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Analysis of Reactor Trips Originating in Balance of Plant Systems

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Science Applications International Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

This report documents the results of an analysis of balance-of-plant (BOP) related reactor trips at commercial U.S. nuclear power plants over a 5-year period, from January 1, 1984, through December 31, 1988. The study was performed for the Plant Systems Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The objectives of the study were:

1. to improve the level of understanding of BOP-related challenges to safety systems by identifying and categorizing such events;

2. to prepare a computerized data base of BOP-related reactor trip events and use the data base to identify trends and patterns in the population of these events;

3. to investigate the risk implications of BOP events that challenge safety systems;

4. to provide recommendations on how to address BOP-related concerns in a regulatory context.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS (Continued)

EXECUTIVE SUMMARY

This report documents the results of a characterization and subsequent analysis of balance-of-plant (BOP)-related reactor trips at commercial U.S. nuclear power plants over the 5-year period from January 1, 1984 through December 31, 1988. The study was performed for the Plant Systems Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

Objectives

The objectives of the study were to:

1. improve the level of understanding of BOP-related challenges to safety systems by identifying and categorizing such events;

2. prepare a computerized data base of BOP-related reactor trip events and use the data base to identify trends and patterns in these events;

3. investigate the risk implications of BOP events that challenge safety systems; and

4. provide recommendations on how to address BOP-related concerns in a regulatory context.

Sources of BOP Information

The primary sources of information used in the study were:

- an earlier investigation of BOP events reported in NUREG/CR-4783, BOP Regulatory Issues, January 1987;

- Licensee Event Reports (LERs) accessed through the Sequence Coding and Search System (SCSS) maintained for USNRC by Oak Ridge National Laboratory; and

- a study by the NRC Office for Analysis and Evaluation of Operational Data (AEOD), NUREG-1275 Volume 5, Operating Experience Feedback Report - Progress in Scram Reduction, March 1989.

Additional sources of information were industry organizations (e.g., INPO, Owners Groups), NRC documents (e.g., NUREG and NUREG/CR reports, AEOD reports, inspection reports, generic letters, notices and bulletins) and information on foreign scram reduction programs, e.g., Proceedings of a Nuclear Energy Agency Symposium on Reducing the Frequency of Nuclear Reactor Scrams, Tokyo, Japan, 1986.

BOP Data Base Development

As part of this study, a computerized data base of BOP-related reactor trips was created, based on information provided in Licensee Event Reports (LERs) over the period January 1, 1984, through December 31, 1988. The Sequence Coding and Search System (SCSS) for the LER data base was used to identify potentially relevant LERs. The LER search on 47 BOP-related SCSS codes produced 2030 LERs with some level of BOP involvement.

The 2030 LER printouts were examined individually against predetermined criteria for BOP relevance, and 1405 events were considered appropriate for entry into a BOP trip data base. LERs were not included in the BOP trip data base if any of the following conditions applied:

- BOP involvement was incidental to the reactor trip, i.e., not in the causation sequence.

- The trip occurred during special tests or evolutions during shutdown conditions and would not have occurred when the reactor was critical or at power. Events occurring at shutdown conditions that could have occurred at power or with substantial decay heat in the core were included in the BOP trip data base.

- The trip resulted from loss of offsite power or other events external to the plant systems.

The BOP trip data base was developed on PC-dBase III Plus software. Each event record identifies the BOP system (e.g., feedwater), subsystem (e.g., feedwater control), and component (e.g., feedwater control valve) as applicable. Up to three potential causes of the event may be specified. A narrative event description is also provided.

Supplementary data bases were also found to be necessary for conducting analyses of trends and patterns. The supplementary data bases contain plant data and critical hours data. The supplementary plant data base includes the following data elements:

- o Operating license (OL) date
- o Nuclear Steam Supply System (NSSS) vendor
- o Architect/engineer
- o Turbine/generator manufacturer.

The critical hours supplementary data base includes:

- o Critical hours per year for each plant of the years 1984 through 1988
- o Total critical years accumulated during the period 1984 through 1988.

Trend and Pattern Evaluations

The 5-year, 1405-event BOP trip data base was searched for trends and patterns in the data. Searches were performed on BOP trips per plant per calendar year; BOP trips per plant per critical year; general causation of BOP trips (i.e., component failure, human-related, design-related, etc.); multiple cause BOP trips; systems, subsystems, and components implicated in BOP trip causation; and trend observations by architect/engineer, plant age and plant power level at trip. Several special searches (e.g., feedwater trips by NSSS vendor by year) were also performed to help understand the results of earlier searches.

A basic distinction was made between mature plants and new plants. Mature plants were defined as those which received operating licenses (OLs) before

January 1, 1983. Thus, all plants in the mature plant category had held an OL for at least one year before the start of the LER period covered by the study--January 1, 1984. This definition of mature plants resulted in a constant population of 76 plants for trend and pattern analysis. A "floating" definition of mature plants (e.g., 1 or 2 years after the OL date) was considered but not adopted because it would have introduced another variable (plant population) into the trend and pattern analysis.

BOP Trips per Calendar Year

The mature nuclear plants showed a substantial reduction in BOP trips over the 5-year period, from an average of 2.8 BOP trips per calendar year in 1984 to 1.6 BOP trips per calendar year in 1988.

BOP Trips per Critical Year

The mature nuclear plants showed a substantial reduction in BOP trips over the 5-year period, from an average of 4.4 BOP trips per critical year in 1984 to an average of 2.3 BOP trips per critical year in 1988.

General Causation of BOP Trips

The general causation categories used in the study were component failure, human-related, design-related, procedure-related, and spurious or unknown. Nearly half (47 percent) of the BOP trips were caused by one or more component failures, and nearly one-third (31 percent) were human-related. The human-related BOP trips were further categorized by the activity in progress as follows: 40 percent operations, 40 percent maintenance, 14 percent surveillance, 6 percent other.

Multiple-Cause BOP Trips

Approximately 70 percent of the BOP trips were determined to be single-cause events However, a substantial fraction (27 percent) would not have occurred in the absence of a second condition, and a few trips (3 percent) would not have occurred in the absence of two additional conditions.

Causation of BOP Trips--Systems Implicated

The two largest system contributors to BOP trips were the feedwater system, causing 40 percent of the trips, and the turbine/generator system, contributing about 30 percent. The next largest contributors, the AC power and main steam systems, contributed about 12 percent and 6.5 percent, respectively. Other systems, contributing 3 percent or less to BOP trips over the study period, include air, circulating water, DC power, and instrumentation and control systems.

Causation of BOP Trips--Subsystems Implicated

Feedwater control was the dominant contributing subsystem to feedwater-related BOP trips. Within the turbine/generator system, the dominant contributing subsystem was instrumentation and control, primarily the electro-hydraulic control (EHC) subsystem. Feedwater control and T/G I&C subsystem problems (component failure or human-related) combined caused about 40 percent of the total BOP trips.

Causation of BOP Trips--Components Implicated

The clearly dominant "component" contributor to BOP trips was the human, generally causing about 30 percent of all BOP trips across the major system contributors. The next largest component contributors, generally much less significant than the human, were pumps, valves, electrical switchgear, and circuit cards. For the dominant systems, the data are characterized by a majority of the trips coming from very small contributions from very large numbers of components.

Trends in BOP Trips as a Function of Architect/Engineer

The BOP data base was searched to see if positive or negative performance in terms of BOP trips could be correlated with the architect/engineer (A/E) responsible for designing the BOP. For the major A/E firms that have designed several nuclear units--Bechtel, Stone & Webster, Sargent & Lundy and Ebasco--no clear trends were evident in the data as a function of the A/E firm that designed the BOP.

Trends in BOP Trips as a Function of Plant Age

The data on BOP trips as a function of plant age were widely scattered; even the annual average values at a given age showed a large degree of variability. The overall trend, determined by a linear least squares fit of the annual average data, showed a reduction of about one BOP trip (during the 5 years considered in the study) for every 2 years of increasing age.

Trends in BOP Trips as a Function of Power Level

Approximately half of the BOP trips observed over the study period occurred above 75 percent power, and those trips were dominated by problems in the turbine/generator system. Nearly 30 percent of the observed trips occurred below 25 percent power, and they were dominated by problems in the feedwater system. The remaining trips were distributed evenly between the 25 percent to 50 percent range and the 50 percent to 75 percent range in power level.

BOP Trips vs. Feedwater System Design Characteristics

Because of the predominance of trips initiated by feedwater system problems, an analysis was done to determine if feedwater system design characteristics were associated with differences in BOP trip frequency. Three aspects of feedwater system design were analyzed: the number of feedwater pumps, feedwater supply capacity per pump, and the type of pumps (motor-driven versus turbine-driven).

The data indicates that plants with three feedwater pumps perform only marginally better than plants with two feedwater pumps in terms of both feedwater trips and overall BOP trips. This parameter does not appear to be significant in terms of BOP trip performance. Similarly, plants with excess feedwater capacity (e.g., 100 percent capacity with one pump out of service) performed only marginally better than plants without excess capacity in terms of both feedwater trips and overall BOP trips.

The only significant trend observed during these feedwater system evaluations was that amost all of the best performers have motor-driven feed pumps and that almost all of the worst performers have turbine-driven feed pumps. All of the top nine performers in overall BOP trips (that is, fewest trips per critical year) had motor-driven feed pumps, while five of six of the worst performers had turbine-driven feed pumps. If feedwater-system-induced reactor trips are considered instead of BOP trips, six of the worst seven have turbine-driven feed pumps and eight of the nine best have motor-driven feed pumps. Similarly, if only feedwater-control-induced reactor trips are considered, six of the seven worst performers have turbine-driven pumps, while four of the top seven have motor-driven feed pumps.

Risk Implications of BOP Events

The objective of this task was to evaluate the impact of BOP-related events on the risk, as measured by estimated core melt frequency, of nuclear power plant operation. The task was divided into two parts. First, a quantitative analysis was performed to estimate the risk impact of reactor trips caused by BOP system failures. Second, a qualitative evaluation was performed of the impact of BOP-related events on safety system availability, as reflected by the events having a relatively high risk ranking as reported in the Accident Sequence Precursor program for the years 1984 through 1986.

The results of the delta risk analysis and the evaluation of BOP-related precursor events both show that the reliability of BOP system can have a significant impact on the risk profile of nuclear power plants. For BWRs, in particular, plant core melt frequency appears to be highly sensitive to the frequency of BOP-related transients. The delta risk analysis showed that core melt frequency differed by a factor of 2 to 4 as a function of BOP performance for BWRs. The difference for PWRs was comparatively small, only a factor of 1.1 to 1.3.

For the years 1984 through 1986, 35 precursor events were identified that had estimated conditional probabilities of severe core damage greater than or equal to 1×10^{-4}. Twenty-three of these 35 events (66 percent) had BOP initiators. Thus, the fraction of BOP initiation of the more significant precursor events is approximately the same as the fraction of BOP initiation of reactor trips in general.

Twelve of the 23 precursor events that were considered to be BOP-related and had a high probability of resulting in core damage occurred at BWRs. This is a disproportionate number of such events as BWRs, since approximately two-thirds of all operating U.S. reactors are PWRs. This finding supports the conclusion that BOP-related events are more important, from a risk perspective, at BWRs.

Findings and Recommendations

The major finding of this study was the dramatic reduction in BOP-related trips at commercial nuclear power plants over the 5-year study period from January 1, 1984 through December 31, 1988. This improved performance reduces the urgency of regulatory action to address BOP-related safety concerns. However, regulatory actions can be taken to (1) address the problems of licensees whose BOP trips performance is substantially less favorable than the industry average, and (2) maintain or further improve the performance levels achieved toward the end of the study period.

Findings

1. For the 76 mature nuclear units (OL before January 1, 1983) in the study data base, the average number of BOP trips per unit was reduced from 4.4 per critical year in 1984 to 2.4 per critical year in 1988.

2. On a calendar year basis, for the 76 mature nuclear units in the study data base, the average number of BOP trips per unit was reduced from 2.8 per calendar year in 1984 to 1.6 per calendar year in 1988.

3. Nearly 30 percent of the BOP-related trips resulted from multiple-cause events.

4. Approximately 70 percent of the BOP-related trips resulted from a single event.

5. Considering BOP trips resulting from both single and multiple causes, nearly four out of every five events contributing to BOP trips were either component/equipment failures (47 percent) or human actions (31 percent).

6. NSSS Owners Groups with aggressive trip reduction programs are apparently achieving results in the form of reduced frequencies of BOP-related trips.

7. At the system level, BOP trip causation was dominated by the condensate/feedwater system (40 percent of total trips) and the turbine/generator system (30 percent of total trips).

8. At the subsystem level, BOP trips causation was dominated by the feedwater control subsystem (61 percent of feedwater-related trips; 25 percent of total trips) and the turbine/generator instrumentation and control subsystem (60 percent of turbine/generator related trips; 18 percent of total trips).

9. At the component level, excluding the human "component," BOP trip causation was not dominated by any single component or small group of components.

10. Nearly all the units with the best BOP trip performance (fewest BOP-related trips) have motor-driven feedwater pumps; nearly all the units with the poorest BOP trip performance (highest numbers of BOP trips) have turbine-driven feedwater pumps.

11. From a risk perspective, BOP-related transients contribute significantly more, on a fractional basis, to the estimated core melt frequencies of BWRs than they do to PWRs.

12. BOP-related transients are the initiating events for approximately two-thirds of the more significant accident precursor events.

Recommendations

The dramatic reduction in the number of BOP-related reactor trips at commercial nuclear power plants over the 5-year period ending December 31, 1988 reduces the urgency of regulatory actions directed at BOP performance improvements. However, regulatory actions can and should be taken to (1) maintain the trend toward decreasing numbers of BOP-related reactor trips among NRC licensees, and (2) address the problems of licensees whose performance is substantially less favorable than the industry average.

General Recommendations

1. Communicate to licensees and applicants, in the form of an informational generic letter, the results of recent studies on BOP-related trips and overall scram reduction experience.

2. Identify, monitor and communicate with licensees who are not achieving an acceptably low frequency of BOP-related trip events at their facilities.

3. NRC should work with INPO, the Owners Groups, and EPRI to assist licensees in achieving and maintaining an acceptably low frequency of BOP-related trip events at their nuclear plants.

4. NRC should formally incorporate BOP trip avoidance experience into the Systematic Assessment of Licensee Performance (SALP) process, e.g., as an element in the Safety Assessment/Quality Verification category.

Specific Recommendations

1. Establish a responsibility center within NRC to specifically monitor and evaluate BOP-related reactor trip experience.

2. NRC should expand the role of BOP systems in ongoing NRC activities, specifically in the areas of inspections, maintenance policy, Technical Specifications improvements, human factors and training, severe accident policy/IPEs, the Accident Sequence Precursor program, and advanced reactors/standardization.

3. NRC should expand the evaluation of the risk implications of BOP events to additional PRA studies to test the validity of the risk-related findings made herein.

4. NRC should investigate the implications of the relatively large numbers of multiple-cause events for statistical and risk analyses.

1. INTRODUCTION

This report documents a study of reactor trips related to balance-of-plant (BOP) system failures at commercial U.S. nuclear power plants. The study was performed to support assessment of the safety implications of BOP-related trips and to contribute to identification of ways to achieve and maintain low occurrence frequencies for such trips. The study was performed by Science Applications International Corporation for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Plant Systems Branch.

1.1 Background

For the past several years, the NRC staff has been concerned with the non-safety-related balance-of-plant (BOP) systems and the effects that failures in the BOP systems have on the safety of the plant. For the purposes of this study, the BOP is considered to consist of what is often referred to as the secondary system (all systems associated with the steam power conversion cycle) and supporting systems, such as instrument air and cooling water. The basic concern is the frequency of challenges to plant safety systems that come about as a result of failures in the BOP systems. Because the BOP systems are often designed without any redundancy, there can be any number of single active failures in the BOP systems that can result in a reactor plant trip, usually because of a turbine trip or a loss of main feedwater. Such challenges to the safety systems could be considered a weakness in the defense-in-depth philosophy that has always been the cornerstone of nuclear power plant regulation.

A previous analysis of BOP regulatory issues by Mitre Corporation, (NUREG/CR-4783, Reference 1) found that during 1984 and 1985 BOP-related trips constituted about 70 percent of the total reactor trips. It can be argued that BOP designs that incorporate redundancy are able more often to accommodate plant transients and equipment failures without requiring a reactor trip and a subsequent challenge to safety systems. Similarly, plant maintenance practices and techniques, plant operating characteristics, and even plant aging can increase the challenges to safety systems. A careful study of the operational data and experiences, combined with the use of quantitative risk assessment techniques, was needed to enable the NRC to better understand the sources of challenges to safety systems and to estimate the effect on public risk of BOP-related trips.

1.2 Objectives

The overall objectives of this study were to perform a comprehensive review and evaluation of BOP-related challenges to safety systems, to examine the risk implications of these events, and to make recommendations for resolving BOP-related concerns. The study examined the initiators of BOP challenges, the frequency of these initiators, the degree of design sensitivity or tolerance to these initiators through design features such as redundancy, and the effects on public risk of excessive BOP challenges to safety systems. Specific objectives were to identify generic BOP-related problems, common cause events, similarities and effectiveness of utility/industry programs, and effectiveness of NRC-related activities and to evaluate them with emphasis on developing an overall approach to the resolution of BOP-related concerns.

1.3 Scope of the Study

The initial task was to identify and evaluate available information concerning BOP-related events and activities. Sources of information included (1) BOP-related studies by NRC and NRC contractors, (2) evaluations performed as a result of NRC requirements or requests, (3) generic issues and unresolved safety issues, (4) documentation of operating events (e.g., Licensee Event Report , (5) information from the activities of the Advisory Committee on Reactor Safeguards (ACRS), (6) information generated by the NRC Office for Analysis and Evaluation of Operational Data (AEOD), and (7) efforts performed by utilities and industry groups (e.g., owners groups). These sources of information are discussed in more detail in Appendix L.

Licensee Event Reports (LERs) were obtained for evaluation through the use of the Sequence Coding and Search System (SCSS) data base of LERs maintained for the NRC by Oak Ridge National Laboratory. For the purposes of this study, a reactor trip was defined as an actuation of the reactor protection system, automatic or manual, independent of whether or not actual control rod motion occurred.

1.3.1 Definition of BOP

One of the early tasks in the scope of the study was to define balance-of-plant systems. Definitions of BOP are numerous, and are a function of the context in which the term is used. For the purposes of this study, it was decided to devise an operational (as opposed to theoretical) definition of BOP in terms of the system codes used in the SCSS. The "definition" of BOP used in the study incompasses 47 SCSS codes and related titles provided later in Table 2-1. This resulted in a comprehensive list of BOP systems, including all portions of the power conversion system, AC and DC power, instrumentation, several air and water systems, and others.

1.3.2 Mitre Report

The Mitre report on BOP regulatory issues, mentioned in Section 1.1 above, was used as a point of departure for this study. Four major differences between this study and that reported in the Mitre report are: (1) the definition of BOP used herein included about three times as many BOP systems, not just those associated with power conversions (2) the LER data base evaluated herein covered a period of 5 years, 1984 through 1988, instead of 2 years; (3) this study examined the risk implications of BOP performance; and (4) one task in this study was the preparation of a BOP-specific reactor trip data base, to facilitate the identification of trends and patterns in the population of BOP-related events.

1.3.3 AEOD Report

This study of BOP-related reactor trips was performed in parallel with a portion of a broader-scoped NRC AEOD study that examined progress being made by licensees in reducing the frequencies of reactor trips from all causes. The study performed by AEOD was reported in NUREG-1275, Volume 5 (Reference 2). The BOP study differs from the AEOD study in that it:

- o looks exclusively at BOP-related trip events;
- o includes the preparation of a BOP-related reactor trip data base to identify the relative contributions of component failures

(single and multiple), design adequacies, human errors (operation, maintenance, test), and procedural inadequacies;

- o includes the performance of detailed trend and pattern analyses of the BOP data base on many parameters, including plant, year, age, NSSS vendor, A/E, turbine manufacturer, general cause, system, subsystem, and component implicated; and

- o includes calculations of the estimated incremental risks associated with BOP failures.

1.4 Organization of the Report

The development and use of the BOP data base are described below in Sections 2 and 3, respectively. Insights gained from searching the BOP data base are summarized in Section 4. The results of a brief overview of the risk implications of BOP systems failures are presented in Section 5, including an estimate of the incremental risk associated with favorable versus unfavorable BOP performance, based on selected probabilistic risk assessment studies and on information concerning BOP influence on accident precursor events. Section 6 presents the findings and recommendations of the study. Detailed data are provided in Appendices A-L.

References

1. NUREG/CR-4783, "Analysis of Balance of Plant Regulatory Issues," Mitre Corporation, January 1987.

2. NUREG-1275, Vol. 5, "Operating Experience Feedback Report - Progress in Scram Reduction, Commercial Power Reactors," USNRC, Office for Analysis and Evaluation of Operational Data, March 1989.

2. BOP DATA BASE DEVELOPMENT

A data base of BOP-related reactor trips was created as part of this study. BOP trip data were drawn from the Licensee Event Report (LER) data base maintained by Oak Ridge National Laboratory (ORNL). The Sequence Coding and Search System (SCSS) for the LER data base was used to identify potentially relevant LERs. Table 2-1 lists the 47 SCSS codes, and the corresponding BOP systems, used in the LER search. The LER search on the 47 SCSS codes covered the 5-year period from January 1, 1984, through December 31, 1988. Approximately 2030 trips involving BOP systems were identified.

