licensee or vendor who intends to use a safety analysis computer code to support licensing actions should demonstrate their proficiency in using the code by submitting code verification performed by them, not others.

Additionally, the team found that the NRC has acted inconsistently relative to its expectations in this area. In some cases, computer codes have been endorsed for use with little or no validation accomplished.

Action: Evaluate the agency's expectations and policy relative to code validation. Develop and issue additional guidance and requirements if appropriate, and develop and implement inspection methodology to verify licensee conformance as appropriate. (Generic: NRR/RES)

Status: Ongoing

Item 2: Adequacy of NRC review of analysis codes

The ISA formed an expert panel of consultants with extensive experience in the area of analysis code development to assess and critique the results of the ISA's efforts. The reports submitted by these consultants (attached) included observations and suggestions for improving the NRC's process of reviewing analytic codes.

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Action: Review the attached consultant reports and evaluate the need to make changes to the existing NRC processes as suggested in the reports. Implement changes as appropriate. (Generic: NRR/RES)

Status: Ongoing

Item 3: Compliance With Safety Evaluation Reports

During the Maine Yankee ISA, compliance with safety evaluation report (SER) conditions imposed on the use of analytic codes was verified for 66 conditions effecting 13 codes. While compliance was confirmed, an audit trail to assure compliance was not always available, necessitating, in come cases, additional analyses to verify compliance. The team found that the Regulatory status of an SER condition was unclear.

Additionally, the ISA team found that the quality of NRC code reviews was mixed. This may have stemmed from the fact that there was no standard review plan for code reviews. Consequently, no guidance or requirements existed for: development of an agreed upon set of identified and ranked phenomena, processes, or key parameters; validation; code modeling detail; sensitivity studies; or peer review by experts in the field.

Action (a): Evaluate the agency's expectations regarding the tracking and closeout of SER conditions relative to compliance, auditability, and reportability. Issue appropriate industry and inspection guidance as needed. (Generic: NRR)

Status: Ongoing

Action (b): Evaluate the need to develop a standard review plan for code reviews. Develop and issue appropriate guidance. (Generic: NRR)

Status: Ongoing

Item 4: Adequacy of Licensing Reviews for Power Uprates

The ISA team identified a number of mechanical components for which confirmation of operability at the upgraded power level of 2700 MW_t could not be confirmed. Additionally, the team noted that documentation of NRC actions on parameters related to the design and licensing bases for Maine Yankee was not identifiable and retrievable from NRC sources.

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Action (b): Evaluate the Agency's process for technical review and approval of licensee requested power uprates. Implement changes to the process as appropriate. Based on the results of this review, determine whether any previously approved power uprates should be reevaluated and to what extent. (Generic: NRR)

Status: Ongoing

Action (c): Evaluate the need and the feasibility of establishing an NRC licensing and design bases database for all plants to centrally collect all documentation necessary to support plant licensing. Take actions as appropriate. (Generic: NRR)

Status: Ongoing

Item 5: Clarity and Intent of NRC Safety Guide 1

During the ISA review of containment spray system and high pressure safety injection system net positive suction head (NPSH), the team found the guidance provided by NRC Safety Guide 1, "Net Positive Suction head for Emergency Core Cooling and Containment Heat Removal System Pumps," issued on November 2, 1970, to be problematical with regard to relying on containment pressure for assuring NPSH for emergency core cooling and containment heat removal pumps.

NRC Safety Guide 1 states, "NPSH for emergency core cooling and containment heat removal system pumps caused by increases in temperature of the pumped fluid under loss of coolant accident conditions can be accommodated without reliance on the calculated increase in containment pressure." Furthermore it states: "Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss of coolant accidents."

Maine Yankee asserted that they were not committed to Safety Guide I. Consequently, they assumed containment to be at or above the saturation pressure for the sump fluid temp rather than at pre-accident containment pressure (nominally atmospheric). The issue of whether or not the containment can be assumed to be pressurized at the saturation pressure for the sump fluid temperature should be addressed.

Action: Review and clarify the staff's criteria relative to relying on containment overpressure for ensuring appropriate NPSH for emergency core cooling and containment heat removal pumps. The staff is already conducting a separate program to determine if and how all plants, including Maine Yankee, meet these criteria. Upon review of this information, the staff will determine the measures to be taken for those plants not in compliance with the criteria. (Generic: RES/NRR)

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Status: Ongoing

Item 6: Adequacy of the NRC Inspection Program

The ISA team observed that the licensee and NRC staff failed to recognize and/or appropriately evaluate degraded/nonconforming conditions. A number of potential operability issues existed; however, neither the licensee nor the staff aggressively pursued resolution of the issues. Further, although staff inspections and oversight reviews of Maine Yankee conducted prior to the ISA identified significant performance issues, they did not fully convey the broad performance problems and weaknesses identified by the ISA team. These issues included problems with safety system testing programs, licensee-developed technical specification interpretations, and design basis adequacy.