The information collected from the LER search was analyzed to determine whether the reactor trip was directly related to a failure of a BOP component or function. If so, the trip information was incorporated into the BOP data base.

Of the 2030 LERs reviewed, 1405 BOP-related events were considered appropriate for entry into the BOP data base. LERs were not included in the BOP data base if any of the following conditions applied:

- o BOP involvement was incidental to the trip, i.e., not in the causation sequence.

- o The trip occurred during special tests or evolutions during shutdown conditions and would not have occurred when the reactor was critical or at power. Accidents occurring at shutdown conditions that could conceivably have occurred at power, or with substantial decay heat in the core were included in the BOP data base.

- o The trip resulted from loss of offsite power or other events external to the plant systems.

The BOP data base was developed based on PC-dBase III Plus software. The various data elements or "fields" are presented in Figure 2-1, which also shows the format used for entering applicable data into the data base.

Table 2-1
47 Sequence Coding and Search System (SCSS) codes

BP	Main Steam Pressure Relief
BA	Auxiliary Feedwater
CA	Component Cooling Water
CB	Essential Water
CC	Essential Air
EA	AC >35kv (exclude events with loss of offsite power)
EB	600v <AC <35kv
EC	AC <600v
ED	Vital AC
EE	DC
FA	Main Steam
FB	Turbogenerator
FC	Turbogenerator Turbine Steam Sealing
FD	Main Condenser
FE	Noncondensable Gases Extraction
FF	Turbine Bypass
FH	Steam Extraction
FI	Condensate and Feedwater
FK	Moisture Separators/Reheaters
FP	Condensate Demineralizer
FR	Circulating Water
FT	Seal Water
HL	Turbine Bldg. HVAC
HR	Pumping Stations HVAC
HS	Misc. Structures HVAC
HT	Chilled Water System
IB	Computer
IF	Fire Detection
II	Turbogenerator I&C
IT	Feedwater Control
IZ	Nonnuclear Instrumentation
KC	Control and Service Air
KD	Demineralized Water
KF	Fire Protection
KT	Raw Cooling Water
KW	Raw Service Water
SL	Turbine Bldg.
SP	Pumping Stations
SR	Cooling Towers
ST	Switchyard
SW	Miscellaneous/Unknown Structures
WI	Plant Drainage
WK	Equipment Drainage
WL	Roof Drainage
ZX	Other
ZY	Unknown
ZZ	Multiple Known

BOP Data Base Format

Plant: Plant-Name
Form: 1-4 digit identifying number

Event ID: LER ID number   **Power Level:** 0-100%
Event Date: MM/DD/YY   **Trip Type:** Automatic/Manual

BOP System: System name (up to 30 characters)
BOP Subsystem: Subsystem name (up to 30 characters)
BOP Component: Component type (up to 40 characters)

Cause 1: Root causes of event: Component
Cause 2: Failures, human errors, etc.
Cause 3: (up to 40 characters each)

Impact 1: Events, other than plant trips, resulting from BOP
Impact 2: event, e.g., safety system failures (up to 40 characters each)
Impact 3:

Event Description: Text description of event

Figure 2-1. BOP Data Base Format

The "Form" entry in Figure 2-1 is an LER-specific identification number for locating the LER from which the data was taken. The event record identifies the BOP system (e.g., feedwater), subsystem (e.g., feedwater control), and component (e.g., feedwater control valve) as applicable. Up to three potential causes of the event may be specified. A narrative event description is also provided. Appendix A contains a sample of 30 entries from the BOP data base.

Supplementary data bases were also found to be necessary for conducting analyses of trends and patterns. The supplementary data bases contain plant data and critical hours (number of hours the reactor was critical) data. The supplementary plant data base includes the following data elements:

- Operating license (OL) date
- Nuclear Steam Supply System (NSSS) vendor
- Architect/engineer
- Turbine-generator manufacturer.

The critical hours supplementary data base includes:

- Critical hours per year for each plant for the years 1984 through 1988
- Total critical years accumulated during the period 1984 through 1988.

Printouts of these supplementary data bases are included in Appendices J and K.

3. TREND AND PATTERN EVALUATIONS USING THE BOP DATA BASE

Many searches were performed on the BOP data base to look for trends and patterns in the data. The searches were performed either by automatically querying the data base with structured dBase III program codes or by manually searching the data with embedded dBase III commands. Table 3-1 lists the initial searches performed on the BOP data base, some of which also required use of the supplementary plant data base (e.g., those involving NSSS vendor, architect/engineer, turbine-generator manufacturer). Searches addressing BOP trips per critical year per plant required use of the supplementary data base containing the critical hours data. Additional searches were performed as questions arose on the results of the initial searches.

A basic distinction was made between mature plants and new plants. Mature plants were defined as those receiving operating licenses before January 1, 1983. Thus, all plants in the mature plant category had held an operating license (OL) at least 1 year before the start of the LER period covered by the study -- January 1, 1984. This definition of mature plants resulted in a constant population of 76 plants for trend and pattern analysis. A "floating" definition of mature plants was considered but not used because it would have introduced another variable (plant population) into the trend and pattern analysis.

3.1 BOP Trips per Calendar Year

Table 3-2 presents the average number of BOP trips per calendar year (raw data) for the years 1984 through 1988, grouped by NSSS vendor. Mature units are distinguished from new units. The individual plant data used to compile the averages are given in Appendix B.

The data for the mature Westinghouse units show a clear downward trend, with the 1987 and 1988 values approximately half the 1984 value. This probably reflects the work of the Westinghouse Owners Group in reducing trip frequencies. Trends in the data for mature Babcock and Wilcox (B&W) units are not as clear, but the average number of trips was reduced by a factor of 2 between 1985 and 1986, and the lower value was sustained in 1987 and 1988. The average BOP trip frequencies for mature Combustion Engineering (CE)

Table 3-1
BOP Data Base Search Logic

1. BOP trips by plant
2. BOP trips by plant by year
3. BOP trips by NSSS vendor
4. BOP trips by architect/engineer
5. BOP trips by NSSS and architect/engineer combinations
6. BOP trips by operating license date by plant
7. BOP trips by turbine-generator manufacturer by plant
8. BOP trips by cause by year
9. BOP trips by system/subsystem combinations
10. BOP trips by subsystem/component combinations
11. BOP trips by power level

Table 3-2
Average BOP Trips per Unit per Calendar Year (1984 through 1988)

Mature units (OL Before Jan 83)	1984	1985	1986	1987	1988
B&W (8 units)	2.43	4.13	1.86	1.86	1.62
CE (9 units)	2.33	2.78	2.67	3.33	1.33
GE (26 units)	2.35	2.50	2.04	2.26	1.46
W (33 units)	3.36	3.00	2.97	1.52	1.76
All vendors	2.76	2.92	2.47	2.00	1.59

New units (OL After Jan 83)	1984	1985	1986	1987	1988
B&W (none)	-	-	-	-	-
CE (2 to 6 units)	5.00	7.75	4.00	3.20	1.00
GE (3 to 11 units)	10.67	2.83	4.63	3.80	3.00
W (2 to 15 units)	10.00	10.20	6.00	7.08	3.53
All vendors	8.86	6.81	5.00	5.21	2.87

units increased by 50 percent from 1984 to 1987 but dropped significantly in 1988. The data for mature General Electric (GE) units do not indicate a trend up or down until 1988, when the average trip rate dropped below two per year, comparable to the rates for units in the other three vendor groups.

A comparison was made between total annual BOP trip frequency as identified in this study and the frequency as identified in the previous study of BOP-related regulatory issues performed by Mitre Corporation (Reference 1). The comparison was made for the calendar years common to the two studies, 1984 and 1985. The results are shown below.

	Total BOP Trips		
	1984	1985	2-yr total
Mitre study	148	145	293
Present study	179	251	430

The reason for these differences is in the definition of BOP for the two studies. The BOP definition used in the Mitre study was limited to the power conversion systems (14 SCSS codes), whereas the present study included the power conversion systems plus many other systems--electrical, instrumentation and control, cooling water, air systems, etc. (47 SCSS codes).

3.2 BOP Trips per Critical Year--Annual

A more meaningful indication of the frequency of BOP trips of interest is the compilation of BOP trips per critical year, where the raw data per calendar year are normalized to the time the unit was critical. (Note that this normalization parameter is not entirely consistent, because some entries in the BOP data base represent conditions when the reactor was subcritical).

Table 3-3 presents the average number of BOP trips per critical year for the years 1984 through 1988, grouped by NSSS vendor. Mature units are once again distinguished from new units. The individual plant data used to compile the averages are given in Appendix C.

Table 3-3
Average BOP Trips per Unit per Critical Year 1984 through 1988

Mature units (OL Before Jan 83)	1984	1985	1986	1987	1988
B&W (8 units)	3.09	7.14	3.10	2.34	2.23
CE (9 units)	3.36	3.88	3.45	4.51	1.71
GE (26 units)	4.15	4.07	3.46	3.68	2.52
W (33 units)	5.11	3.89	4.18	2.18	2.42
All vendors	4.36	4.23	3.77	2.97	2.33

New units (OL After Jan 83)	1984	1985	1986	1987	1988
B&W (none)	-	-	-	-	-
CE (2 to 6 units)	7.46	16.4	6.04	4.13	1.28
GE (3 to 11 units)	68.10	7.02	9.69	6.99	4.04
W (2 to 15 units)	27.40	23.50	9.24	12.20	5.02
All vendors	24.40	15.80	8.53	8.66	3.93

The trends in the critical year data are generally the same as observed in the raw (calendar year) data, i.e.:

- o Westinghouse units show a clear downward trend from 1984 through 1987 with a slight increase in 1988.

- o B&W units show a downward trend after 1985.

- o CE units show an upward trend, increasing by about 50 percent from 1984 through 1987, but decreasing substantially in 1988.

- o GE units show a general downward trend, with a significant decrease in 1988.

- o All four vendors groups show a significant improvement in BOP trip performance in 1988 versus 1984.

Table 3-4 lists the 10 "best and worst" BOP performers for the 5-year period. This information shows the range of plant performance and the distribution of "good" and "poor" performances among the NSSS vendors.

3.3 BOP Trips per Critical Year - Cumulative Average

Data on the cumulative average number of BOP trips per critical year, for the years 1984 through 1988, are given in Table 3-5. The individual plant data from which the averages were calculated are given in Appendix D. These data show remarkable consistency among the mature units of the different NSSS vendors, at slightly less than four BOP trips per critical year for 1984 through 1988, with a spread (highest to lowest) of only 18 percent. These data indicate that the conditions or parameters that cause variations in BOP trip frequency do not strongly reflect NSSS vendor, a result that is not surprising, although the degree of uniformity is somewhat surprising.

3.4 BOP Trips by General Cause

General causes of BOP trips defined for the purposes of this study were component failure, human-related, procedure-related, design-related, and spurious or unknown. Table 3-6 presents the breakdown of general causes of

Table 3-4
Range of BOP Performance
(BOP Trips, 1984-1988)

Top 10 BOP Performers

Units	NSSS	Number of BOP Trips	Average BOP Trips per Critical Year
Prairie Island 2	W	1	0.2
Fort Calhoun	CE	1	0.3
Point Beach 2	W	3	0.7
Point Beach 1	W	3	0.8
Prairie Island 1	W	5	1.1
San Onofre 1	W	3	1.3
Duane Arnold	GE	5	1.4
North Anna 2	W	7	1.6
Farley 2	W	7	1.6
Quad Cities 1	GE	7	1.8

Bottom 10 BOP Performers

Units	NSSS	Number of BOP Trips	Average BOP Trips per Critical Year
Salem 2	W	34	10.2
Grand Gulf 1	GE	27	9.4
Dresden 3	GE	22	7.2
Indian Point 3	W	26	7.1
Rancho Seco	B&W	11	7.0
Maine Yankee	CE	25	6.4
Davis Besse	B&W	13	6.3
Indian Point 2	W	23	6.3
D.C. Cook 2	W	18	6.1
Diablo Canyon 1	W	18	6.0

Table 3-5
BOP Trips per Critical Year per Unit, 5-year Cumulative Average

Mature Units

NSSS Vendor	5-Year Cumulative Average	Standard Deviation
B&W (8 units)	3.98	1.75
CE (9 units)	3.40	1.64
GE (26 units)	4.02	2.44
W (33 units)	3.76	2.12

New Units

NSSS Vendor	5-Year Cumulative Average	Standard Deviation
B&W (0 units)	-	-
CE (5 units)	6.31	2.69
GE (11 units)	10.67	7.31
W (15 units)	14.02	12.29

Table 3-6
BOP Trips by Cause
(All units 1984-1988)

Cause	Percent
Component failure	46.5
Human-related	30.9
Design-related	5.6
Procedure-related	5.1
Spurious or unknown	4.8
Environment	1.5
Other causes	5.6
	100.0

BOP trips in these categories. The percent column is percent of causes, not trips, to account for multiple-cause events. The percent of trips is not easily extractable from the data (because of multiple cause events), but is not expected to differ markedly from the percent of causes listed. The breakdown by general cause--47 percent component failure, 31 percent human related, 22 percent all other cause categories--is not surprising.

By comparison, the Mitre report (Reference 1, p. xix) estimated that about half the BOP trips are caused by single component failures in the power conversion systems and about half are caused by personnel errors. As discussed in Section 3.5, this study evaluated multiple-cause events, and thus disagrees with the Mitre conclusion that about half the BOP trips are caused by single component failures. Our estimate is about one-third are caused by single component failures. Similarly, our evaluation indicates that about a third, rather than half, of the BOP trips are human-related. This does not include design- and procedure-related problems as human-related.

A comparison with the AEOD report on scram reduction (Reference 2, Vol. 5) is less pertinent because the AEOD data are for all trips (not just BOP trips) and for mature plants (not all plants). Normalized data from the AEOD report (Reference 2, Table 3-11, p. 24) indicate, for the time period 1984 through 1987, that about 60 percent of the trips were caused by equipment failure and about 25 percent by human error.

3.5 BOP Trips by Single or Multiple Causes

Table 3-7 presents the results of an evaluation of all single- and multiple-cause BOP trips. Although most of the trips (70 percent) can be traced back to a single BOP cause, a significant fraction (27 percent) resulted from two causes, and a small fraction (3 percent) from three causes. There is a subjective element to these categorizations, but an attempt was made to distinguish those BOP trips which probably would not have occurred in the absence of a second (or third) causative mechanism.

Table 3-7
Single- and Multiple-Cause BOP Trips
(All units 1984-1986)

<u>Single cause</u>

	No. of trips
Component failure	487
Human-related	333
Procedure-related	46
Design-related	34
Environment	4
Spurious or unknown	74
Other	9
Total single cause	987 (70%)

<u>Double cause</u>

379 (27%)

<u>Triple cause</u>

39 (3%)

Total BOP trips 1405

3.6 BOP Trips by System and Subsystem

The breakdown of BOP trips by system and subsystem is presented in Table 3-8. The feedwater system was implicated in about 40 percent of the total BOP trips, and the feedwater control subsystem was involved in 61 percent of the feedwater-related trips. The turbine-generator (T/G) system accounted for about 30 percent of the total trips; most of the turbine-generator-related trips, about 60 percent, involved the T/G instrumentation and control subsystem. The next largest system contributors to BOP trips were the AC power systems, about 12 percent; the main steam system, about 6.5 percent; and air systems, about 3 percent. Clearly the dominant contributors to BOP trips were the feedwater control and the T/G instrumentation and control subsystems, causing about 42 percent of the total BOP trips. The detailed information on BOP trips by system and subsystem is presented in Appendix E.

3.7 BOP Trips by System and Component

The breakdown of BOP trips by system and component, shown in summary form in Table 3-9, indicated that human error clearly dominated as the source of the failures. The human error contribution was about one-third of the total for each of the major system contributors to BOP trips--feedwater, turbine/generator, AC power, and main steam. In each case, the next largest contributor was much smaller than the human error contribution, indicating that a very large number of individual components was involved, each contributing a very small fraction to the system failure rates. The detailed information on BOP trips by system and component is presented in Appendix F.

3.8 BOP Trip Frequency and Feedwater System Design Characteristics

Because of the predominance of trips initiated by feedwater system problems, an analysis was done to determine if feedwater system design characteristics were associated with differences in BOP trip frequency. Three aspects of feedwater system design were analyzed: the number of feedwater pumps, feedwater supply capacity per pump, and the type of pumps (motor-driven versus turbine driven).

Table 3-8
BOP Trips by System and Subsystem
(All units 1984-1988)

System	Number of Trips	Subsystem	Number of Trips	Percent
Feedwater	561			39.9
		Feedwater control	344	
		Unspecified	135	
		Condensate	26	
		Feedwater heater	23	
		Others	33	
Turbine-generator	419			29.8
		T/G I&C	250	
		Unspecified	87	
		Condenser	33	
		Generator	9	
		Lube oil	8	
		Others	32	
AC power	168			12.0
		High voltage	77	
		Vital AC (120V)	47	
		Medium voltage AC	31	
		Others	13	
Main steam	90			6.4
		Unspecified	47	
		Moisture separator reheater	20	
		Others	23	
Air systems	44			3.1
I&C (general)	31			2.2

Table 3-9
BOP Trips by System and Component
(All units 1984-1988)

System	Number of Trips	Component	Number of Trips	Percent Human-Related
Feedwater	561			
		Human	213	38.0
		FW regulating valve	38	
		Circuit card	29	
		Pump	31	
		Valves	23	
		Unknown	21	
Turbine-generator	419			
		Human	128	30.5
		Circuit card	17	
		Unknown	25	
AC power	168			
		Human	51	30.4
		Transformer	23	
		Circuit breaker	10	
Main steam	90			
		Human	34	37.8
		Valve	6	

The base population for this analysis was a set of 60 plants represented in the BOP data base, for which data on feedwater system characteristics were available. Data on all three analysis variables were not available for all 60 plants. Thus the specific analysis results described below address somewhat smaller subpopulations that differ slightly in membership.

Comparison of various BOP trip rates per critical year for plants with two feedwater pumps and plants with three feedwater pumps revealed no clear advantage for either two- or three-pump plants. For the population of 60 plants, 15 use three pumps and 45 use two pumps. Although the three-pump plants consistently performed better than the two-pump plants, the difference was not large. The results of the comparisons made are summarized below.

Avg. Number of BOP-Related Trips per Critical Year

	Total	FW systems	FW control system
2-pump FW plants	4.5	1.9	1.2
3-pump FW plants	4.2	1.6	0.9

The number of feedwater pumps does not convey the excess pumping capacity for feedwater. Two pumps each with 50 percent capacity and three pumps with 33.3 percent capacity have the same excess pumping capacity, namely zero.

To learn the effect that excess feedwater pumping capacity might have on BOP trips, data on 51 mature plants were examined. Each plant was rated according to what percentage of full feedwater flow could be delivered with one pump out of service ("N-1 capacity"). For example, a plant that has two 50 percent pumps can supply only 50 percent if one pump is lost; a plant with three 50 percent pumps can supply 100 percent. The intent here was to determine if plants with large excess feedwater pumping capability had fewer BOP trips. There were eight plants with N-1 capacity of 100 percent and 13 plants with N-1 capacity of ≥78 percent. The plants with N-1 capacity of 78 percent or higher experienced only very slightly improved statistics; even the plants with N-1 capacity of 100 percent were only 15 percent better (fewer trips per critical year) than the average of all of the rest.

Clearly, excess feedwater capacity was not a major factor in creating good performers. A summary of these data is presented below. The complete data can be found in Appendix G.

	Avg. Number of BOP-Related Trips per Critical Year		
	Total BOP	FW system	FW control system
N-1 capacity = 100%	3.8	1.4	1.2
N-1 capacity ≥ 78%	4.2	1.8	1.4
N-1 capacity < 78%	4.4	1.9	1.2

The fact that many other factors besides the capacity or number of feedwater pumps enter into BOP and feedwater trip performance can be seen in the fact that some of the worst performers have high excess feedwater capacity and that most of the high capacity feedwater plants are not in the best performer group. In fact 3 of the top 10 performers have no excess feedwater capacity.

Finally, one trend observed during these evaluations is that most of the best performers have motor-driven feed pumps and that almost all of the worst performers have turbine-driven feed pumps. All of the top nine performers in overall BOP trips (that is, fewest trips per critical year) had motor-driven feed pumps, while five of six of the worst performers had turbine driven feed pumps. If feedwater-system-induced reactor trips are considered instead of BOP trips, six of the worst seven have turbine-driven feed pumps and eight of nine of the best have motor-driven feed pumps. Similarly, if only feedwater-control-induced reactor trips are considered, six of the seven worst performers have turbine-driven pumps, while four of the top seven have motor-driven. Summarized below is a comparison of trips at 57 plants classified by type of feedwater pumps.

	Avg. Number of BOP-Related Trips per Critical Year		
	Total BOP	FW system	FW control system
Motor driven FW plants	3.2	1.1	0.8
Turbine driven FW plants	5.5	2.3	1.5

3.9 BOP Trips by Plant, NSSS Vendor and A/E

Because BOP systems are the subject of this study, it is possible that the failure frequencies would show some trends or patterns as a function of the architect/engineer. The results of our searches of the BOP data base indicate that this is not the case; i.e., there are no clear patterns observed among the major A/E firms who have engineered several units.

Table 3-10 presents data on A/E firms, number of plants and number of trips, grouped by NSSS vendor. The average number of BOP trips per plant was derived from Table 3-10. The results ranged from 9 to 14 trips per plant over the 5 years of data for the major A/E firms--Bechtel, Stone & Webster, Sargent & Lundy, and Ebasco. BOP trip data for individual plants, with NSSS vendor and A/E firms identified, are presented in Appendix H.

3.10 BOP Trips by Plant Age

The BOP data base was searched for information on the age-dependence of BOP trip frequencies. The resulting data are presented in Figure 3-1 for mature plants, i.e., those receiving an operating license before January 1, 1983. The age of a unit was defined as 1986 (the middle of the study period) minus the year of the unit operating license. Each data point represents the average total number of BOP trips for units of the same age over the 5-year time period.

As can be seen from Figure 3-1, the data are characterized by a wide scatter; the average values for plants of different age show large spikes (both up and down). A linear least squares fit of the average data provided a downward slope of about half a trip per year of plant operation,

Table 3-10
BOP Trips by NSSS Vendor and A/E Firm
(All units 1984-1988)

NSSS Vendor	A/E Firm	No. of Trips	No. of Plants
B&W	Bechtel	38	3
B&W	Duke and Bechtel	27	3
B&W	Gilbert	24	2
B&W	Total:	89	8
CE	Bechtel	110	9
CE	Ebasco	59	3
CE	Gibbs and Hill	1	1
CE	Stone & Webster	25	1
CE	Total:	195	14
GE	Bechtel	113	11
GE	Burns & Roe	54	3
GE	Detroit Edison and S&L	20	1
GE	Ebasco	17	2
GE	Gilbert	13	1
GE	Niagara Mohawk Power Corp.	10	1
GE	Sargent & Lundy	92	7
GE	Southern Company and Bechtel	33	2
GE	Stone & Webster	51	4
GE	TVA	8	3
GE	United Engineers	19	2
GE	Total:	430	37
W	American Electric Power	28	2
W	Bechtel	108	9
W	Bechtel and Sargent & Lundy	24	1
W	Duke Power Company	77	4
W	Duquesne Light/Stone & Webster	26	2
W	Ebasco	36	2
W	Fluor Pioneer	22	3
W	Gilbert	30	2
W	Pacific Gas & Electric	39	2
W	Public Service Electric & Gas	50	2
W	Sargent & Lundy	77	6
W	Southern Company and Bechtel	17	2
W	Stone & Webster	91	7
W	TVA	17	2
W	United Engineers	49	2
W	Total:	691	48
Total No. of Trips:		1405	

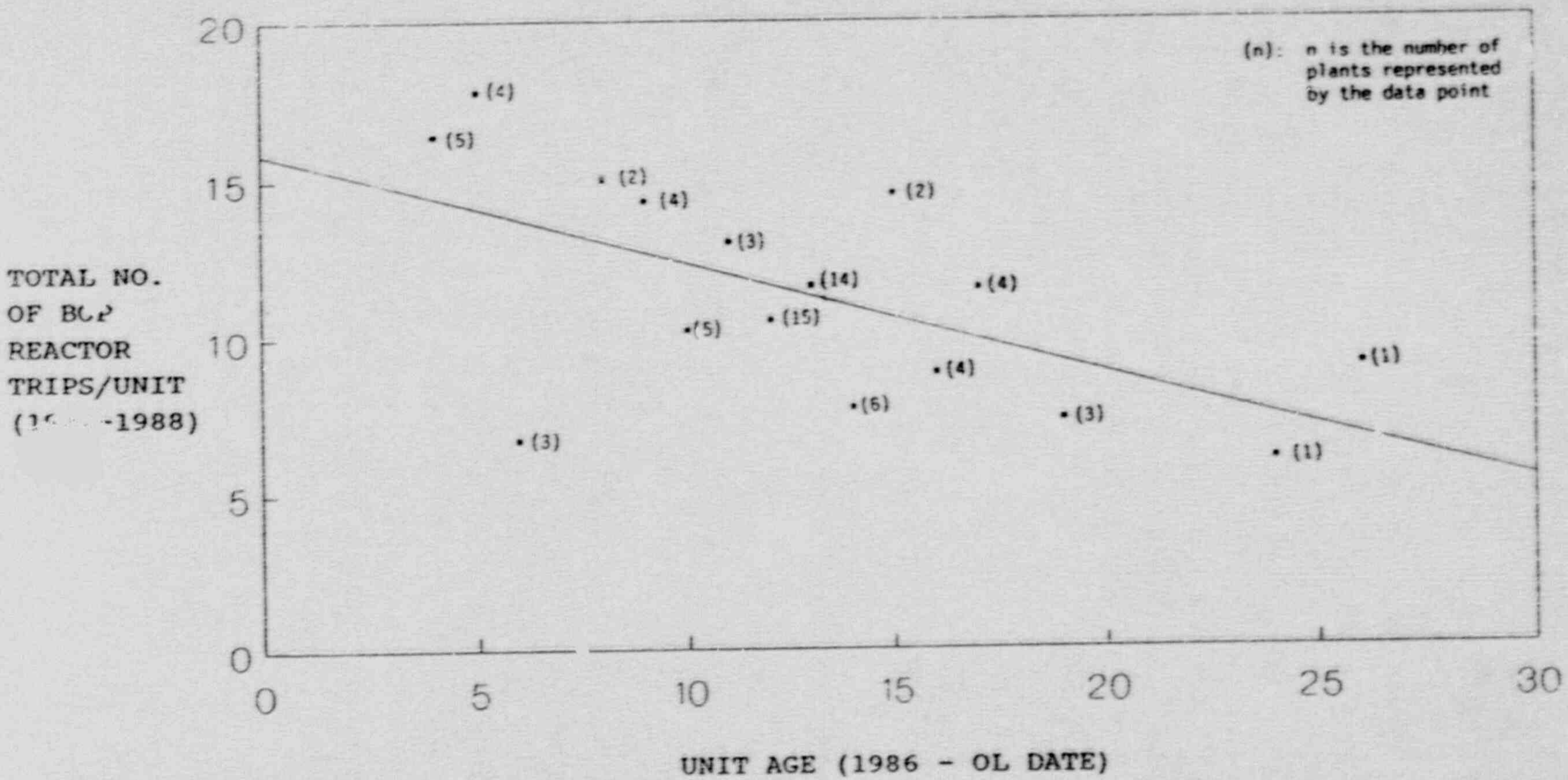

Figure 3-1. Age-Dependence of BOP-Related Trips

suggesting that BOP-related trips tend to decrease over the operating life of a plant.

3.11 BOP Trips as a Function of Power Level

A summary of the BOP trips as a function of power level at which the trips occurred is given in Table 3-11. Half the BOP trips occurred above 75 percent power, and these were dominated by turbine-generator problems. Because most plants spend most of their time above 75 percent, this is not a surprising result. In fact, a higher fraction might have been expected at high power. Nearly 30 percent of the BOP trips occurred at or below 25 percent power, and these were dominated by feedwater problems. The relatively high percentage of BOP trips at reduced power levels could be an indication of the difficulty of operating a nuclear power plant at reduced power levels. The remaining trips were divided evenly between the 25 percent to 50 percent range and the 50 percent to 75 percent range. The detailed data by plant, from which Table 3-11 was derived, are given in Appendix I.

3.12 Origin of the Reduction in BOP Trips

Several special-purpose searches of the BOP data base were performed to identify the origin of the dramatic reduction in the number of BOP trips between 1984 and 1988.

Table 3-12 presents data for BOP trips by general cause by year. The top number in each set is the value for mature units only, i.e., the 76 units that received OLs before January 1, 1983. The bottom number in each set is the value for all units, which varied in number from 86 in 1984 to 108 in 1988. Note that the data for 1984 does not generally fit the trend, and the largest reductions are usually between the 1987 and 1988 data. Overall, both the component failure and human-related causes (by far the two largest contributors) showed substantial reductions over the 5-year study period.

Table 3-13 shows feedwater trips by year by reactor vendor, along with normalized per-unit values, to account for the varying number of units over the time period. The total number of feedwater trips was reduced 20 percent from 1984 through 1988, in spite of a 25 percent increase in the number of

Table 3-11
BOP Trips by Power Level
(All units 1984-1988)

Power Level Range, Percent	Number of BOP Trips	Percent of Total BOP Trips
0-25	398	28.3
(0-5)	(137)	(9.8)
(5-25)	(261)	(18.6)
25-50	148	10.5
50-75	148	10.5
75-100	711	50.6
Total BOP Trips	1405	100%

Table 3-12
BOP Trips by General Cause by Year
(1984-1988)

Cause	Number of Causes 1984	1985	1986	1987	1988	Total
Component failure	138*	150	121	96	82	587
	176	209	173	172	135	865
Human-related	76	99	78	62	37	352
	101	142	129	130	73	575
Design-related	12	16	13	14	6	61
	19	27	22	25	12	105
Procedure-related	11	12	9	8	15	55
	14	21	20	19	21	95
Environment	8	3	5	4	0	20
	8	6	6	5	3	28
Unknown/spurious	21	16	15	4	8	64
	23	22	21	13	11	90
Other	27	19	18	17	3	84
	31	24	20	23	6	104
Totals	293	315	259	205	151	1223
	372	451	391	387	261	1862

*Top Value: Mature units only (OL before January 1, 1983)
Bottom Value: All units

Table 3-13
Feedwater Trips by Year by Vendor
(All units, 1984-1988)

NSSS Vendor	Feedwater Trips 1984	1985	1986	1987	1988
B&W	5 (0.62)*	15 (1.87)	5 (0.62)	7 (0.88)	7 (0.88)
CE	17 (1.50)	29 (2.20)	12 (0.86)	19 (1.30)	11 (0.73)
GE	25 (0.86)	24 (0.83)	28 (0.93)	41 (1.28)	18 (0.60)
W	56 (1.60)	69 (1.77)	69 (1.77)	60 (1.36)	46 (0.96)
Total	103	137	114	127	82

* (xxx) - average trip/unit

units. There were 45 fewer feedwater trips in 1988 than there were in 1987 (35 percent reduction), and half of this reduction came from GE BWRs. The number of feedwater trips per unit year decreased substantially between 1984 and 1988 for CE, W, and GE units, but increased substantially for B&W units.

Table 3-14 presents data on turbine trips by year by reactor vendor, along with normalized per-unit values, to account for the varying number of units over the time period. The total number of turbine trips was reduced 25 percent from 1984 through 1988, in spite of a 25 percent increase in the number of units. A large reduction of 30 trips (30 percent) occurred between 1987 and 1988, and more than half of this reduction is from W units. The per-unit values show substantial reductions for all reactor vendors except B&W, which stayed the same between 1984 and 1988.

Table 3-15 presents a breakdown of human-related causes by year. Focusing on the two major contributors, operations and maintenance, once again the 1984 data does not fit the trend. There were substantial increases in the two areas between 1984 and 1985, and an even larger decrease in the maintenance-related causation between 1987 and 1988. Overall, there was a 45 percent reduction in human-related BOP trips causation between 1987 and 1988, with about half the reduction coming from the maintenance area. Between 1984 and 1988, a 30 percent reduction in human causation was achieved, in spite of a 25 percent increase in the number of units

In summary, no single factor can be identified as the major reason for the substantial reduction in BOP-related trips between 1984 and 1988 or, in many cases, between 1987 and 1988. In terms of general causation, fewer component failures and fewer human errors both contributed to the reduction in BOP trips. At the systems level, both feedwater and turbine/generator related trips decreased substantially, especially between 1987 and 1988.

Table 3-14
Turbine Trips by Year by Vendor (All units)

NSSS Vendor	Feedwater Trips 1984	1985	1986	1987	1988
B&W	3 (0.37)*	8 (1.00)	3 (0.37)	0 (0.00)	3 (0.37)
CE	6 (0.55)	15 (1.15)	15 (1.07)	16 (1.06)	4 (0.27)
GE	46 (1.60)	29 (1.00)	31 (1.03)	29 (0.91)	26 (0.87)
W	32 (0.91)	32 (0.82)	42 (1.08)	49 (1.10)	31 (0.64)
Total	87	84	91	94	64

* (xxx) - average trip/unit

Table 3-15
Type of Human-Related Cause by Year
(All units, 1984-1988)

Types of Human-Related Cause	1984	1985	1986	1987	1988	Total	Percent
Operations	44	61	53	39	33	230	40.0
Surveillance	15	12	20	22	10	79	13.7
Maintenance	35	63	48	57	26	229	39.8
Others	7	6	8	12	4	37	6.4
Total	101	142	129	150	73	575	100.0

References

1. NUREG/CR-4783, "Analysis of Balance of Plant Regulatory Issues," Mitre Corporation, January 1987.

2. NUREG-1275, "Operational Experience Feedback Report - Progress in Scram Reduction," Vol. 5, L C. Bell and P.D. O'Reilly, USNRC, Office for Analysis and Evaluation of Operational Data, March 1989.

4. INSIGHTS INTO BOP EVENTS

This section summarizes the insights gained from the search for trends and patterns in the 5-year, 1405-event BOP data base. General observations are followed by findings on BOP trips per calendar year; BOP trips per critical year (annual and 5-year average values); general causation of BOP trips (i.e., component failure, human-related, design-related, etc.); multiple cause BOP trips; systems, subsystems, and components implicated in BOP trip causation; and trend observations by architect/engineer, plant age, and plant power level. For the purposes of these evaluations, mature plants (as used for the calendar year and critical year data) were defined as those receiving operating licenses before January 1, 1983; all later plants were defined as new plants.

4.1 General Observations

- Data on the annual average number of BOP trips grouped by NSSS vendor indicate that the owners groups with aggressive trip reduction programs are achieving results in the form of reduced frequencies of BOP trips.

- Data on BOP trip causation by system and component (Appendix F) indicate that, for the major system contributors (the feedwater and turbine-generator systems), a majority of the trips are caused by very small contributions from a very large number of components. This implies that to achieve further improvements, component reliability improvement programs must be very broad-based, and not focused on a few major contributors.

- Data on the general causes of BOP trips indicate that programs directed toward achieving further reductions in BOP trip frequencies will need to contain both a technical element (component, system or functional reliability improvement) and a human performance element (fewer human errors in operation, maintenance, surveillance, and testing).

4.2 Specific Trends and Patterns

4.2.1 BOP Trips per Calendar Year

- Mature Westinghouse units on average showed a downward trend, decreasing by more than a factor of 2 between 1984 and 1987, with a slight increase in 1988, to 1.76 BOP trips per calendar year.

- Mature GE units showed no clear trend between 1984 and 1987, but achieved a 35 percent reduction between 1987 and 1988, to 1.46 BOP trips per calendar year.

- Mature CE units showed an increasing trend of about 40 percent between 1984 and 1987, but a decrease by more than a factor of 2 between 1987 and 1988, to 1.33 BOP trips per calendar year.

- Mature B&W units showed a substantial downward trend after 1985, with more than a 60 percent reduction by 1988, to 1.62 BOP trips per calendar year. (Note: There are only eight B&W units, and the unit with the least favorable BOP trip history - Rancho Seco - did not operate between late 1985 and early 1988.)

- Overall, the mature nuclear plants showed a substantial reduction in BOP trips over the 5-year period, from an average of 2.8 BOP trips per calendar year in 1984, to 1.6 BOP trips per calendar year in 1988.

The 1988 average BOP trip frequency of 1.6 trips per unit per calendar year corresponds to a total unplanned trip frequency of approximately 2.4 trips per unit per calendar year. Data from Reference 1, for the years 1980 through 1984, indicate that this level of performance for U.S. nuclear plants is approaching that for Japanese and German reactors, the world's best in terms of minimizing unplanned reactor trips. Although the data are not directly comparable, some indication of comparative performance can be drawn from the following values:

Japan

Trips only while critical

PWRs > 1000 MWe, 1980 through 1984

0.00 to 1.50 unplanned trips per unit-year

BWRs > 1000 MWe, 1980 through 1984

0.33 to 5.00 unplanned trips per unit-year

Germany

Trips with turbine on line

PWRs, 1981 through 1984

0.45 to 1.40 unplanned trips per unit-year

BWRs, 1981 through 1984

0.99 to 2.80 unplanned trips per unit-year

The concept of what constitutes an acceptably low frequency of unplanned reactor trips also needs to be addressed. Then-ONRR Director Harold Denton, speaking at an NEA Symposium in Tokyo in April 1986, recommended a goal of achieving a trip frequency (during power operation) of no more than 2 trips per unit per year by 1990 (Reference 2). Similarly, the Institute of Nuclear Power Operations has established a 1990 goal of 1.5 unplanned automatic trips per unit per year while critical, for units with a capacity factor of 25 percent or greater (Reference 3).

4.2.2 BOP Trips per Critical Year, Annual Data

Trips per critical year is a more meaningful parameter than trips per calendar year because it reflects the fraction of time that the reactor was being operated. The trends in trips per critical year generally follow the trends in trips per calendar year, although some trends are magnified by the data on critical hours per calendar year.

- Mature Westinghouse units showed a decrease of approximately 60 percent between 1984 and 1987, followed by a 10 percent increase in 1988, to 2.42 BOP trips per critical year.

- o Mature GE units showed a clearer downward trend on a critical year basis than was evident in the calendar year data and decreased by approximately 40 percent between 1984 and 1988, to 2.52 BOP trips per critical year.

- o Mature CE units showed a generally increasing trend between 1984 and 1987, followed by a 60 percent reduction between 1987 and 1988, to 1.71 BOP trips per critical year.

- o Mature B&W units showed an upward spike in 1985, but a 30 percent decrease overall between 1984 and 1988, to 2.23 BOP trips per critical year.

- o Overall, the mature nuclear plants showed a substantial reduction in BOP trips over the 5-year period, from an average of 4.4 BOP trips per critical year in 1984 to an average of 2.3 BOP trips per critical year in 1988.

The values for annual average BOP trips per unit per critical year (mature plants only) compared favorably with comparable values derived from the AEOD report NUREG-1275, Volume 5 (Reference 4). The AEOD values for trips per 1000 critical hours were multiplied by 8.76 to convert to trips per critical year. The resulting values were multiplied by 0.67 to approximate the BOP-related portion of the total trips. The resulting comparison is given below.

AEOD/SAIC Comparison
Annual Average BOP Trips per Unit per Critical Year
(Mature Units, 1984-1988)

	1984	1985	1986	1987	1988
From SAIC BOP data base	4.4	4.2	3.8	3.0	2.3
Derived from AEOD report data	4.9	4.6	4.0	3.1	2.3

Differences in the values from the two sources can be attributed to (1) differences in the definition of mature plants (AEOD used a "floating" definition; SAIC used a fixed population), (2) the approximation that BOP trips are two-thirds of the total trips, and (3) a difference in the definition of reactor trip (AEOD required control rod motion; SAIC did not).

4.2.3 BOP Trips per Critical Year, 5-Year Average Data

These data are quite uniform for mature plants among the four different NSSS vendors, ranging between 3.4 and 4.0 BOP trips per critical year for the 5-year period. The weighted average over the period (weighted by number of plants for each NSSS vendor) is 3.8 BOP trips per critical year.

4.2.4 Causation of BOP Trips - General

The general causation categories defined for the BOP study were component-related failure, human-related, design-related, procedure-related, and spurious or unknown. Searches of the BOP data base on these parameters indicated that nearly half (47 percent) of the BOP trips were caused by one or more component failures, nearly one-third (31 percent) were human-related, and the other categories were minor contributors.

It should be noted that these causation categories are not always clearly discernible in the LER descriptions, nor are they always clearly differentiated from each other. A design or procedural inadequacy, for example, could be termed human-related. However, in the preparation of the data base, an attempt was made to differentiate among the categories. For example, if an operator or technician correctly followed a procedure that was flawed and caused a BOP-related trip, this trip was categorized as procedure-related. If a procedure was not followed, and an incorrect or inadvertent human action caused a trip, this trip was categorized as human-related.

The human-related trips were further broken down as follows:

- o Operations activities 40%
- o Maintenance activities 40%

- o Surveillance activities 14%
- o Other activities 6%

A breakdown of the human-related trips as a function of activity by year (1984 through 1988) is given in Chapter 3, Table 3-15.

Both this study and the AEOD scram reduction report (Reference 4) identified equipment/component failure and human actions as the two largest contributors to reactor trips, either total trips, in the AEOD study, or BOP-related trips, in this study. However, the fractional contributors of these two general causes were different, as shown below:

	AEOD Study (Total trips, 1984-1987)	SAIC Study (BOP trips, 1984-1988)
Equipment/component failure	63%	47%
Human-related	25%	31%

This indicates that the human contributors to reactor trips originating in NSSS-related systems is smaller than it is in trips originating in BOP-related systems.

4.2.5 Multiple-Cause BOP Trips

Approximately 70 percent of the BOP trips were determined to be single-cause events. However, a substantial proportion (27 percent) would not have occurred in the absence of a second condition, and a few trips (3 percent) would not have occurred in the absence of two additional conditions.

An example of a double-cause trip is a situation where one channel of a BOP-related trip parameter instrument has failed undetected, and a second channel is actuated or taken out of service for testing, causing a trip.

One of the counter-intuitive findings of the study was that about 30 percent of the BOP trips are categorized as multiple-cause events. A detailed review of the data base entries for multiple-cause events indicates some "softness" in the data, and different reviewers might have categorized some events differently. However, it is evident that multiple-cause trips are more prevalent than previously assumed by many observers, and this finding has implications for statistical and risk assessment analysts. Most of the multiple-cause trips were not from coincident independent failures or common-cause events, but rather from pre-existing conditions (e.g., degraded operability states of various systems or components) that were revealed when a related system or component was actuated.

4.2.6 Causation of BOP Trips - Systems Implicated

The two largest system contributors to BOP trips were the feedwater system, causing 40 percent of the trips, and the turbine-generator system, contributing about 30 percent. The next largest contributors, the AC power and main steam systems, contributed about 12 percent and 6.5 percent, respectively. Other systems, contributing 3 percent or less to BOP trips over the study period, include air, circulating water, DC power, and instrumentation and control systems.