Action (a): Evaluate the inspection program and inspector training and guidance with regard to testing programs for safety systems relative to its scope, rigor, and analysis of results. Implement inspection program changes and develop new guidance as appropriate. (Generic: NRR)

Status: Ongoing

Action (b): Evaluate the inspection program and guidance with regard to review of licensee developed technical specification interpretations to assure consistency with the intent of the approved technical specifications. Implement inspection program changes and develop new guidance as appropriate. (Generic: NRR)

Status: Ongoing

Action (c): Evaluate the inspection program and guidance with regard to the assessment of the adequacy of plant design basis, including a review of the disposition of significant findings from previous licensee efforts such as design basis documentation or design basis reconstitution programs. (Generic: NRR)

Status: Ongoing

Item 7: Adequacy of Agency Expectations Regarding Licensee Performance

The ISA relied on the existing agency benchmark for assessing performance utilized in the NRC Systematic Assessment of Licensee Performance Program (SALP). Although SALP category rating definitions, functional areas, and assessment criteria have evolved over time, the Commission raised questions about the SALP definitions. In addition, a number of questions were raised during the October 10, 1996, public meeting on the ISA findings at Wiscasset Maine.

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Action: Evaluate the appropriateness of the existing SALP definitions of superior, good, and acceptable performance in light of the NRC's contemporary expectations for licensee performance. Revise these definitions as necessary. (Generic: NRR)

Status: Ongoing

Item 8: Cumulative Effect of Operator Workarounds

The ISA found that operators at Maine Yankee were required to take numerous actions to compensate for weaknesses in plant design. Some of these would require operators to take time consuming manual actions such as donning steam suits and deploying a 350 foot extension cord during significant plant transients. Additionally, the team found that Maine Yankee had been slow to resolve a work allocation issue which appeared to direct the two on-shift senior operators to leave the control room in the event of a fire coincident with a medical emergency. The cumulative effect of all these actions had not been evaluated by the licensee or the NRC.

Current NRC policy which would restrict credit for operator action or define the time which may be available for operators to take action is limited. The staff has typically relied on guidance provided in ANSI/ANS 58.8, "Time Response Design Criteria for Safety Related Operator Actions." However, the staff has allowed deviations from this guidance when licensee's have provided empirical evidence that operators can take the required actions within the required time constraints.

Action (a): Evaluate the current guidance and policies with regard to the cumulative effect of operator workarounds. Develop and issue additional or revised guidance as appropriate. (Generic: NRR/RES)

Status: Ongoing

Action (b): Evaluate the need to develop inspection policy and guidance directed at assessing the cumulative effect of operator workarounds. Develop and issue guidance as appropriate. (Generic: NRR)

Status: Ongoing

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Item 9: Agency Policy Regarding Licensee Design Basis Recovery Efforts

The ISA team found that the licensee had identified significant design bases issues involving safety-related systems as part of their Design Basis Reconstitution (DBR) program. DBR reviews had been performed in ten functional areas. Another nine functional areas had been scheduled, but had been delayed due to resource limitations and priority changes. These areas included the emergency diesel generator, electrical distribution, and ventilation. The licensee and the ISA team found design weaknesses in each of these areas.

Action: Evaluate the current Agency policy regarding licensee design basis recovery efforts. Consider the need to require or encourage licensees to accelerate and complete efforts to recover and reconstitute their design basis, especially older facilities where some information may be missing, difficult to find, or inaccurate. (Generic: NRR)

Status: Ongoing

Item 10: Public Involvement in the Assessment Process

The planning and conduct of the ISA included extensive State participation through three team members, two process reviewers, a five member citizen's group and periodic briefings with the Governor, a public observation entrance meeting and a public participation meeting to convey the findings. However, the team received complaints during the public meeting and via written correspondence that there was insufficient opportunity for "public participation."

Action (a): Evaluate the need to provide guidance for public participation (via a two part meeting) at the beginning of a review to explain and discuss the scope and objective of the review. Develop and issue guidance as appropriate. (Generic: AEOD/OPA/NRR)

Status: Ongoing

Action (b): Evaluate the need to allow at least one week from issuance of a report to a "public participation meeting" on the findings. Revise or issue new guidance as appropriate. (Generic: AEOD/NRR)

Status: Ongoing

Action (c): Evaluate the need to make additional copies of the entire report available by sending multiple copies to the local PDR in addition to Internet access. Revise or issue new guidance as appropriate. (Generic: AEOD/NRR)

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References: (1) Memorandum from J. Taylor to Office Directors and Region I Administrator, "Staff Actions Resulting From The Independent Safety Assessment of the Maine Yankee Atomic Power Station," dated November 27, 1996.

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Same as in item 8

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) describes activities conducted during 1996. The report is published in three parts. NUREG-1272, Vol. 10, 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 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 and reports to the NRC's Operations Center. NUREG-1272, Vol. 10, No. 2, covers nuclear materials and presents a review of the events and concerns during 1996 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued from CY 1980 through 1996. NUREG-1272, Vol. 10, No. 3, covers technical training and presents the activities of the Technical Training Center in support of the NRC's mission in 1996.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)		13. AVAILABILITY STATEMENT
nuclear plants	safety performance	unlimited
operating experience	AEOD recommendations	14. SECURITY CLASSIFICATION
abnormal occurrences	AEOD studies	(This Page) unclassified
significant events	Incident Investigation Program	
reactor trips	Diagnostic Evaluation Program	(This Report) unclassified
performance indicators	Committee to Review Generic Requirements	
NRC staff actions	system reliability	15. NUMBER OF PAGES
operating factors	accident sequence precursors	
incident response	allegations	16. PRICE