There is general agreement among recent studies that problems in the condensate/feedwater system are the leading cause of BOP-related (as well as total) reactor trips. The next largest system contributor to reactor trips is the turbine/generator system. Together, these two systems cause about 70 percent of the BOP trips and about half of the total reactor trips. Table 4-1 compares this study's estimates of system contributors to BOP-related trips to those derived from the AEOD report (Reference 4) and from the Mitre report (Reference 5). The Mitre estimates are considered somewhat distorted because they consider only the power conversion systems and not AC and DC power, water systems, air systems, etc. Other differences in the studies are mature plants versus all plants, trip definitions, and the time periods of the studies.

4.2.7 Causation of BOP Trips - Subsystems Implicated

Feedwater control was the dominant contributing subsystem to feedwater-related BOP trips. Within the turbine-generator system, the dominant contributing subsystem was instrumentation and control, primarily the electro-hydraulic control (EHC) subsystem. Feedwater control and

Table 4-1
Comparison of Estimates of
Initiating System Contributors to BOP-Related Reactor Trips

	Percent of BOP Trips		
	SAIC Study	AEOD* Study	Mitre** Study
Condensate/feedwater system	40	43	50
Turbine/generator system	30	27	29
AC power systems	12	15	-
Main steam system	6.5	8	7
Others	11.5	7	14

* Data from Table 3-13 of Reference 4 renormalized to include only BOP systems.

** Data derived from Table 2-11 of Reference 5.

turbine/generator I&C subsystem problems (component failure or human-related) together caused about 40 percent of the total BOP trips.

The AEOD report, NUREG-1275, Volume 5 (Reference 4) also identified the feedwater control and turbine/generator EHC subsystems as major contributors to reactor trips.

4.2.8 Causation of BOP Trips - Components Implicated

The clearly dominant "component" contributor to BOP trips was the human, generally causing about 30 percent of all BOP trips across the major system contributors. The next largest component contributors, generally much less significant than the human, were pumps, valves, electrical switchgear, and circuit cards. For the dominant systems, the data are characterized by a majority of the trips coming from very small contributions from very large numbers of components.

The AEOD scram reduction study (Reference 4) indicates that problems with feedwater regulating valves and feedwater pumps each contribute about 20 percent to the frequency of feedwater-related trips. This study, which examined component contributions in detail (see Appendix F) estimates that contributors from these components are less than half as large as the AEOD estimates, in the range of 7 to 8 percent each. This difference is probably attributable to different treatment of the human "component" in the two studies.

The fact that large numbers of components are each contributing small amounts to the feedwater-related BOP trips complicates resolution of the issue, and points toward using "integral" measures such as the adjustments to steam generator level trip setpoints being pursued by the Electric Power Research Institute in conjunction with the PWR Owners Groups.

4.2.9 Trends in BOP Trips as a Function of Architect/Engineer

The BOP data base was searched to see if positive or negative performance in terms of BOP trips could be correlated with the architect/engineer (A/E) responsible for designing the BOP. For the major A/E firms that have designed several nuclear units--Bechtel, Stone & Webster, Sargent & Lundy and Ebasco--no clear trends were evident in the data as a function of the A/E firm that designed the BOP.

4.2.10 Trends in BOP Trips as a Function of Plant Age

The data on BOP trips as a function of plant age were widely scattered; even the annual average values at a given age showed a large degree of variability. The overall trend, determined by a linear least squares fit of the annual average data, showed a reduction of about one BOP trip (during the 5 years considered in the study) for every 2 years of increasing age.

4.2.11 Trends in BOP Trips as a Function of Power Level

Approximately half of the BOP trips observed over the study period occurred above 75 percent power, and those trips were dominated by problems in the turbine-generator system. Nearly 30 percent of the observed trips occurred below 25 percent power, and they were dominated by problems in the feedwater system. The remaining trips were distributed evenly between the 25 percent to 50 percent range and the 50 percent to 75 percent range in power level.

Because most nuclear units spend most of their time about 75 percent power, it is surprising that only about half of the BOP-related trips occur in this power range. The fraction of BOP-related trips at high power levels is expected to increase as plant operators resolve feedwater control problems at the lower power levels, e.g., steam generator level instabilities, or the transition from manual to automatic feedwater control.

References

1. "Comparative Overview of Selected Scram Statistics," USNRC Office for Analysis and Evaluation of Operational Data, Proceedings of an NEA Symposium, Reducing the Frequency of Nuclear Reactor Scrams, Tokyo, Japan, April 1986.

2. "Reactor Scrams in the United States, A Regulator's Point of View," H.R. Denton, Director, ONRR, USNRC, Proceedings of an NEA Symposium, Reducing the Frequency of Nuclear Reactor Scrams, Tokyo, Japan, April 1986.

3. "1988 Performance Indicators for the U.S. Nuclear Power Industry," Institute of Nuclear Power Operations, March 1989.

4. NUREG-1275, Volume 5, "Operating Experience Feedback Report - Progress in Scram Reduction, Commercial Power Reactors," USNRC, Office for Analysis and Evaluation of Operational Data, March 1989, with Addendum.

5. NUREG/CR-4783, "Analysis of Balance of Plant Regulatory Issues," Mitre Corporation, January 1987.

5. EVALUATION OF RISK IMPLICATIONS

The objective of this task was to evaluate the impact of BOP-related events on the risk, as measured by estimated core melt frequency, of nuclear power plant operation. The task was divided into two parts. First, a quantitative analysis was performed to estimate the risk impact of reactor trips caused by BOP system failures. Second, a qualitative evaluation was performed of the impact of BOP-related events on safety system availability; this evaluation addressed events that did not necessarily result in a plant trip but did degrade the capability of a safety system.

The risk impact of BOP-related reactor trips was estimated by a parametric analysis of six probabilistic risk assessments (PRAs). Data on trip initiating events were selected from the BOP data base, representing "good" and "bad" BOP performance plants (as defined based on frequency of BOP-related trips). These initiating event data were used to replace the trip frequency used in each of the PRAs for initiating events such as turbine trip and loss of feedwater. The estimated core melt frequency was then recalculated and an assessment was made of the change in core melt frequency for each PRA using its data, the "good" performance data, and the "bad" performance data.

The impact of BOP-related events on safety system availability for events other than BOP-initiated reactor trips cannot be simply evaluated. Events have occurred, as reported in LERs, in which BOP system failures affected safety systems. It is difficult, however, to quantify the effects because of lack of consistency within the data base and lack of total system data (particularly an indication of the number of successful component demands). Therefore, a qualitative evaluation was performed, using data from the NRC's Accident Sequence Precursor Program. By identifying the BOP-related initiating events within the population of Precursor events, an estimate was made of the importance of BOP systems to accident sequences involving safety system degradation (i.e., sequences characterized by inadequate core cooling and resultant core damage).

5.1 Impact of BOP System Performance on Plant Core Melt Frequency

To evaluate the impact of BOP system performance on calculated core melt frequencies, six PRAs, representing five different nuclear power plants, were selected:

- "Connecticut Yankee Probabilistic Safety Study" (Reference 1).
- "Probabilistic Risk Assessment: Limerick Generating Station" (Reference 2).
- "A Review of the Limerick Generating Station Probabilistic Risk Assessment" (Reference 3).
- "Millstone Unit 1 Probabilistic Safety Study" (Reference 4).
- "A Review of the Millstone 3 Probabilistic Safety Study" (Reference 5).
- "Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3" (Reference 6).

  (Note: "PRA" is used as a generic term in this section, encompassing both the probabilistic safety studies and the probabilistic risk assessments listed above.)

These six PRAs provided analytical frameworks for estimating the effects of BOP system failures on risk as measured by calculated plant core melt frequency. The BOP-related transient initiator frequencies used in each PRA were varied according to the frequencies of actual BOP-related reactor trip causes, as extracted from the BOP data base developed for this study.

The frequencies of actual BOP-related reactor trip causes were drawn from a subset of the events in the BOP data base. Only mature plants were considered (plants that received an operating license before January 1, 1983). From the mature plants, the 10 plants with the best BOP trip performance and the 10 plants with the worst BOP trip performance during the period 1984 through 1987 were selected. Best and worst performance were determined based on the frequency of BOP-related trips per critical year.

A wide range of BOP trip performance resulted. The 10 plants with the best performance had 36 BOP-related plant trips during the 4-year period 1984 through 1987, an average of 0.9 trips per plant per year. The 10 plants with the worst BOP performance had a total of 194 BOP-related plant trips for the same time period, an average of 4.85 trips per plant per year. In addition, the worst performers generally had lower availability factors for the 4 years studied. This results in a larger difference between the best and worst performers when trips per critical year are used as the basis for comparison.

The transient initiator categories used in the PRAs were maintained. Trip causes from the BOP data base were assigned to the appropriate PRA categories.

Each PRA grouped transient initiators in a slightly different way. The Connecticul Yankee PRA, for example, separated BOP-related initiators into General Plant Transients, Loss of Feedwater Events, Inadvertent Opening of a Relief Valve Events, and other system-failure-related transients (e.g., Loss of Service Water). In comparison, the Millstone Unit 1 PRA separated the BOP-related initiators into Transients, Loss of Feedwater Transients, and Loss of the Power Conversion System (PCS) Transients.

For each of the PRAs, the data on actual BOP-related plant trip causes were combined and used in the manner most consistent with the transient initiator categories in the PRA. For example, the Oconee 3 and Millstone 1 PRAs used both a Loss of Feedwater and a Loss of PCS transient initiator category; thus both categories were used in the categorization of the BOP study data applicable to the Oconee 3 and Millstone 1 PRAs. However, the remaining PRAs used only a Loss of Feedwater <u>or</u> a Loss of PCS initiator; thus, for those PRAs, the BOP data were combined so that each BOP-related trip event contributed to the appropriate PRA transient initiator category. Similarly, the Connecticut Yankee PRA was the only one to handle the Inadvertent Opening of a Relief Valve as a separate transient initiator. This category was therefore included in the analysis of Connecticut Yankee, but for the remaining PRAs this type of event was treated as a Plant Transient.

Table 5-1 shows the number of BOP-related trips that occurred during the 4-year analysis period in each applicable PRA transient initiator category; the "good" and "poor" BOP performance plants are compared. Table 5-2 shows the average frequency of BOP-related trips in each transient initiator category, per calendar year and per critical year, over the 4-year period. As in Table 5-1, "good" and "poor" performance plants are compared.

The data from Table 5-2 were used to modify the transient initiator frequencies in each of the PRAs. The BOP-related trips per critical year were converted to an equivalent trips per calendar year using the following equation:

$$F_p = F_{BOP} \cdot A_p$$

where F_p = Initiator frequency for use in plant PRA
F_{BOP} = Initiator frequency from BOP data base (trips per critical year)
A_p = Plant availability factor

For example: The General Plant Transient category for the Connecticut Yankee PRA is equivalent to the Plant Transients category of Table 5-2. The average frequency of Plant Transients for the "good" performance PWRs from Table 5-2 is 0.82 per critical year. Connecticut Yankee had an availability factor of 0.713. Therefore, the frequency of Plant Transients at Connecticut Yankee, using the data for the "good" PWR plants would be:

$$F_p = (.82)\ (.713)$$
$$= .58/\text{yr.}$$

The BOP-related trip frequencies were converted to trips per critical year and then back to trips per calendar year when used in the PRAs for two reasons. First, the 10 plants in the "poor" BOP performance group had generally lower availability factors than the 10 plants in the "good" BOP performance group. In particular, some of the plants with lower availability factors had extended periods (in one case, over 2 years) during which the plant was not operating. By converting to trips per critical year, these periods of plant inactivity were eliminated and no longer distorted the initiator frequency calculations. Second, the events included

Table 5-1
BOP Induced Transients
Based on 1984-1987 LER Data

Transient Initiator Category	"Good" Performance Plants PWR	BWR	Total	"Poor" Performance Plants PWR	BWR	Total
Plant Transients	21	7	28	117	30	147
Loss of Main Feedwater	3	1	4	8	12	20
Loss of Power Conversion	3	1	4	6	20	26
Steam Line Relief Valve Opens	0	0	0	1	0	1

Table 5-2
Average Frequency of BOP-Induced Transients
Based on 1984-1987 LER Data

Transient Initiator Category	Trips Per Year PWR	Trips Per Year BWR	Trips Per Year Total	Trips Per Critical Year PWR	Trips Per Critical Year BWR	Trips Per Critical Year Total
			"Good" Performance Plants			
Plant Transients	0.66	0.88	0.71	0.82	1.24	0.89
Loss of Main Feedwater	0.09	0.13	0.10	0.12	0.18	0.13
Loss of PCS	0.09	0.13	0.10	0.12	0.18	0.13
			"Poor" Performance Plants			
Plant Transients	4.20	2.50	3.70	6.80	4.6	6.20
Loss of Main Feedwater	0.29	1.00	0.50	0.47	1.8	0.84
Loss of PCS	0.21	1.70	0.65	0.36	3.1	1.10
Steam Line Relief Valve Opens	0.04	0.00	0.03	0.06	0.0	0.04

in the BOP data base generally occurred at power. (There are some exceptions.) The data base for the 10 "good" and 10 "poor" performance plants is therefore restricted to 23.7 reactor years for the "poor" plants and 31.3 reactor years for the "good" plants.

The data conversion described above was used to produce plant-specific, BOP-related transient initiator frequencies. The results are presented in Table 5-3, along with the initiator frequencies used in the PRAs.

For the different plant transients considered, two PRA frequencies are given in Table 5-3. The first is the frequency used for all events contained within this category. The second frequency, listed as "PRA (BOP)," is a subset containing only those events associated with the BOP. Excluded from this group are the reactor transients such a spurious safety injection, spurious RPS actuation, etc.

It can be seen from Table 5-3 that the initiator frequencies used in the PRAs tend to be within the range of the data for the "good" and "poor" performance plants. There are a few exceptions. The most notable exception is the Millstone 1 Loss of Feedwater initiator frequency. This frequency is more than 30 percent less than the frequency obtained from the BOP data base for "good" performance BWRs. The Millstone 1 PRA used plant-specific data as the basis for its Loss of Feedwater initiating event frequency. Except for this one case, all six PRAs used data that, in comparison with the data derived from the BOP data base, are either within the expected range or conservative (i.e., higher than the "poor" performance plant data from the BOP data base).

The data in Table 5-3 were used to modify the core melt frequency calculations of each PRA. The first step was to determine the contribution to core melt frequency of each transient initiator as presented in the PRA. In some cases, only the dominant core melt accident sequences were provided in the PRA report; in other cases, the total contribution from each initiator was available. The next step was to replace the transient initiator frequencies used in the PRA with the LER-based, "good" and "poor" initiator frequencies shown in Table 5-3. After this substitution, the core melt frequencies attributable to BOP-related transients were calculated.

Table 5-3
Comparison of PRA Transient Initiator Frequencies to
BOP-Related Transient Initiator Frequencies
Based on 1984-1987 LER Data (events/yr)

	Plant Transients	Loss of Main Feedwater	Loss of Power Conversion System	Inadvertent Steam Relief Valve Opening
Connecticut Yankee				
PRA (all)	3.14	--	--	--
PRA (BOP)	1.93	0.36	--	4.2E-3
Good plants	0.58	0.17	--	0.00
Poor plants	4.80	0.59	--	0.04
Limerick (PECo)				
PRA (all)	3.98	--	--	--
PRA (BOP)	3.63(1)	--	1.78(2)	--
Good	0.77	--	0.22	--
Poor	2.80	--	3.00	--
Limerick (BNL)				
PRA	8.17(3,4)	--	1.23(5)	--
Good	0.77	--	0.22	--
Poor	2.80	--	3.00	--
Millstone 1				
PRA (all)	3.11	--	--	--
PRA (BOP)	2.67	0.096	0.435	--
Good	1.04	0.15	0.15	--
Poor	3.86	2.60	1.51	--
Millstone 3 (LLL)				
PRA (all)	7.24	--	--	--
PRA (BOP)	3.73	--	2.32	--
Good	0.55	--	0.16	--
Poor	4.60	--	0.60	--
Oconee				
PRA	5.70(3,6)	0.64(7)	0.21	--
Good	0.62	0.09	0.09	--
Poor	5.20	0.36	0.27	--

(1) Becomes 3.2 for ATWS sequences
(2) Becomes 2.2 for ATWS sequences
(3) Insufficient information to separate Reactor Trips from BOP Trips
(4) Becomes 7.39 for ATWS sequences
(5) Becomes 2.01 for ATWS sequences
(6) Becomes 7.0 for some sequences (all transient initiators combined)
(7) Becomes 0.7 for ATWS sequences

The following is an example using the General Plant Transient initiator category for the Connecticut Yankee PRA. The frequencies for this transient initiator category are: 3.14/year using data in the PRA; 0.58/year using the "good" plant performance data from the BOP data base; and 4.8/year using the "poor" plant performance data from the BOP data base. The Connecticut Yankee PRA provided a total core melt frequency contribution of 5.34E-5/year for the accident sequences initiated by a General Plant Transient. From this information, the conditional probability of a core melt at Connecticut Yankee, given a General Plant Transient initiator, is

$$\frac{5.34E\text{-}5/\text{year}}{3.14/\text{year}} = 1.70E\text{-}5$$

Using the data derived from the BOP data base, the BOP-transient-induced core melt frequency for Connecticut Yankee would be 9.5E-6/year (using the "good" plant performance data) or 8.2E-5/year (using the "poor" plant performance data).

Similar calculations can be made for the Loss of Feedwater transient initiator and the Inadvertent Opening of a Relief Valve transient initiator. The results of these calculations for Connecticut Yankee are shown in Table 5-4.

Table 5-5 shows the final results for each of the PRAs examined as part of this risk impact evaluation. Two sets of results are provided for the Limerick plant. The first set is based on the PRA performed by the utility, Philadelphia Electric Company (PECo); the second set is based on the results of the Brookhaven National Laboratory (BNL) review of the PRA. The BNL review resulted in an estimated core melt frequency an order of magnitude higher than the original utility assessment. Because of this disparity, both PRAs were evaluated.

In the evaluation of both the Limerick PRA review performed by BNL and the Oconee 3 PRA, it was not possible to separate the BOP transients from the reactor transients in the general Plant Transients initiator category. The totals for the "good" and "poor" core melt frequencies for these two PRAs are therefore slightly low because they do not include the contribution of reactor transients to the core melt frequency. However, the difference

Table 5-4
Contributions to Core Melt Frequencies at Connecticut Yankee Due to Changes in BOP-Related Transient Initiator Frequencies

	PRA	Low BOP Transient*	High BOP Transient*
General plant transient	5.3E-5/yr	9.5E-6/yr	8.2E-5/yr
Loss of main feedwater	3.8E-5/yr	6.4E-6/yr	2.2E-5/yr
IORV	5.9E-6/yr	--	5.9E-5/yr
Other	4.7E-4/yr	4.7E-4/yr	4.7E-4/yr
Total	5.5E-4/yr	4.9E-4/yr	6.3E-4/yr

* CMF calculated using study data on the frequency of BOP-related transient initiators at PWRs rated best/worst in terms of BOP performance.

Table 5-5
Summary of Calculated Core Melt Frequency Results
CMF(/yr)

	PRA	Low BOP Transient(1)	High BOP Transient(1)	ΔCMF(2)
Connecticut Yankee	5.5E-4	4.9E-4	6.3E-4	1.5E-4
Limerick (PECo)	1.5E-5	9.6E-6	1.7E-5	7.3E-6
Limerick (BNL)	1.0E-4	5.3E-5(3)	1.5E-4(3)	9.6E-5
Millstone 1 (Rev 0)	8.1E-4	8.0E-4	3.5E-3	2.7E-3
Millstone 3 (LLL)	1.0E-4	9.1E-5	9.6E-5	4.4E-6
Oconee 3	5.4E-5	4.6E-5(3)	5.3E-5(3)	6.7E-6

(1) CMF calculated using study data on the frequency of BOP-related transient initiators at 10 best and 10 worst BOP trip performance plants.

(2) ΔCMF = High BOP Transient CMF minus Low BOP Transient CMF.

(3) This does not include the contribution from reactor transients.

between the "good" and "poor" core melt frequencies is accurate since the contribution of reactor transient initiated core melt sequences would be the same for both cases.

As can be seen from Table 5-5, the impact on the core melt frequencies of the six PRAs varied considerably from plant to plant. The impact was greatest for Millstone 1, where an increase of 2.7E-3/year resulted from the use of the high BOP transient frequency versus the use of the low BOP transient frequency, i.e., "poor" plant data versus "good" plant data. This increase is due to two factors. One is the relatively high frequency of transients initiated by a loss of feedwater at the "poor" BOP performance BWRs. The second is the unique design of the Millstone 1 high pressure injection system. Millstone 1 utilizes the feedwater system to provide high pressure injection. Therefore a loss of feedwater not only trips the plant but also results in the failure of the high pressure injection system. (From plant specific data, the Millstone 1 PRA used a Loss of Feedwater initiator frequency lower than the corresponding "low BOP transient" frequency from the study data, 0.096 versus 0.15).

For the remaining plants the impact of the BOP system transients varies from 4 percent to nearly 100 percent of the total core melt frequency. Using the delta between the core melt frequencies resulting from the use of "good" and "poor" plant data as the measure of the impact of BOP system behavior, the three PWRs showed the least impact due to BOP related events; for all three PWRs, the difference was less than 30 percent. This is due in part to the smaller differences between the "good" and "poor" plant data for the Loss of Feedwater and Loss of PCS transients for PWRs compared to the BWR data. But it is also indicative of the contribution of BOP-related transients to the total core melt frequencies for BWRs and PWRs. In the BWR PRAs considered, BOP-related transients contributed a third or more of the total core melt frequency. For the PWR PRAs the BOP-related transients contributed only approximately 10 percent of the total core melt frequency. BOP-related transients contributed significantly more to the core melt frequencies of BWRs than PWRs, and the results of BWR PRAs are therefore affected by changes in BOP transient frequencies to a greater extent than the results of the PRAs for PWRs.

The events for the "good" and "poor" performance plants in the BOP data base have been categorized as Plant Transients, Loss of Feedwater events, Loss of PCS events and in one case a Spurious Opening of a Steam Relief Valve. This set of events does not include all of the types of BOP initiators generally found in a PRA. System failures that cause a plant trip and also affect the operability of systems used to mitigate the consequences of a plant trip are also considered as initiating events. For example, loss of air is considered as an initiating event that usually causes a plant trip, a loss of feedwater, and a degraded operating condition for the auxiliary feedwater (AFW) system in PWRs. These events can also be considered BOP-related transient initiators, and in some cases they contribute a significant fraction of the core melt frequency for a plant.

The analysis discussed above addressed the difference in plant risk due to the difference between the reliability characteristics of "good" and "poor" performance plants. Because the types of system failures that could trip a plant and also degrade a mitigating system's performance did not appear in the data for the "good" and "poor" performance plants, no difference in plant risk due to those types of failures could be calculated. Those types of failures generally have relatively low frequencies, on the order of 1E-3/year. It is therefore not surprising that there are no such events in the limited portion of the data base used in this analysis, which represents only approximately 55 critical years of reactor operation. The data base did include some partial failures of support systems, for example AC power and air systems. Those failures resulted in either a trip or a trip and loss of feedwater and were included in the categories used in the analysis.

Table 5-6 presents data that more completely address the importance of BOP systems to plant risk. This table includes the delta risk calculations described previously but it also includes the total contribution of BOP-related transient initiators as calculated in the six PRAs. This table shows that for the PWRs (Connecticut Yankee, Millstone 3, and Oconee 3), the contribution of BOP-related transients is significantly higher than the delta risk calculations would imply. The BOP support system initiators contribute more to the PWR PRA results than to the BWR PRA results. Although the delta risk calculations for the PWRs show a relatively minor impact on plant risk (as little as a 4 percent change) the importance of all BOP-related initiating events is somewhat greater.

Table 5-6
Contribution of BOP-Related Transients to Core Melt Frequency

Plant	Total CMF (/yr)	BOP-Related CMF (/yr)*	%	BOP ΔCMF (/yr)+	% of Total CMF
BWR					
Limerick (PECo)	1.5E-5	7.8E-6	52	7.3E-6	49
Limerick (BNL)	1.0E-4	<5.9E-5	<59	9.6E-5	96
Millstone 1	8.1E-4	2.9E-4	36	2.7E-3	333
PWR					
Connecticut Yankee	5.5E-4	8.1E-5	15	1.5E-4	27
Millstone 3	1.0E-4	1.4E-5	14	4.4E-6	4
Oconee 3	5.4E-5	<2.8E-5	<52	6.7E-6	12

(*) CMF due to transients initiated by events involving BOP systems

(+) The difference between the plant CMF using "good" and "poor" BOP performance plant data

5.2 BOP Influence on Accident Precursor Events

The three most recent reports from the Accident Sequence Precursor Program (References 7-9) were examined to evaluate the influence of BOP failures on accident precursor events.

The Accident Sequence Precursor Program reviews reports (LERs) of operational events at light water reactors to identify and categorize precursors to potential severe core damage accidents. The accident sequences considered in the program are those which could lead to inadequate core cooling. Accident sequence precursors are defined as events that are important elements in those accident sequences characterized by inadequate core cooling and resulting core damage. The precursor events of interest could be either initiating events or events that contribute to such sequences subsequent to the sequence initiator. This BOP influence evaluation focused exclusively on initiating events.

During 1984, approximately 2400 LERs were prepared by licensees to report operational events in accordance with NRC reporting requirements. Of these 2400 events, approximately 900 were selected for detailed review, and 48 of these were judged to meet the definition of accident precursor events. After inserting these events into the appropriate places of accident sequence event trees and quantifying the sequences, 18 were estimated to have an associated conditional probability of severe core damage $\geq 1 \times 10^{-4}$. That is, given the precursor event, there was a probability $\geq 1 \times 10^{-4}$ that the operability states of other systems and components would be such that inadequate core cooling and severe core damage would result. Information on these 18 precursor events is provided in Table 5-7.

Eleven of the 18 precursor events with a comparatively high probability of core damage had BOP initiators. These are also identified in Table 5-7 and consist of five feedwater/condensate system degradations, two station transformer failures that caused loss of offsite power, and four one-of-a-kind events (main generator bearing failure, MSIV spurious closure, moisture separator high-level trip, and a surveillance procedure inadequacy). Seven of the 18 comparatively high-probability core damage precursor events did not have BOP initiators.

Table 5-7
Summary
1984 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$**

Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
1. La Salle 1	373/84-054	No	RCIC turbine trip; clogged prefilter in actuator oil system.	RCIC turbine oil RCIC turbine steam	2.3×10^{-3}
2. La Crosse	409/84-011	Yes	Loss of offsite power; station transformer failure.	Diesel generators Safety Injection	9.9×10^{-4}
3. Quad Cities 1	254/84-014	No	Two LPCI valves would not open for RHR mode.	RHR; 1 HPCI MOV	6.7×10^{-4}
4. La Salle 2	374/84-017	Yes	Loss of feedwater, human error.	Turbine and motor-driven FW pumps	3.8×10^{-4}
5. Susquehanna 2	388/84-006	No	LPCI Train "A" inoperable.	LPCI train "B" RHR	3.3×10^{-4}
6. Brunswick 1	325/84-006	Yes	Loss of feedwater, human error.	Condensate; Instrument air	2.6×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report, Volumes 3 and 4, May 1987.

** P_{CD} = Probability of core damage.

Table 5-7
Summary
1984 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$** (Continued)

	Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
7.	Susquehanna 2	388/84-013	No	Test - Loss of T/G and loss of offsite power.	Diesel Generators (4-all failed to start); RCIC	2.2×10^{-4}
8.	St. Lucie 2	389/84-004	Yes	Main feedwater pump trip.	Auxiliary FW (started and tripped) Main Steam Safety Valve	2.0×10^{-4}
9.	St. Lucie 2	389/84-011	Yes	Main generator bearing failure - loss of load.	Auxiliary FW (started and tripped)	2.0×10^{-4}
10.	Indian Pt. 3	286/84-015	Yes	Loss of offsite power; station transformer failure.	Diesel Generators (1 breaker failed to close)	1.9×10^{-4}
11.	Davis Besse	346/84-003	Yes	MSIV closure during SFRCS testing.	MSIV, MSSV, SFRCS, AFW	1.5×10^{-4}
12.	Brunswick 2	324/84-018	Yes	Moisture separator high-level trip.	Condensate, Turbine Bypass	1.4×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report, Volumes 3 and 4, May 1987.

** P_{CD} = Probability of core damage.

Table 5-7
Summary
1984 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$** (Continued)

Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
13. Susquehanna 1	387/84-010	No	Stuck open SRV during ADS test.	ADS, SRVs	1.4×10^{-4}
14. Browns Ferry 3	296/84-012	Yes	Low RV water level; inadequate surveillance procedure.	SRVs, Condensate Booster Pumps	1.2×10^{-4}
15. Brunswick 1	325/84-014	No	Erroneous APRM signal to RPS.	RCIC, SRV, MSIV	1.2×10^{-4}
16. Duane Arnold	331/84-001	Yes	FW recirculation valve failed open, decreasing FW flow.	SRV, HPCI, RCIC	1.2×10^{-4}
17. Browns Ferry 3	296/84-015	Yes	Failure of condensate pump, low RV level.	Condensate pump breaker, CRD pumps	1.1×10^{-4}
18. Browns Ferry 1	259/84-027	No	MSRV leakage.	TORUS	1.0×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report, Volumes 3 and 4, May 1987.

** P_{CD} = Probability of core damage.

During 1985, approximately 3000 LERs were prepared by licensees to report operations events in accordance with NRC reporting requirements. Of these 3000 events, approximately 1400 were selected for detailed review, and 63 of these were judged to meet the definition of accident precursor events. After inserting these events into the appropriate places of accident sequence event trees and quantifying the sequences, 11 were estimated to have an associated conditional probability of severe core damage $\geq 1 \times 10^{-4}$. Information on these 11 precursor events is provided in Table 5-8.

Nine of the 11 precursor events with a comparatively high probability of core damage had BOP initiators. These are also identified in Table 5-8 and consist of seven feedwater/condensate system degradation, one auxiliary transformer degradation, and one turbine pressure regulator failure. Only two of the 11 precursor events with a comparatively high probability of core damage did not have BOP initiators.

During 1986, approximately 2900 LERs were prepared by licensees to report operations events in accordance with NRC reporting requirements. Of these events, 1320 were selected for detailed review, resulting in 34 that were judged to meet the definition of accident precursor events. After inserting these events into accident sequence event trees and quantifying the sequences, six events were estimated to have an associated conditional probability of severe core damage $\geq 1 \times 10^{-4}$. Information on these six precursor events is provided in Table 5-9.

Three of the 6 precursor events with a comparatively high probability of core damage had BOP initiators. These involved (1) loss of pressure in the turbine governor oil system, (2) fuse-related problems in an electrical bus control circuit, and (3) a faulted controller for condenser steam dump valves.

Summary

For the 3-year period 1984 through 1986, 145 precursor events were identified from LERs, and 35 of these precursors had estimated conditional probabilities of severe core damage $\geq 1 \times 10^{-4}$. Twenty-three of these 35 more significant precursor events (66 percent) involved BOP initiators. Thus, the fraction of BOP initiation of the more significant precursor

Table 5-8
Summary
1985 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$**

	Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
1.	Davis Besse	346/85-013	Yes	Loss of feedwater; control system failure.	AFW, PORV	1.1×10^{-2}
2.	Hatch 1	321/85-018	No	Spurious SRV actuation from flooded electrical panel.	SRV, MSIV (Unrelated HPCI trip)	1.8×10^{-3}
3.	San Onofre 1	206/85-017	No	Partial loss of offsite power.	AC vital buses diesel generators	9.4×10^{-4}
4.	Turkey Pt. 3	250/85-021	Yes	Failure of feedwater control valve to open.	AFW, Instrument Air	9.0×10^{-4}
5.	Trojan	344/85-009	Yes	High temperature trip on auxiliary transformer.	AC Power, AFW (multiple trips.)	4.5×10^{-4}
6.	Davis Besse	346/85-002	Yes	Feedwater control system failure.	AFW (inadvertent trip), SFRCS	3.0×10^{-4}
7.	Oyster Creek	219/85-012	Yes	Turbine pressure regulator failure; MSIV closure.	MSIV, SDV, Fire Deluge System, Relief Valves, RCIC	2.3×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report, Volumes 1 and 2, December 1986.

** P_{CD} = Probability of core damage.

Table 5-8
Summary
1985 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$** (Continued)

	Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
8.	Hatch 1	321/85-010	Yes	Loss of vital AC, FW pump runback, FW pump trip.	HPCI, RCIC, SBGTS, RWCU, MFW, RPS	2.3×10^{-4}
9.	Grand Gulf	416/85-050	Yes	Loss of feedwater; condensate and FW pump trips.	Hotwell level indication, HPCS, RCIC	1.8×10^{-4}
10.	Hatch 2	366/85-030	Yes	Loss of feedwater; condensate booster pump failure.	HPCI, RCIC, SBGTS, Containment Isolation	1.2×10^{-4}
11.	Browns Ferry 1	259/85-016	Yes	Loss of feedwater; controller failure of turbine-driven feed pumps.	HPCI, RCIC, MSIV, Containment Isolation	1.0×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report, Volumes 1 and 2, December 1986.

** P_{CD} = Probability of core damage.

Table 5-9
Summary
1986 Precursor Report Data*
Events With Conditional Estimated $P_{CD} \geq 1 \times 10^{-4}$**

	Plant	LER	BOP Initiator	Initiating Event	Involved Systems	Conditional Estimated P_{CD}
1.	Catawba 1	413/86-031	No	Weld failure in letdown line	CVCS, letdown heat exchanger, MCC, 600v AC	3.3×10^{-3}
2.	Turkey Point 3	250/86-039	Yes	Loss of turbine governor oil system pressure	Turbine control oil, PORV, auto rod control	1.4×10^{-3}
3.	Robinson 2	261/86-005	Yes	Failed or loose fuse in electrical control circuit	AC distribution emergency diesels PORV, MSIV, SI	3.0×10^{-4}
4.	Indian Pt. 2	247/86-035	No	Loose wires in RPS relay circuits	RPS, AFW	2.9×10^{-4}
5.	Catawba 2	414/86-028	No	Inadvertent opening of SG PORVs	SI, AFW, auxiliary shutdown panels, CVCS	1.1×10^{-4}
6.	Indian Pt. 2	247/86-017	Yes	Faulted controller caused opening of condenser steam dump valves	SI, MSIV, steam dump controller	1.0×10^{-4}

* Reference: NUREG/CR-4674, Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report, Volumes 5 and 6, May 1988.

** P_{CD} = Probability of core damage.

events is about the same as the fraction of BOP initiation of reactor trips in general. A summary of the BOP influence on precursor events for the years 1984 through 1986 is given in Table 5-10.

5.3 Summary of Risk Implications

The results of the delta risk analysis (Section 5.1) and the evaluation of BOP-related precursor events (Section 5.2) both show that the reliability of BOP systems can have a significant impact on the risk profiles of nuclear power plants. For BWRs, in particular, plant core melt frequency appears to be highly sensitive to the frequency of BOP-related transients. Using the study data for "poor" BOP performance BWRs, the delta risk analysis yielded core melt frequencies that were 2 to 4 times greater than the frequencies obtained using the data for the "good" BOP performance BWRs. The corresponding differences for PWRs were comparatively small, ranging from a factor of 1.1 to a factor of 1.3.

Twelve of the 23 precursor events that were considered to be BOP-related and had a high probability of resulting in core damage occurred at BWRs. This is a disproportionate number of such events at BWRs, since approximately two-thirds of all operating U.S. reactors are PWRs. This finding supports the conclusion that BOP-related events are more important, from a risk perspective, at BWRs.

The overall impact of BOP system performance is greater than shown by the delta risk analysis for both BWRs and PWRs. Each of the PRAs used in the analysis included transient initiators with relatively low frequencies that can be considered BOP-related but did not appear in the study data base for the "good" and "poor" BOP performance plants. Therefore the risk associated with those types of events is not reflected in the delta risk calculations. However, if those types of events are included in the BOP contribution to core melt frequency, the contribution of BOP-related transients ranges from 14 percent to approximately 50 percent for PWRs, and from 36 percent to 59 percent for BWRs.

Table 5-10
Summary of BOP Influence on Precursor Events, 1984-1986

Precursor events designated

48 events in 1984, out of approximately 2400 LERs

63 events in 1985, out of approximately 3000 LERs

34 events in 1986, out of approximately 2900 LERs

1984 precursor events

18 of 48 events had estimated conditional $P_{CD} \geq 1 \times 10^{-4}$

11 of 18 events had BOP initiators

5 feedwater/condensate system degradation
2 station transformer failure LOOP
1 main generator bearing failure
1 MSIV spurious closure
1 moisture separator high-level trip
1 surveillance procedure inadequacy

7 of 18 events did not have BOP initiators

1985 precursor events

11 of 63 events had estimated conditional $P_{CD} \geq 1 \times 10^{-4}$

9 of 11 events had BOP initiators

7 feedwater/condensate system degradation
1 auxiliary transformer degradation
1 turbine pressure regulator failure

2 of 11 events did not have BOP initiators

1986 precursor events

6 of 34 events had estimated conditional $P_{CD} \geq 1 \times 10^{-4}$

3 of 6 events had BOP initiators

1 loss of turbine governor oil system pressure
1 faulted or loose fuse in an electrical bus control circuit
1 faulted controller for condenser steam dump valves

3 of 6 events did not have BOP initiators

For 1984 through 1986

23 of 35 "high P_{CD}" events (66%) had BOP initiators

References

1. "Connecticut Yankee Probabilistic Safety Study," Northeast Utilities Service Company, Report No. NUSCO-149, February 1986.

2. "Probabilistic Risk Assessment: Limerick Generating Station," Philadelphia Electric Company, March 1981.

3. NUREG/CR-3028, "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, February 1983.

4. "Millstone Unit 1 Probabilistic Safety Study," Northeast Utilities Service Company, Report No. NUSCO-147, July 1985.

5. NUREG/CR-4142, "A Review of the Millstone 3 Probabilistic Safety Study," Lawrence Livermore National Laboratory, April 1986.

6. "Oconee PRA: A Probabilistic Risk Assessment of Oconee Unit 3," Duke Power Company and the Electric Power Research Institute, Report No. NASC/60, June 1984.

7. NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: 1985, A Status Report," Volumes 1 and 2, December 1986.

8. NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report," Volumes 3 and 4, May 1987.

9. NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report," Volumes 5 and 6, May 1988.

6. FINDINGS AND RECOMMENDATIONS

The major finding of this study was the dramatic reduction in BOP-related trips at commercial nuclear power plants over the 5-year study period from January 1, 1984 through December 31, 1988. This improved performance reduces the urgency of regulatory action to address BOP-related safety concerns. However, regulatory actions can be taken to (1) address the problems of licensees whose BOP trip performance is substantially less favorable than the industry average, and (2) maintain or further improve the performance levels achieved toward the end of the study period.

6.1 Findings

1. **For the 76 mature nuclear units (OL before January 1, 1983) in the study data base, the average number of BOP trips per unit was reduced from 4.4 per critical year in 1984 to 2.4 per critical year in 1988.**

   The average mature unit over the 5-year period experienced 3.8 BOP trips per critical year. The corresponding value for the best-performing unit in the data base was 0.2 BOP trips per critical year; the worst-performing unit experienced 11.2 BOP trips per critical year.

2. **On a calendar year basis, for the 76 mature nuclear units in the study data base, the average number of BOP trips per unit was reduced from 2.8 per calendar year in 1984 to 1.6 per calendar year in 1988.**

   The average mature unit over the 5-year period experienced 2.3 BOP trips per calendar year. The best-performing unit experienced 1 BOP trip in 5 years; the worst-performing unit experienced 34 BOP trips in 5 years.

3. **Nearly 30 percent of the BOP-related trips resulted from multiple-cause events.**

   This is a surprisingly large fraction of multiple-cause BOP trip events. Although there is some "softness" in the data, it is clear that multiple-cause events are more prevalent than previously assumed by many observers, and this finding has implications for statistical and risk assessment analysts. Most of the multiple-cause trips were not from coincident independent failures or common-cause events, but rather from pre-existing conditions (e.g., degraded operability states of various systems or components) that were revealed when a related system or component was actuated.

4. **Approximately 70 percent of the BOP-related trips resulted from a single event.**

   A single component failure was the causative mechanism in 49 percent of these single-cause trips, and a single human action accounted for approximately 34 percent. The balance of the single-cause events were of design, procedures or environmental origin, with a few classified as spurious or unknown.

5. **Considering BOP trips resulting from both single and multiple causes, nearly four out of every five events contributing to BOP trips were either component/equipment failures (47 percent) or human actions (31 percent).**

   Clearly the two most dominant general contributors to BOP trip causation are component/equipment failures and human actions. The value cited for human actions does not include design or procedural inadequacies, which were categorized separately. It follows from this finding that, in order to be successful, programs directed at achieving reductions in BOP-related trip frequencies will need to contain both a technical element (component, system or functional reliability improvement) and a human performance element (a reduction in human errors in operations, maintenance and surveillance).

6. **NSSS Owners Groups with aggressive trip reduction programs are apparently achieving results in the form of reduced frequencies of BOP-related trips.**

   Table 3-2, which shows annual average BOP trips per unit per calendar year by NSSS vendor, suggests that:

- the Westinghouse Owners Group's Trip Reduction and Assessment Program (TRAP) began to show results in 1985;

- the Babcock and Wilcox Owners Group's Safety and Performance Improvement Program (SPIP) began to show results in 1986; and

- the General Electric Owners Group's Scram Frequency Reduction Program and the Combustion Engineering Owners Group's Scram Reduction Program did not begin to show results until 1988.

7. **At the system level, BOP trip causation was dominated by the condensate/feedwater system (40 percent of total trips) and the turbine/generator system (30 percent of total trips).**

   The degree of dominance of the two major contributors to BOP trip causation, the condensate/feedwater system and the turbine/generator system, was significant. The next largest contributor, AC power systems, contributed only 12 percent. Proceeding down the list, main steam systems contributed 6.4 percent, air systems about 3 percent, and the other major systems contributed 2 percent or less (e.g., instrumentation and control systems, circulating water systems, etc.).

8. **At the subsystem level, BOP trip causation was dominated by the feedwater control subsystem (61 percent of feedwater-related trips; 25 percent of total trips) and the turbine/generator instrumentation and control subsystem (60 percent of turbine/generator related trips; 18 percent of total trips).**

   Taken together, feedwater control and turbine/generator I&C problems caused more than 40 percent of the total BOP trips. Many of the feedwater control problems were at low power levels, often associated with manual feedwater control or the transition from manual to automatic feedwater control. The turbine/generator I&C problems centered primarily on the electro-hydraulic control (EHC) system for the turbine.

9. **At the component level, excluding the human "component," BOP trip causation was not dominated by any single component or small group of components.**

A majority of the BOP-related trips was caused by aggregated small contributions from a very large number of different components. Pumps, valves and circuit cards were the largest contributors in most cases, but none of these contributed a large fraction of the total. This complicates the task of achieving further improvements by requiring that a component reliability improvement program be very broad-based, and not focused on a few major contributors.

10. **Nearly all the units with the best BOP trip performance (fewest BOP-related trips) have motor-driven feedwater pumps; nearly all the units with the poorest BOP trip performance (highest numbers of BOP trips) have turbine-driven feedwater pumps.**

Feedwater systems with motor-driven feedwater pumps perform more reliably than systems with turbine-driven feedwater pumps. In addition, plants with excess feedwater capacity perform only marginally better than plants without excess feedwater capacity. Apparently, the combination of feedwater control characteristics and reactor trip setpoints on steam generator level do not usually allow operators enough time to utilize excess feedwater pump capacity to avoid a trip when a feedwater pump is lost.

11. **From a risk perspective, BOP-related transients contribute significantly more, on a fractional basis, to the estimated core melt frequencies of BWRs than they do to PWRs.**

Core melt frequency estimates in BWR PRAs are more affected by changes in BOP transient frequencies than are the corresponding estimates for PWR PRAs. Based on a limited number of PRA comparisons, the incremental core melt frequencies between "good" and "poor" performers in terms of BOP-related trips were factors of 2 to 4 for BWRs and factors of 1.1 to 1.3 for PWRs.

12. BOP-related transients are the initiating events for approximately two-thirds of the more significant accident precursor events.

The NRC's Accident Sequence Precursor program estimates the conditional probability of severe core damage associated with the occurrence of operating events. For the years 1984 through 1986, 35 operating events were calculated to have estimated conditional probabilities of severe core damage greater than or equal to 1×10^{-4}. Two-thirds of these events (23 events) had BOP initiators. This two-thirds fraction is approximately the same as the BOP-related contribution to total unplanned reactor trips.

6.2 Recommendations

The dramatic reduction in the number of BOP-related reactor trips at commercial nuclear power plants over the 5-year period ending December 31, 1988, reduces the urgency of regulatory actions directed at BOP performance improvements. However, regulatory actions can and should be taken to (1) maintain the trend toward decreasing numbers of BOP-related reactor trips among NRC licensees, and (2) address the problems of licensees whose performance is substantially less favorable than the industry average.

6.2.1 General Recommendations

1. Communicate to licensees and applicants, in the form of an informational generic letter, the results of recent studies on BOP-related trips and overall scram reduction experience.

This generic letter should point out where improvements in trip reduction can be made while formally acknowledging the recent improved performance of most licensees. Transmitted with this informational generic letter should be a copy of this BOP-specific study and a copy of Volume 5 of NUREG-1275, "Operating Experience Feedback Report - Progress in Scram Reduction," March 1989. This generic letter, with the attached reports, will provide licensees with a basis for making decisions on their plant-specific programs for minimizing unplanned reactor trips.

2. **Identify, monitor and communicate with licensees who are not achieving an acceptably low frequency of BOP-related trip events at their facilities.**

For the purposes of identifying licensees in need of increased regulatory attention, an "acceptably low frequency of BOP-related trip events" could be defined as the 5-year (1984-1988) industry average, 3.8 BOP trips per critical year, plus one standard deviation (approximately 2.2) or about 6 BOP trips per critical year. It is recommended that licensees who do not achieve a frequency less than about 6 BOP trips per critical year, in any given year, be candidates for increased regulatory attention to BOP performance. Actions could include consultations with the licensee on how the problem is being addressed, and special inspections on BOP systems reliability, the adequacy of root cause analysis of reactor trip events, and performance trends.

3. **NRC should work with INPO, the Owners Groups, and EPRI to assist licensees in achieving and maintaining an acceptably low frequency of BOP-related trip events at their nuclear plants.**

Because of the limited reach of NRC's regulations into the BOP systems, improvements in BOP performance will (and have) come largely through industry initiatives on the basis of economics and reliability. However, based on the findings of this study, NRC could stimulate improvements in BOP systems performance by working with industry in the following areas:

a. Encourage a steadily increasing level of industry performance of root cause analyses of reactor trips.

b. Encourage Owners Groups and individual utilities to continue their aggressive pursuit of trip reduction programs.

c. Process requests for BOP-related changes to Technical Specifications in a timely manner.

d. Investigate why turbine-driven main feedwater pumps do not perform as reliably as motor-driven main feedwater pumps.

e. Investigate how to make better use of excess feedwater pumping capacity (where it exists) to reduce the frequency of feedwater-related reactor trips.

4. **NRC should formally incorporate BOP trip avoidance experience into the Systematic Assessment of Licensee Performance (SALP) process, e.g., as an element in the Safety Assessment/Quality Verification category.**

This action would increase the visibility level of BOP performance trends among licensee management, resident inspectors, NRC Project Managers, and NRC senior management. This action could be coordinated with programs on performance indicators, maintenance improvements and routine inspections.

6.2.2 Specific Recommendations

1. **Establish a responsibility center within NRC to specifically monitor and evaluate BOP-related reactor trip experience.**

The functions of this responsibility center would be to identify "outliers" in terms of BOP trip experience; compare licensee and overall industry performance with goals established by NRC and by industry; compare industry performance with that in foreign countries; and periodically report to the NRC management on the state of BOP systems performance in the industry.

2. **NRC should expand the role of BOP systems in ongoing NRC activities, specifically in the areas of inspections, maintenance policy, Technical Specifications improvements, human factors and training, severe accident policy/IPEs, the Accident Sequence Precursor program, and advanced reactors/standardization, as discussed below.**

a. Inspections

NRC should assure that Resident Inspectors periodically evaluate the BOP trip experience of their units. The special BOP inspection program should be re-instituted for plants with particularly poor BOP trip performance histories.

b. Maintenance Policy/Rulemaking

Draft Regulatory Guide DG-1001, "Maintenance Programs for Nuclear Power Plants," should suggest a goal of less than or equal to one BOP trip per calendar year on the same basis (e.g., capacity factor greater than 25 percent) as the INPO 1990 goal for total reactor trips of 1.5 per calendar year. NRC should evaluate BOP trip performance as a function of whether a licensee's maintenance program provides the same level of attention to BOP systems as is given to safety systems. Specific emphasis should be given to main feedwater control systems and turbine electro-hydraulic control systems.

c. Technical Specifications Improvements

NRC should evaluate, in coordination with licensees, BOP-related safety limits, limiting conditions for operation, surveillance frequencies and trip setpoints for their effects on BOP trip causation. Specific emphasis should be given to steam generator level trip setpoints, steam flow/feed flow mismatch trips, and the frequency of turbine control valve testing.

d. Human Factors and Training

The NRC programs on human factors and training should include an element on avoiding BOP trips caused by operations and maintenance errors (and to a lesser degree surveillance testing errors) by both licensed and unlicensed operations personnel.

e. Severe Accident Policy/IPEs

Because of the influence of BOP system failures on core melt frequency estimates, the NRC review of IPEs should compare the BOP initiating event frequencies used in the analyses with the experience values reported herein.

f. Accident Sequence Precursor Program

NRC should build on the evaluation reported herein and perform an in-depth evaluation of the influence of BOP system failures or degradations on the higher-ranking accident precursor events.

g. Advanced Reactors and Standardization

With regard to BOP considerations, the NRC reviews of advanced reactors and standardized designs should focus on improvements in the main feedwater control and turbine electro-hydraulic control systems. Further, NRC should encourage the use of motor-driven (rather than turbine-driven) main feedwater pumps.

3. **NRC should expand the evaluation of the risk implications of BOP events to additional PRA studies to test the validity of the risk-related findings made herein.**

Based on a comparison with 6 PRA studies (3 PWR, 3 BWR, two on the same BWR) this study concluded that the incremental difference in core melt frequency estimates between "good" and "poor" BOP performers was a factor of 2 to 4 for BWRs and a factor of 1.1 to 1.3 for PWRs. NRC should expand this evaluation to more PRA studies to test the validity of these estimates.

4. **NRC should investigate the implications of the relatively large numbers of multiple-cause events for statistical and risk analyses.**

The methods used in statistical or risk analyses for estimating common-cause or dependent-failure events may not adequately account for the types of multiple failures found in this study for magnitudes as large as 20 to 30 percent of total failures. NRC should examine the implications of these higher frequencies of multiple-cause events, which are (in general) neither dependent failures nor common-cause events. Also, the trade-offs associated with additional component testing or more frequent testing would uncover more undetected degradations, but it could also result in more inadvertent trips associated with the testing.

APPENDIX A

Sample Entries from the BOP Data Base

*** BOP RELATED EVENT ***
=================

Form: 88
Plant Name: Dresden 2

Event ID: 237/85-035 Power Level: 00%
Event Date: 09/29/85 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: Steam Relief
BOP Component: Bellows

Cause 1: Design Related
Cause 2: Component Failure: Bellows

Event Description: Steam flow through seal steam relief valves may have damaged the bellows (expansion joint) during normal system operation. The damaged bellows resulted in a low condenser vacuum causing a turbine trip and a subsequent reactor scram.

*** BOP RELATED EVENT ***
=================

Form: 229
Plant Name: Quad Cities 1

Event ID: 254/86-030 Power Level: 90%
Event Date: 10/16/86 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Human

Cause 1: Human Related: Maintenance

Event Description: While testing the electro-hydraulic control system, test personnel generated a turbine bypass valve open signal. Subsequent excess steam flow caused MSIV closure and a reactor trip on MSIV position.

*** BOP RELATED EVENT ***

Form: 230
Plant Name: Quad Cities 1

Event ID: 254/86-038
Event Date: 12/09/86

Power Level: 15%
Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: Condenser
BOP Component: Unknown

Cause 1: Unknown
Cause 2: Human Related: Operations

Event Description: Adequate condenser vacuum could not be maintained during startup. Personnel attempted to continue startup hoping the condition would improve. It didn't and the plant tripped. The reason for failure to maintain condenser vacuum was not given.

*** BOP RELATED EVENT ***

Form: 231
Plant Name: Quad Cities 1

Event ID: 254/87-005
Event Date: 03/17/87

Power Level: 92%
Trip Type: Auto

BOP System: Main Steam
BOP Component: Valve

Cause 1: Component Failure: Valve
Cause 2: Human Related: Maintenance

Event Description: Turbine stop valve closure caused a turbine trip and a reactor trip. The stop valve closure was caused by a high level in the moisture separators which in turn was partially due to a stuck open level control valve. Operator attempts to repair the valve contributed to this event because other level control valves could not properly handle sufficient flow.

*** BOP RELATED EVENT ***

Form: 232
Plant Name: Palisades

Event ID: 255/84-015 Power Level: 48%
Event Date: 08/04/84 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Human

Cause 1: Human Related: Maintenance

Event Description: EHC system repairs were not properly performed. The resulting vibration in the EHC system caused a turbine trip and reactor trip.

*** BOP RELATED EVENT ***

Form: 233
Plant Name: Palisades

Event ID: 255/85-010 Power Level: 98%
Event Date: 08/11/85 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Transformer

Cause 1: Component Failure: Transformer

Event Description: A motor operated auto transformer operated erratically during a voltage adjustment, The erratic performance caused a loss of generator load and a turbine/reactor trip.

*** BOP RELATED EVENT ***

Form: 276
Plant Name: Robinson 2

Event ID: 261/87-020
Event Date: 07/10/87

Power Level: 100%
Trip Type: Auto

BOP System: DC Power
BOP Component: Wire

Cause 1: Component Failure: Wire

Event Description: The plant tripped on July 10, 1987 due to a Feedwater regulator valve failure caused by an electrical short in the DC wire to one of the two safeguard solenoids for the valve operator. The solenoid failed due to entrapped water in the solenoid condulet. The Feedwater regulator valve closure resulted in Steam/Feedwater flow mismatch coincident with a low SG level.

*** BOP RELATED EVENT ***

Form: 276
Plant Name: Robinson 2

Event ID: 261/87-020
Event Date: 07/16/87

Power Level: 72%
Trip Type: Auto

BOP System: Feedwater
BOP Subsystem: Feedwater Control
BOP Component: Feedwater regulator valve

Cause 1: Component Failure: Feed reg valve

Event Description: On July 16, 1987, the reactor tripped on low SG level coincident with Steam/Feedwater flow mismatch caused by the failure of the same Feedwater regulator valve as described in the 7/10/87 event. This time, the valve failure was caused by the impaired function of the valve positioner.

*** BOP RELATED EVENT ***

Form: 278
Plant Name: Robinson 2

Event ID: 261/88-001 | Power Level: 66%
Event Date: 01/19/88 | Trip Type: Auto

BOP System: Turbine Generator
BOP Component: Regulator valve

Cause 1: Surveillance
Cause 2: Component Failure: Regulator valve

Event Description: Normal surveillance testing of the turbine was conducted. Due to wear and tear, an air operated pressure regulator valve did not function properly and was unable to withstand the back pressure after the turbine was returned to service. A pressure loss in the turbine caused the turbine to trip, which subsequently caused a reactor trip.

*** BOP RELATED EVENT ***

Form: 280
Plant Name: Monticello

Event ID: 263/85-008 | Power Level: 100%
Event Date: 04/11/85 | Trip Type: Auto

BOP System: AC Power
BOP Subsystem: High Voltage Offsite
BOP Component: Transformer

Cause 1: Human Related: Maintenance

Event Description: A phase fault occurred while a transformer was being restored from maintenance. The fault was caused by a "non-plant" worker who forgot to remove grounding cable after the completion of the work. Because the tripping control system was not yet in service, the turbine control system initiated reactor scram.

*** BOP RELATED EVENT ***

Form: 281
Plant Name: Monticello

Event ID: 263/85-010
Event Date: 06/12/85

Power Level: 100%
Trip Type: Auto

BOP System: Main Steam
BOP Component: Human

Cause 1: Human Related: Surveillance

Event Description: During surveillance testing of the main steam low pressure instrumentation, a human error contrary to the approved procedure was committed leading to MSIV closure, which then lead to a reactor trip. The technician failed to properly valve in and out the appropriate pressure switch channels.

*** BOP RELATED EVENT ***

Form: 368
Plant Name: Salem 1

Event ID: 272/87-007
Event Date: 06/02/87

Power Level: 100%
Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control

Cause 1: Environment: Lightning

Event Description: A turbine/reactor trip occured when lightning struck in the vicinity of the DEANS switching station causing a momentary loss of the 500KV transmission line and actuating the SALEM/DEANS "cross trip scheme" for "generator protection". This X-trip was established to prevent potential generator instability at Salem 1.

*** BOP RELATED EVENT ***

Form: 370
Plant Name: Salem 1

Event ID: 272/88-009 Power Level: 100%
Event Date: 03/30/88 Trip Type: Manual

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Indicator

Cause 1: Component Failure: Indicator
Cause 2: Human Related: Operations

Event Description: During full power operation, EHC pump 12 tripped and the standby EHC pump 11 failed to auto start. With both pumps failed, the control oil system pressure decreased and the turbine governor valves drifted closed. The reactor was then manually tripped due to increasing T avg. Prior to the event, the EHC oil had been leaking, and constant refill was required. However, the level indicator malfunctionned, and constantly indicated normal or full level, although oil level was actually at the pump low level lockout setpoint. Thus the lack of communication and level instrumentation failure were the root causes of this event.

*** BOP RELATED EVENT ***

Form: 371
Plant Name: Diablo Canyon 1

Event ID: 275/84-015 Power Level: 2%
Event Date: 05/08/84 Trip Type: Auto

BOP System: Main Steam
BOP Subsystem: Steam Relief
BOP Component: Circuit card

Cause 1: Component Failure: Circuit card

Event Description: A failed pressure control module in the steam dump control system allowed several 40% steam dump valves to open, initiating a high steam flow coincident with LO-LO Tavg that tripped the reactor. This event occurred during startup.

*** BOP RELATED EVENT ***

Form: 374
Plant Name: Diablo Canyon 1

Event ID: 275/84-030 Power Level: 21%
Event Date: 11/24/84 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Human

Cause 1: Human Related: Construction
Cause 2: Component Failure: Valve

Event Description: This event was caused by a loose wire in the turbine control system, causing the system to malfunction. Additionally, the 40% condenser dump valves failed to open resulting in a turbine/reactor trip. The cause for the dump valve failure to open was traced to the installation of control wiring according to an incorrect drawing of the electrical connections.

*** BOP RELATED EVENT ***

Form: 471
Plant Name: Prairie Island 1

Event ID: 282/86-010 Power Level: 100%
Event Date: 12/12/86 Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Human

Cause 1: Human Related: Maintenance

Event Description: During troubleshooting of the turbine EHC, a multichannel event triggered recorder was being connected to the EHC cabinet. Incorrect use of this device caused a turbine trip/reactor trip.

*** BOP RELATED EVENT ***

Form: 475
Plant Name: Fort Calhoun 1

Event ID: 285/86-004 Power Level: 085%
Event Date: 08/01/86 Trip Type: Manual

BOP System: AC Power
BOP Subsystem: High Voltage
BOP Component: Bus duct

Cause 1: Component Failure: Bus duct
Cause 2: Component Failure: Bus duct insulation

Event Description: An operator noticed smoke coming from the plant isolated phase bus duct. During the following controlled shutdown, the condition worsened (the smoke intensified). The operators manually scrammed the reactor at 85% power. The arcing of the bus duct was due to a breakdown of the insulation on the bus duct.

*** BOP RELATED EVENT ***

Form: 477
Plant Name: Indian Point 3

Event ID: 286/84/005 Power Level: 90%
Event Date: 02/20/84 Trip Type: Auto

BOP System: Feedwater
BOP Subsystem: Feedwater Control
BOP Component: Solenoid valve

Cause 1: Component Failure: Solenoid valve

Event Description: A Steam/Feedwater flow mismatch caused a reactor trip. The flow mismatch was caused by closure of a Feedwater regulator valve due to failure of a trip solenoid. The solenoid failure was caused by water leakage into the solenoid terminal box.

*** BOP RELATED EVENT ***

Form: 893
Plant Name: Duane Arnold

Event ID: 331/86-017
Event Date: 06/13/86

Power Level: 5%
Trip Type: Manual

BOP System: Air
BOP Component: Air line

Cause 1: Component Failure: Air line
Cause 2: Environment: Contamination

Event Description: Dessicant material in instrument air flow lines caused a fluctuation in the position of Feedwater control valves, which caused a trip of Feedwater block valves. This led to a loss of Feedwater, which led to a manual reactor trip on low reactor water level.

*** BOP RELATED EVENT ***

Form: 896
Plant Name: Fitzpatrick

Event ID: 333/84-009
Event Date: 03/22/84

Power Level: 67%
Trip Type: Auto

BOP System: Feedwater
BOP Component: Pump

Cause 1: Component Failure: Pump

Event Description: Failure of a Feedwater pump bearing caused a loss of the Feedwater pump which led to a loss of Feedwater. This resulted in a reactor trip on low reactor water level.

*** BOP RELATED EVENT ***

Form: £97
Plant Name: Fitzpatrick

Event ID: 333/84-010 | Power Level: 25%
Event Date: 03/25/85 | Trip Type: Auto

BOP System: Feedwater
BOP Subsystem: Feedwater Control
BOP Component: Control oil

Cause 1: Component Failure: Control oil

Event Description: Control oil leakage resulted in a loss of Feedwater due to a Feedwater pump trip. This lead to a reactor trip on low reactor water level.

*** BOP RELATED EVENT ***

Form: 1074
Plant Name: Limerick

Event ID: 352/87-048 | Power Level: 090%
Event Date: 09/19/87 | Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Weld

Cause 1: Human Related: Maintenance

Event Description: A weld failure resulted in low EHC oil pressure which caused a turbine trip and reactor trip. Review revealed an inadequate weld.

*** BOP RELATED EVENT ***

Form: 1075
Plant Name: Hope Creek 1

Event ID: 354/86-034
Event Date: 07/12/86

Power Level: 003%
Trip Type: Manual

BOP System: Main Steam
BOP Component: Pressure transmitter

Cause 1: Component Failure: Pressure transmitter
Cause 2: Component Failure: Pressure transmitter

Event Description: Two erroneous high steam flow signals caused an MSIV closure. Operators elected to shutdown the plant. Cause of the failure of the 2 pressure transmitters was not determined.

*** BOP RELATED EVENT ***

Form: 1192
Plant Name: Arkansas Nuclear One - 2

Event ID: 368/87-008
Event Date: 11/14/87

Power Level: 100%
Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: T/G Instrumentation & Control
BOP Component: Human

Cause 1: Human Related: Maintenance

Event Description: Improper calibration settings in the turbine vibration trip system resulted in a spurious high vibration signal (in the turbine journal bearing vibration trip logic) that led to a turbine trip and a subsequent reactor trip.

*** BOP RELATED EVENT ***

Form: 1196
Plant Name: McGuire 1

Event ID: 369/84-024
Event Date: 08/21/84

Power Level: 100%
Trip Type: Auto

BOP System: AC Power
BOP Subsystem: High Voltage
BOP Component: Computer

Cause 1: Component Failure: Computer
Cause 2: Design Related

Event Description: After corrective maintenance, the restarted switchyard computer opened power circuit breakers, causing a reactor and turbine trip.

*** BOP RELATED EVENT ***

Form: 1316
Plant Name: Waterford 3

Event ID: 382/87-028
Event Date: 12/11/87

Power Level: 90%
Trip Type: Auto

BOP System: Main Steam
BOP Component: Solenoid valve

Cause 1: Component Failure: Solenoid valve

Event Description: During MSIV testing, one MSIV went partially shut due to a failed solenoid valve. This resulted in a reactor trip.

*** BOP RELATED EVENT ***

Form: 1317
Plant Name: Susquehanna 1

Event ID: 387/84-013
Event Date: 03/03/84

Power Level: 74%
Trip Type: Auto

BOP System: Turbine Generator
BOP Subsystem: Thrust Bearing Wear Detector
BOP Component: Thrust bearing wear detector

Cause 1: Spurious Signal

Event Description: During weekly preventive maintenance activities, the turbine tripped on a spurious trip of the TBWD pressure switches. The reactor tripped following the fast closure of the turbine control valves. The cause of the turbine trip was not determined and is considered to have been a spurious occurrence.

*** BOP RELATED EVENT ***

Form: 1764
Plant Name: Clinton 1

Event ID: 461/87-060
Event Date: 10/02/87

Power Level: 90%
Trip Type: Auto

BOP System: DC Power
BOP Component: Human

Cause 1: Human Related: Operations

Event Description: An operator incorrectly opened a crosstie breaker between 2 non-class 1E 125VDC distribution channels. The reactor tripped on a reactor high water level signal.

*** BOP RELATED EVENT ***

Form: 1768
Plant Name: Wolf Creek 1

Event ID: 482/85-039 Power Level: 006%
Event Date: 06/06/85 Trip Type: Auto

BOP System: Feedwater
BOP Subsystem: Steam Relief
BOP Component: Human

Cause 1: Human Related: Surveillance

Event Description: Level oscillations caused by a steam dump control system test resulted in a low SG level and a subsequent reactor trip.

*** BOP RELATED EVENT ***

Form: 1851
Plant Name: Palo Verde 1

Event ID: 528/86-020 Power Level: 60%
Event Date: 02/03/86 Trip Type: Auto

BOP System: Feedwater
BOP Subsystem: Feedwater Control
BOP Component: Circuit card

Cause 1: Component Failure: Circuit card

Event Description: A failed control board in the Feedwater control system resulted in a temporary inability of the operators to control Feedwater pump speed from the control room. Loss of manual Feedwater control led to a Low SG Level that subsequently tripped the reactor.

APPENDIX B

BOP Trips per Calendar Year by Plant

(Raw Data, 1984-1988)

Table B.1
BOP Trips By Year - Babcock and Wilcox Units

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
ARKANSAS NUCLEAR ONE 1	DEC 74	3	8	1	2	0	14
CRYSTAL RIVER 3	DEC 76	2	8	0	2	2	14
DAVIS BESSE	APR 77	2	4	1	4	2	13
OCONEE 1	FEB 73	2	4	2	0	1	9
OCONEE 2	OCT 73	0	3	3	2	1	9
OCONEE 3	JUL 74	3	2	2	0	2	9
RANCHO SECO	AUG 74	5	3	-	-	3	11
TMI 1	APR 74	-	1	4	3	2	10
	TOTALS	17	33	13	13	13	89
MATURE UNITS (OL BEFORE JAN 83)	AVG	2.43	4.13	1.86	1.86	1.62	

Table B.2
BOP Trips By Year - Combustion Engineering Units

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
ARKANSAS NUCLEAR ONE 2	JUL 78	5	4	1	2	0	12
CALVERT CLIFFS 1	JUL 74	4	6	2	5	3	20
CALVERT CLIFFS 2	AUG 76	1	1	3	4	2	11
FORT CALHOUN 1	MAY 73	0	0	1	0	0	[illegible]
MAINE YANKEE	JUN 73	6	8	6	2	3	25
MILLSTONE 2	AUG 75	2	0	4	5	0	11
PALISADES	OCT 72	1	2	2	4	0	9
PALO VERDE 1	DEC 84	-	5	6	2	4	17
PALO VERDE 2	DEC 85	-	-	6	2	1	9
PALO VERDE 3	JAN 88	-	-	-	-	0	0
SAN ONOFRE 2	SEP 82	1	4	3	3	0	11
SAN ONOFRE 3	SEP 83	3	2	3	2	0	10
ST. LUCIE 1	MAR 76	1	0	2	5	4	12
ST. LUCIE 2	APR 83	7	5	3	5	0	20
WATERFORD 3	DEC 84	-	19	2	5	1	27
	TOTALS	31	56	44	46	18	195
(9) MATURE PLANTS (OL BEFORE JAN 83)	AVG	2.33	2.78	2.67	3.33	1.33	
NEW PLANTS	AVG	5.00	7.75	4.00	3.20	1.00	

Table B.3
BOP Trips by Year - General Electric Units

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
BIG ROCK POINT	AUG 62	2	2	0	0	2	6
BROWNS FERRY 1	DEC 73	2	1	1	-	-	4
BROWNS FERRY 2	AUG 74	1	-	-	-	-	1
BROWNS FERRY 3	AUG 76	2	1	-	-	-	3
BRUNSWICK 1	NOV 76	3	1	5	1	1	11
BRUNSWICK 2	DEC 74	1	1	1	2	3	8
CLINTON 1	APR 87	-	-	-	7	2	9
COOPER	JAN 74	2	1	2	6	2	13
DRESDEN 2	DEC 69	2	[illegible]	3	5	0	14
DRESDEN 3	JAN 71	[illegible]	3	4	7	1	22
DUANE ARNOLD	FEB 74	3	0	1	0	1	5
FERMI 2	MAR 85	-	5	4	6	5	20
FITZPATRICK	OCT 74	3	5	2	3	0	13
GRAND GULF 1	JUL 82	7	[illegible]	3	2	2	27
HATCH 1	AUG 74	3	2	1	4	5	15
HATCH 2	JUN 78	2	2	5	3	6	18
HOPE CREEK 1	JUL 86	-	-	7	4	5	16
LA SALLE 1	APR 82	6	4	0	6	0	16
LA SALLE 2	DEC 83	8	0	4	1	0	13
LIMERICK	OCT 84	-	2	1	2	0	5
MILLSTONE 1	OCT 70	0	2	3	3	1	9

Table B.3
BOP Trips by Year - General Electric Units (Continued)

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
MONTICELLO	SEP 70	0	2	1	3	2	8
NINE MILE POINT 1	AUG 69	1	6	1	2	0	10
NINE MILE POINT 2	OCT 86	-	-	-	5	10	15
OYSTER CREEK	APR 69	2	4	3	1	1	11
PEACH BOTTOM 2	AUG 73	0	4	3	0	1	8
PEACH BOTTOM 3	JUL 74	2	1	8	2	0	13
PERRY 1	MAR 86	-	-	1	8	4	13
PILGRIM	JUN 72	1	2	3	-	0	6
QUAD CITIES 1	OCT 71	2	0	3	1	1	7
QUAD CITIES 2	APR 72	1	2	0	4	4	11
RIVER BEND 1	NOV 85	-	3	13	2	4	22
SHOREHAM			(1)				1
SUSQUEHANNA 1	JUL 82	4	2	0	1	2	9
SUSQUEHANNA 2	MAR 84	4	3	2	1	0	10
VERMONT YANKEE	FEB 73	2	0	0	3	3	8
WPPSS 2	DEC 83	20	3	5	2	0	30
	TOTALS	93	82	90	97	68	430
MATURE PLANTS (OL BEFORE JAN 83)	AVG	2.35	2.54	2.04	2.26	1.46	
NEW PLANTS	AVG	10.67	2.83	4.63	3.80	3.00	

Table B.4
BOP Trips by Year - Westinghouse Units

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
BEAVER VALLEY 1	JAN 76	3		1	3	2	14
BEAVER VALLEY 2	AUG 87	-	-	-	10	2	12
BRAIDWOOD 1	MAY 87	-	-	(1)	6	2	9
BRAIDWOOD 2	DEC 87	-	-	-	-	10	10
BYRON 1	FEB 85	-	14	2	2	2	20
BYRON 2	JAN 87	-	-	-	9	4	13
CALLAWAY 1	JUN 84	10	12	4	1	6	33
CATAWBA 1	JUN 85	-	9	4	5	0	18
CATAWBA 2	MAY 86	-	-	8	7	6	21
CONN YANKEE	JUN 67	1	2	5	1	1	10
COOK 1	OCT 74	3	0	5	2	0	10
COOK 2	DEC 77	5	4	4	5	0	18
DIABLO CANYON 1	SEP 81	4	5	2	4	3	18
DIABLO CANYON 2	AUG 85	-	7	9	3	2	21
FARLEY 1	JUN 77	1	3	2	3	1	10
FARLEY 2	OCT 80	2	3	2	0	0	7
GINNA	SEP 69	1	5	3	0	2	11
INDIAN POINT 2	SEP 73	5	8	6	0	4	23
INDIAN POINT 3	DEC 75	5	6	6	5	4	26
KEWAUNEE	DEC 73	4	6	2	2	2	16
McGUIRE 1	JUN 81	1	4	3	1	2	11
McGUIRE 2	MAR 83	10	7	4	4	2	27

Table B.4
BOP Trips by Year - Westinghouse Units (Continued)

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
MILLSTONE 3	JAN 86	-	-	11	7	4	22
NORTH ANNA 1	NOV 77	5	1	4	2	4	16
NORTH ANNA 2	AUG 80	2	2	3	0	0	7
POINT BEACH 1	OCT 70	0	1	2	0	0	3
POINT BEACH 2	MAY 72	0	1	1	0	1	3
PRAIRIE ISLAND 1	AUG 73	2	1	1	0	1	5
PRAIRIE ISLAND 2	OCT 74	0	0	1	0	0	1
ROBINSON 2	SEP 70	0	7	3	2	3	15
SALEM 1	APR 77	7	1	7	1	3	19
SALEM 2	AUG 81	7	7	9	3	5	31
SAN ONOFRE 1	MAR 67	0	1	1	1	0	3
SEQUOYAH 1	SEP 80	3	1	-	-	2	6
SEQUOYAH 2	SEP 81	4	3	-	-	4	11
SHEARON HARRIS 1	JAN 87	-	-	-	18	3	21
SOUTH TEXAS	MAR 88	-	-	-	-	3	3
SUMMER 1	AUG 82	8	3	5	3	0	19
SURRY 1	MAY 72	3	3	3	1	0	10
SURRY 2	JAN 73	9	1	3	1	3	17
TROJAN	NOV 75	4	3	2	3	0	12
TURKEY POINT 3	JUL 72	7	3	3	4	0	17
TURKEY POINT 4	APR 73	5	5	2	0	1	13
VOGTLE 1	JAN 87	-	-	-	14	7	21

Table B.4
BOP Trips by Year - Westinghouse Units (Continued)

UNIT	OL DATE	BOP TRIPS '84	'85	'86	'87	'88	TOTALS
WOLF CREEK 1	MAR 85	-	12	6	6	0	24
YANKEE ROWE	JUL 60	1	0	3	1	4	9
ZION 1	APR 73	5	3	1	1	4	14
ZION 2	NOV 73	4	1	3	1	2	11
	TOTALS	131	160	147	142	111	6[illegible]1
(33) MATURE PLANTS (OL BEFORE JAN 83)	AVG	3.36	3.00	2.97	1.52	1.76	
NEW PLANTS	AVG	10.00	10.20	6.00	7.08	3.53	

Table B.5
Summary Tables
Average BOP Trips per Calendar Year (1984 Through 1988)

MATURE UNITS (OL BEFORE JAN 83)	'84	'85	'86	'87	'88
B&W (8 UNITS)	2.43	4.13	1.86	1.86	1.62
CE (9 UNITS)	2.33	2.78	2.67	3.33	1.33
GE (26 UNITS)	2.35	2.54	2.04	2.26	1.46
W (33 UNITS)	3.36	3.00	2.97	1.52	1.76

NEW UNITS (OL BEFORE JAN 83)	'84	'85	'86	'87	'88
B&W (NONE)	-	-	-	-	-
CE (2 TO 6 UNITS)	5.00	7.75	4.00	3.20	1.00
GE (3 TO 11 UNITS)	10.67	2.83	4.63	3.80	3.00
W (2 TO 15 UNITS)	10.00	10.20	6.00	7.08	3.53

APPENDIX C

BOP Trips per Critical Year by Plant

(Normalized Data, 1984-1988)

Table C.1 BOP Trips per Critical Year - Babcock and Wilcox Plants

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Arkansas Nuclear One -1	3	.71	4.2	8	.80	10.0	1	.63	1.6	2	.90	2.2	0	.70	0.0	14	3.7
Crystal River 3	2	.95	2.1	8	.50	16.0	0	.42	0.0	2	.61	3.3	2	.85	2.4	14	4.2
Davis Besse	2	.63	3.2	4	.32	12.5	1	.02	50.0	4	.85	4.7	2	.24	8.3	13	6.3
Oconee 1	2	.85	2.3	4	.96	4.2	2	.68	2.9	[illegible]	.79	0.0	1	1.00	1.0	9	2.1
Oconee 2	0	1.00	0.0	3	.77	3.9	3	.83	3.6	2	.98	2.0	1	.80	1.3	9	2.1
Oconee 3	3	.74	4 1	2	.70	2.9	2	.89	2.2	0	.70	0.0	2	.83	2.4	9	2.3
Rancho Seco	5	.61	8.2	3	.33	9.1	0	.00	---	0	.00	---	3	.63	4.8	11	7.0
Three Mile Island 1	0	.00	---	1	.24	4.2	4	.72	5.6	3	.73	4.1	2	.77	2.6	10	4.1
TOTAL	17	5.49	3.1	33	4.62	7.1	13	4.19	3.1	13	5.56	2.3	13	5.82	2.2	89	3.5

Table C.2 BOP Trips per Critical Year - Combustion Engineering Plants

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Arkansas Nuclear One -2	5	.87	5.7	4	.73	5.5	1	.73	1.4	2	.88	3.6	0	.69	0.0	12	3.1
Calvert Cliffs 1		.86	4.6	6	.61	9.8	2	.79	2.5	5	.76	6.6	3	.73	4.1	20	5.3
Calvert Cliffs 2	1	.75	1.3	1	.79	1.3	3	.96	3.1	4	.68	5.9	2	.89	2.2	11	2.7
Fort Calhoun	0	.61	0.0	0	.74	0.0	1	.97	1.0	0	.75	0.0	0	.74	0.0	1	0.3
Maine Yankee	6	.76	7.9	8	.80	10.0	6	.89	6.7	2	.65	3.1	3	.79	3.8	25	6.4
Millstone 2	2	[illegible]	2.0	0	.51	0.0	4	.75	5.3	5	.94	5.4	0	.79	0.0	11	2.8
Palisades	1	.18	5.6	2	.86	2.3	2	.17	11.8	4	.48	8.3	0	.57	0.0	9	4.0
San Onofre 2	1	.60	1.7	4	.60	6.7	3	.74	4.1	3	.71	4.2	0	.94	0.0	11	3.1
St Lucie 1	1	.63	1.6	0	.81	0.0	2	.96	2.1	5	.80	6.3	4	.86	4.7	12	2.9
TOTAL	21	6.24	3.4	25	6.45	3.9	24	6.96	3.4	30	6.65	4.5	12	7.00	1.7	112	3.4

Table C.2 BOP Trips per Critical Year - Combustion Engineering Plants (Con't)

NEW PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Palo Verde 1	0	.00	---	5	.28	17.8	6	.58	10.3	2	.52	3.8	4	.66	6.1	17	8.3
Palo Verde 2	0	.00	---	0	.00	---	6	.25	24.0	2	.80	2.5	1	.65	1.[illegible]	9	5.3
Palo Verde 3	0	.00	---	0	.00	---	0	.00	---	0	.08	0.0	0	.93	0.0	0	0.0
St Lucie 2	7	.84	8.3	5	.85	5.9	3	.84	3.6	5	.84	6.0	0	1.00	0.0	20	4.6
San Onofre 3	3	.50	6.0	2	.55	3.6	3	.84	3.6	2	.81	2.5	0	.68	0.0	10	2.9
Waterford 3	0	.00	---	19	.21	90.4	2	.80	2.5	5	.82	6.1	1	.75	1.3	27	10.4
TOTAL	10	1.34	7.5	31	1.89	16.4	[illegible]	3.31	6.0	16	3.87	4.1	6	4.67	1.3	83	5.5

Table C.3 BOP Trips per Critical Year - General Electric Plants

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Big Rock Point	2	.79	2.5	2	.75	2.7	0	.96	0.0	0	.71	0.0	2	.73	2.7	6	1.5
Browns Ferry 1	2	.92	2.2	1	.19	5.3	1	.60	---	0	.00	---	0	.00	---	4	3.6
Browns Ferry 2	1	.67	1.5	0	.00	---	0	.00	---	0	.00	---	9	.00	---	1	1.5
Browns Ferry 3		.08	25.0	1	.17	5.9	0	.00	---	0	.00	---	0	.00	---	3	12.0
Brunswick 1		.80	3.8	1	.39	2.6	5	.95	5.3	1	.66	1.5	1	.76	1.3	11	3.1
Brunswick 2	1	.30	3.3	1	.81	1.2	1	.48	2.1	2	.95	2.1	3	.64	4.7	8	2.5
Cooper	2	.68	2.9	1	.23	4.3	2	.75	2.7	6	.96	6.3	2	.68	2.9	13	3.9
Dresden 2	2	.74	2.7	4	.57	7.0	3	.81	3.7	5	.66	7.6	0	.79	0.0	14	3.9
Dresden 3	7	.44	15.9	3	.77	3.9	4	.31	12.9	7	.82	8.5	1	.72	1.4	22	7.2
Duane Arnold	3	.75	4.0	0	.54	0.0	1	.84	1.2	0	.65	0.0	1	.75	1.3	5	1.4
Fitzpatrick	3	.81	3.7	5	.66	7.6	2	.92	2.2	3	.70	4.3	0	.69	0.0	13	3.4
Grand Gulf 1	7	.11	63.6	13	.33	39.4	3	.64	4.7	2	.82	2.4	2	.97	2.1	27	9.4
Hatch 1	3	.64	4.7	2	.79	2.5	1	.63	1.6	4	.82	4.9	5	.68	7.4	15	4.2
Hatch 2	2	.35	5.7	2	.84	2.4	5	.74	6.8	3	.97	3.1	6	.72	8.3	13	5.0
LaSalle 1	6	.71	8.5	4	.66	6.1	0	.27	0.0	6	.64	9.4	0	.68	0.0	16	5.4
Millstone 1	0	.80	0.0	2	.84	2.4	3	.94	3.2	3	.80	3.8	1	.99	1.0	9	2.1
Monticello	0	.09	0.0	2	.93	2.2	1	.80	1.3	3	.82	3.7	2	1.00	2.0	8	2.2
Nine Mile Point 1	1	.73	1.4	6	.97	6.2	1	.66	1.5	2	.93	2.2	0	.00	---	10	3.0
Oyster Creek	2	.19	10.5	4	.78	5.1	3	.27	11.1	1	.64	1.6	1	.66	1.5	11	4.3
Peach Bottom 2	0	.29	0.0	4	.33	12.1	3	.83	3.6	0	.20	0.0	1	.00	---	8	4.8
Peach Bottom 3	2	.88	2.3	1	.46	2.2	8	.68	11.8	2	.21	9.5	0	.00	---	13	5.8
Pilgrim	1	.02	50.0	2	.93	2.2	3	.20	15.0	0	.00	---	0	.00	0.0	6	5.2
Quad Cities 1	2	.54	3.7	0	.95	0.0	3	.70	4.3	1	.71	1.4	1	.97	1.0	7	1.8
Quad Cities 2	1	.80	1.3	2	.73	2.7	0	.74	0.0	4	.79	5.1	4	.72	4.2	11	2.9
Susquehanna 1	4	.75	5.3	2	.64	3.1	0	.71	0.0	1	.74	1.4	2	.94	2.1	9	2.4
Vermont Yankee	2	.81	2.5	0	.72	0.0	0	.50	0.0	3	.84	3.6	3	.96	3.1	8	2.1
TOTAL	61	14.69	4.2	65	15.98	4.1	53	15.33	3.[illegible]	59	16.04	3.7	38	15.05	2.5	276	3.6

Table C.3 BOP Trips per Critical Year - General Electric Plants (Con't)

NEW PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Clinton 1	0	.00	---	0	.00	---	0	.00	---	7	.10	70.0	2	.84	2.4	9	9.6
Fermi 2	0	.00	---	5	.00	---	4	.10	40.0	6	.59	10.2	5	.57	8.8	20	15.9
Hope Creek 1	0	.00	---	0	.00	---	7	.23	30.4	4	.86	4.7	5	.81	6.2	16	8.4
LaSalle 2	8	.18	44.4	0	.43	0.0	4	.76	5.3	1	.55	1.8	0	.76	0.0	13	4.9
Limerick 1	0	.00	---	2	.39	5.1	1	.77	1.3	2	.70	2.9	0	.96	0.0	5	1.8
Nine Mile Point 2	0	.00	---	0	.00	---	0	.00	---	5	.19	26.3	10	.34	29.4	15	28.3
Perry 1	0	.00	---	0	.00	---	1	.00	---	8	.09	88.9	4	.79	5.1	13	14.8
River Bend 1	0	.00	---	3	.00	---	13	.55	23.6	2	.68	2.9	4	.94	4.3	22	10.1
Shoreham	0	.00	---	1	.00	---	0	.00	---	0	.00	---	0	.00	---	1	---
Susquehanna 2	4	.24	16.7	3	.81	3.7	2	.68	2.9	1	.97	1.0	0	.70	0.0	10	2.9
WPPSS 2	20	.05	400.	3	.79	3.8	5	.73	6.8	2	.71	2.8	0	.72	0.0	30	10.0
TOTAL	32	.47	68.1	17	2.42	7.0	37	3.82	9.7	38	5.44	7.0	30	7.43	4.0	154	7.9

Table C.4 BOP Trips per Critical Year - Westinghouse Plants

MATURE PLANTS

Plant Name	84			85			86			87			88				
	TRIPS	CRIT. YEARS	TRIPS /CY	TRIPS	CRIT. YEARS	TRIPS /CY	TRIPS	CRIT. YEARS	TRIPS /CY	TRIPS	CRIT. YEARS	TRIPS /CY	TRIPS	CRIT. YEARS	TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Beaver Valley 1	3	.74	4.1	5	.94	5.3	1	.71	1.4	3	.84	3.6	2	.80	2.5	14	3.5
Connecticut Yankee	1	.74	1.4	2	.99	2.0	5	.58	8.6	1	.54	1.9	1	.70	1.4	10	2.8
Cook 1	3	.92	3.3	0	.30	0.0	5	.86	5.8	2	.69	2.9	0	.96	0.0	10	2.7
Cook 2	5	.60	8.3	4	.68	5.9	4	.63	6.3	5	.72	6.9	0	.31	0.0	18	6.1
Diablo Canyon 1	4	.11	36.4	5	.60	8.3	2	.68	2.9	4	.97	4.1	3	.65	4.6	18	6.0
Farley 1	1	.80	1.3	3	.86	3.5	2	.83	2.4	3	.95	3.2	1	.85	1.2	10	2.3
Farley 2	2	.95	2.1	3	.79	3.8	2	.86	2.3	0	.75	0.0	0	1.00	0.0	7	1.6
Ginna	1	.78	1.3	5	.89	5.6	3	.88	3.4	0	.91	0.0	2	.87	2.3	11	2.5
Indian Point 2	5	.54	9.3	8	.97	8.2	6	.58	10.3	0	.72	0.0	4	.85	4.7	23	6.3
Indian Point 3	5	.79	6.3	6	.67	8.9	6	.75	8.0	5	.63	7.9	4	.83	4.8	26	7.1
Kewaunee	4	.86	4.7	6	.83	7.2	2	.87	2.3	2	.90	2.2	2	.88	2.3	16	3.6
McGuire 1	1	.69	1.4	4	.78	5.1	3	.57	5.3	1	.78	1.3	2	.77	2.6	11	3.1
North Anna 1	5	.54	9.3	1	.79	1.3	4	.86	4.7	2	.52	3.8	4	.91	4.4	16	4.4
North Anna 2	2	.70	2.8	2	.97	2.1	3	.83	3.6	0	.78	0.0	0	.99	0.0	7	1.6
Point Beach 1	0	.73	0.0	1	.80	1.3	2	.90	2.2	0	.84	0.0	0	.89	0.0	3	0.8
Point Beach 2	0	.86	0.0	1	.86	1.2	1	.83	1.2	0	.87	0.0	1	.88	1.1	3	0.7
Prairie Island 1	2	.95	2.1	1	.84	1.2	1	.90	1.1	0	.83	0.0	1	.89	1.1	5	1.1
Prairie Island 2	0	.89	0.0	0	.85	0.0	1	.91	1.1	0	1.00	0.0	0	.89	0.0	1	0.2
Robinson 2	0	.07	0.0	7	.90	7.8	3	.81	.37	2	.73	2.7	3	.66	4.5	15	4.7

Table C.4 BOP Trips per Critical Year - Westinghouse Plants (Con't)

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	35 TRIPS	35 CRIT. YEARS	35 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Salem 1	7	.30	23.3	1	.95	1.1	7	.81	8.6	1	.73	1.4	3	.79	3.8	19	5.3
Salem 2	7	.39	17.9	7	.60	11.7	9	.64	14.1	[illegible]	.73	4.1	5	.68	7.4	31	10.2
San Onofre 1	0	.10	0.0	1	.77	1.3	1	.34	2.9	1	.84	1.2	0	.32	0.0	3	1.3
Sequoyah 1	3	.71	4.2	1	.43	2.3	0	.00	---	0	.00	---	2	.01	200.	6	5.2
Sequoyah 2	4	.72	5.6	3	.60	5.0	0	.00	---	0	.00	---	4	.59	6.8	11	5.8
Summer 1	8	.63	12.7	3	.74	4.1	5	.96	5.2	3	.71	4.2	0	.69	0.0	19	5.1
Surry 1	3	.60	5.0	3	.91	3.3	3	.71	4.2	1	.71	1.4	0	.43	0.0	10	3.0
Surry 2	9	.85	10.6	1	.68	1.5	3	.70	4.3	1	.75	1.3	3	.57	5.3	17	4.8
Trojan	4	.56	7.1	3	.78	3.8	2	.81	2.5	3	.54	5.6	0	.67	0.0	12	3.6
Turkey Point 3	7	.84	8.3	3	.62	4.8	3	.80	3.8	4	.22	18.2	0	.62	0.0	17	5.5
Turkey Point 4	5	.58	8.3	5	.90	5.6	2	.35	5.7	0	.51	0.0	1	.57	1.8	13	4.4
Yankee Rowe	1	.73	1.4	0	.87	0.0	3	.95	3.2	1	.83	1.2	4	.85	4.7	9	2.1
Zion 1	5	.72	6.9	3	.61	4.9	1	.63	1.6	1	.79	1.3	4	.77	5.2	14	4.0
Zion 2	4	.72	5.6	1	.67	1.5	3	.89	3.4	1	.64	1.6	2	.80	2.5	11	3.0
TOTAL	111	21.71	5.1	99	25.44	3.9	98	23.43	4.2	50	22.97	2.2	58	23.99	2.4	416	3.5

Table C.4 BOP Trips per Critical Year - Westinghouse Plants (Con't)

NEW PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Beaver Valley 2	0	.00	---	0	.00	---	0	.00	---	10	.11	90.1	2	.94	2.1	12	11.4
Braidwood 1	0	.00	---	0	.00	---	1	.00	---	6	.35	17.1	2	.39	5.1	9	12.2
Braidwood 2	0	.00	---	0	.00	---	0	.00	---	0	.00	---	10	.17	58.8	10	58.8
Byron 1	0	.00	---	14	.15	93.3	2	.89	2.2	2	.71	2.8	2	.74	2.7	20	8.0
Byron 2	0	.00	---	0	.00	---	0	.00	---	9	.27	33.3	4	.99	4.0	13	10.3
Callaway	10	.03	333.	12	.93	12.9	4	.83	4.8	1	.71	1.4	6	.93	9.7	33	9.6
Catawba 1	0	.00	---	9	.41	21.9	4	.62	6.5	5	.64	7.2	0	.80	0.0	18	7.1
Catawba 2	0	.00	---	0	.00	---	8	.16	50.0	7	.97	7.2	6	.74	8.1	21	12.2
Diablo Canyon 2	0	.00	---	7	.16	43.8	9	.78	11.5	3	.69	4.3	2	.70	2.9	21	9.0
McGuire 2	10	.70	14.3	7	.63	11.1	4	.66	6.1	4	.80	5.0	2	.83	2.4	27	7.5
Millstone 3	0	.00	---	0	.00	---	11	.62	17.7	7	.72	9.7	4	.82	4.9	22	10.2
Shearon Harris	0	.00	---	0	.00	---	0	.00	---	18	.51	35.3	3	.75	4.0	21	16.7
South Texas 1	0	.00	---	0	.00	---	0	.00	---	0	.00	---	3	.28	10.7	3	10.7
Vogtle 1	0	.00	---	0	.00	---	0	.00	---	14	.46	30.4	7	.76	9.0	21	16.9
Wolf Creek	0	.00	---	12	.32	37.5	6	.74	8.1	6	.70	8.6	0	.70	0.0	24	9.7
TOTAL	20	.73	27.4	61	2.60	23.5	49	5.30	9.2	92	7.54	12.2	53	10.56	5.0	275	10.3

Table C.5 BOP Trips per Critical Year - All Plants

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Arkansas Nuclear One -1	3	.71	4.2	8	.80	10.0	1	.63	1.6	2	.90	2.2	0	.70	0.0	14	3.7
Arkansas Nuclear One -2	5	.87	5.7	4	.73	5.5	1	.73	1.4	2	.88	3.6	0	.69	0.0	12	3.1
Beaver Valley 1	3	.74	4.1	5	.94	5.3	1	.71	1.4	3	.84	3.6	2	.80	2.5	14	3.5
Big Rock Point	2	.79	2.5	2	.75	2.7	0	.96	0.0	0	.71	0.0	2	.73	2.7	6	1.5
Browns Ferry 1	2	.92	2.2	1	.19	5.3	1	.00	---	0	.00	---	0	.00	---	4	[illegible]
Browns Ferry 2	1	.67	1.5	0	.00	---	0	.00	---	0	.00	---	0	.00	---	1	1.5
Browns Ferry 3	2	.08	25.0	1	.17	5.9	0	.00	---	0	.00	---	0	.00	---	3	12.0
Brunswick 1	3	.80	3.8	1	.39	2.6	5	.95	5.3	1	.66	1.5	1	.76	1.3	11	3.1
Brunswick 2	1	.30	3.3	1	.81	1.2	1	.48	2.1	2	.95	2.1	3	.64	4.7	8	2.5
Calvert Cliffs 1	4	.86	4.6	6	.61	9.8	2	.79	2.5	5	.76	6.6	3	.73	4.1	20	5.3
Calvert Cliffs 2	1	.75	1.3	1	.79	1.3	3	.96	3.1	4	.68	5.9	2	.89	2.2	11	2.7
Connecticut Yankee	1	.74	1.4	2	.99	2.0	5	.58	8.6	1	.54	1.9	1	.70	1.4	10	2.8
Cook 1	3	.92	3.3	0	.30	0.0	5	.86	5.8	2	.69	2.9	0	[illegible]	0.0	10	2.7
Cook 2	5	.60	8.3	4	.68	5.9	4	.63	6.3	5	.72	6.9	0	.31	0.0	18	6.1
Cooper	2	.68	2.9	1	.23	4.3	2	.75	2.7	6	.96	6.3	2	.68	2.9	13	3.9
Crystal River 3	2	.95	2.1	8	.50	16.0	0	.42	0.0	2	.61	3.3	2	.85	2.4	14	4.2
Davis Besse	2	.63	3.2	4	.32	12.5	1	.02	50.0	4	.85	4.7	2	.24	8.3	13	6.3
Diablo Canyon 1	4	.11	36.4	5	.60	8.3	2	.68	2.9	4	.97	4.1	3	.65	4.6	18	6.0
Dresden 2	2	.74	2.7	4	.57	7.0	3	.81	3.7	5	.66	7.6	0	.79	0.0	14	3.9
Dresden 3	7	.44	15.9	3	.77	3.9	4	.31	12.9	7	.82	8.5	1	.72	1.4	22	7.2
Duane Arnold	3	.75	4.0	0	.54	0.0	1	.84	1.2	[illegible]	.65	0.0	1	.75	1.3	5	1.4
Farley 1	1	.80	1.3	3	.86	3.5	2	.83	2.4	3	.95	3.2	1	.85	1.2	10	2.3
Farley 2	2	.95	2.1	3	.79	3.8	2	.86	2.3	0	.75	0.0	0	1.00	0.0	7	1.6
Fitzpatrick	3	.81	3.7	5	.66	7.6	2	.92	2.2	3	.70	4.3	0	.69	0.0	13	3.4
Fort Calhoun	0	.61	0.0	0	.74	0.0	1	.97	1.0	0	.75	0.0	0	.74	0.0	1	0.3
Ginna	1	.78	1.3	5	.89	5.6	3	.88	3.4	0	.91	0.0	2	.87	2.	11	2.5

Table C.5 BOP Trips per Critical Year - All Years (Con't)

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Grand Gulf 1	7	.11	63.6	13	.33	39.4	3	.64	4.7	2	.82	2.4	2	.97	2.1	27	9.4
Hatch 1	3	.64	4.7	2	.79	2.5	1	.63	1.6	4	.82	4 9	5	.68	7.4	15	4.2
Hatch 2	2	.35	5.7	2	.84	2.4	5	.74	6.8	3	.97	3.1	6	.72	8.3	18	5.0
Indian Point 2	5	.54	9.3	8	.97	8.2	6	.58	10.3	0	.72	0.0	4	.85	4.7	23	6.3
Indian Point 3	5	.79	6.3	6	.67	8.9	6	.75	8.0	5	.63	7.9	4	.83	4.8	26	7.1
Kewaunee	4	.86	4.7	6	.83	7.2	2	.87	2.3	2	.90	2.2	2	.88	2.3	16	3.6
LaSalle 1	6	.71	8.5	4	.66	6.1	0	.27	0.0	6	.64	9.4	0	.68	0.0	16	5.4
Maine Yankee	6	.76	7.9	8	.80	10.0	6	.89	6.7	2	.65	3.1	3	.79	3.8	25	6.4
McGuire 1	1	.69	1.4	4	.78	5.1	3	.57	5.3	1	.78	1.3	2	.77	2.6	11	3.1
Millstone 1	0	.80	0.0	2	.84	2.4	3	.94	3.2	3	.80	3.8	1	.99	1.0	9	2.1
Millstone 2	2	.98	2.0	0	.51	0.0	4	.75	5.3	5	.94	5.4	0	.79	0.0	11	2.8
Monticello	0	.09	0.0	2	.93	2.2	1	.80	1.3	3	.82	3.7	2	1.00	2.0	8	2.2
Nine Mile Point 1	1	.73	1.4	6	.97	6.2	1	.66	1.5	2	.93	2.2	0	.00	---	10	3.0
North Anna 1	5	.54	9.3	1	.79	1.3	4	.86	4.7	2	.52	3.8	4	.91	4.4	16	4.4
North Anna 2	2	.70	2.8	2	.97	2.1	3	.83	3.6	0	.78	0.0	0	.99	0.0	7	1.6
Oconee 1	2	.85	2.3	4	.96	4.2	2	.68	2.9	0	.79	0.0	1	1.00	1.0	9	2.1
Oconee 2	0	1.00	0.0	3	.77	3.9	3	.83	3.6	2	.98	2.0	1	.80	1.3	9	2.1
Oconee 3	3	.74	4.1	2	.70	2.9	2	.89	2.2	0	.70	0.0	2	.83	2.4	9	2.3
Oyster Creek	2	.19	10.5	4	.78	5.1	3	.27	11.1	1	.64	1.6	1	.66	1.5	11	4.3
Palisades	1	.18	5.6	2	.86	2.3	2	.17	11.8	4	.48	8.3	0	.57	0.0	9	4.0
Peach Bottom 2	0	.29	0.0	4	.33	12.1	3	.83	3.6	0	.20	0.0	1	.00	---	8	4.8
Peach Bottom 3	2	.88	2.3	1	.46	2.2	8	.68	11.8	2	.21	9.5	0	.00	---	13	5.8
Pilgrim	1	.02	50.0	2	.93	2.2	3	.20	15.0	0	.00	---	0	.00	0.0	6	5.2
Point Beach 1	0	.73	0.0	1	.80	1.3	2	.90	2.2	0	.84	0.0	0	.89	0.0	3	0.8
Point Beach 2	0	.86	0.0	1	.86	1.2	1	.83	1.2	0	.87	0.0	1	.88	1.1	3	0.7

Table C.5 BOP Trips per Critical Year - All Plants (Con't)

MATURE PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Prairie Island 1	2	.95	2.1	1	.84	1.2	1	.90	1.1	0	.83	0.0	1	.89	1.1	5	1.1
Prairie Island 2	0	.89	0.0	0	.85	0.0	1	.91	1.1	0	1.00	0.0	0	.89	0.0	1	0.2
Quad Cities 1	2	.54	3.7	0	.95	0.0	3	.70	4.3	1	.71	1.4	1	.97	1.0	7	1.8
Quad Cities 2	1	.80	1.3	2	.73	2.7	0	.74	0.0	4	.79	5.1	4	.72	4.2	11	2.9
Rancho Seco	5	.61	8.2	3	.33	9.1	0	.00	---	0	.00	---	3	.63	4.8	11	7.0
Robinson 2	0	.07	0.0	7	.90	7.8	3	.81	.37	2	.73	2.7	3	.66	4.5	15	4.7
St Lucie 1	1	.63	1.6	0	.81	0.0	2	.96	2.1	5	.80	6.3	4	.86	4.7	12	2.9
Salem 1	7	.30	23.3	1	.95	1.1	7	.81	8.6	1	.73	1.4	3	.79	3.8	19	5.3
Salem 2	7	.39	17.9	7	.60	11.7	9	.64	14.1	3	.73	4.1	5	.68	7.4	31	10.2
San Onofre 1	0	.10	0.0	1	.77	1.3	1	.34	2.9	1	.84	1.2	0	.32	[illegible]	3	1.3
San Onofre 2	1	.60	1.7	4	.60	6.7	3	.74	4.1	3	.71	4.2	0	.94	0.0	11	3.1
Sequoyah 1	3	.71	4.2	1	.43	2.3	0	.00	---	0	.00	---	2	.01	200.	6	5.2
Sequoyah 2	4	.72	5.6	3	.60	5.0	0	.00	---	0	.00	---	4	.59	6.8	11	5.8
Summer 1	8	.63	12.7	3	.74	4.1	5	.96	5.2	3	.71	4.2	0	.66	0.0	19	5.1
Surry 1	3	.60	5.0	3	.91	3.3	3	.71	4.2	1	.71	1.4	0	.43	0.0	10	3.0
Surry 2	9	.85	10.6	1	.68	1.5	3	.70	4.3	1	.75	1.3	3	.57	5.3	17	4.8
Susquehanna 1	4	.75	5.3	2	.64	3.1	0	.71	0.0	1	.74	1.4	2	.94	2.1	9	2.4
Three Mile Island 1	0	.00	---	1	.24	4.2	4	.72	5.6	3	.73	4.1	2	.77	2.6	10	4.1
Trojan	4	.56	7.1	3	.78	3.8	2	.81	2.5	3	.54	5.6	0	.67	0.0	12	3.6
Turkey Point 3	7	.84	8.3	3	.62	4.8	3	.80	3.8	4	.22	18.2	0	.62	0.0	17	5.5
Turkey Point 4	5	.58	8.3	5	.90	5.6	2	.35	5.7	0	.51	0.0	1	.57	1.8	13	4.4
Vermont Yankee	2	.81	2.5	0	.72	0.0	0	.50	0.0	3	.84	3.6	3	.96	3.1	8	2.1
Yankee Rowe	1	.73	1.4	0	.87	0.0	3	.95	3.2	1	.83	1.2	4	.85	4.7	9	2.1
Zion 1	5	.72	6.9	3	.61	4.9	1	.63	1.6	1	.79	1.3	4	.77	5.2	14	4.0
Zion 2	4	.72	5.6	1	.67	1.5	3	.89	3.4	1	.64	1.6	2	.80	2.5	11	3.0

Table C.5 BOP Trips per Critical Year - All Plants (Con't)
NEW PLANTS

Plant Name	84 TRIPS	84 CRIT. YEARS	84 TRIPS /CY	85 TRIPS	85 CRIT. YEARS	85 TRIPS /CY	86 TRIPS	86 CRIT. YEARS	86 TRIPS /CY	87 TRIPS	87 CRIT. YEARS	87 TRIPS /CY	88 TRIPS	88 CRIT. YEARS	88 TRIPS /CY	TOTAL TRIPS	5 YR AVE.
Beaver Valley 2	0	.00	---	0	.00	---	0	.00	---	10	.11	90.1	2	.94	2.1	12	11.4
Braidwood 1	0	.00	---	0	.00	---	1	.00	---	6	.35	17.1	2	.39	5.1	9	12.2
Braidwood 2	0	.00	---	0	.00	---	0	.00	---	0	.00	---	10	.17	58.8	10	58.8
Byron 1	0	.00	---	14	.15	93.3	2	.89	2.2	2	.71	2.8	2	.74	2.7	20	8.0
Byron 2	0	.00	---	0	.00	---	0	.00	---	9	.27	33.3	4	.99	4.0	13	10.3
Callaway	10	.03	333.	12	.93	12.9	4	.83	4.8	1	.71	1.4	6	.93	9.7	33	9.6
Catawba 1	0	.00	---	9	.41	21.9	4	.62	6.5	5	.69	7.2	0	.80	0.0	18	7.1
Catawba 2	0	.00	---	0	.00	---	8	.16	50.0	7	.82	7.2	6	.74	8.1	21	12.2
Clinton 1	0	.00	---	0	.00	---	0	.00	---	7	.10	70.0	2	.84	2.4	9	9.6
Diablo Canyon 2	0	.00	---	7	.16	43.8	9	.78	11.5	3	.69	4.3	2	.70	2.9	21	9.0
Fermi 2	0	.00	---	5	.00	---	4	.10	40.0	6	.59	10.2	5	.57	8.8	20	15.9
Hope Creek 1	0	.00	---	0	.00	---	[illegible]	.23	30.4	4	.86	4.7	5	.81	6.2	16	8.4
LaSalle 2	8	.18	44.4	0	.43	0.0	4	.76	5.3	1	.55	1.8	0	.76	0.0	13	4.9
Limerick 1	0	.00	---	2	.39	5.1	1	.77	1.3	2	.70	2.9	0	.96	0.0	5	1.8
McGuire 2	10	.70	14.3	7	.63	11.1	4	.66	6.1	4	.80	5.0	2	.83	2.4	27	7.5
Millstone 3	0	.00	---	0	.00	---	11	.62	17.7	7	.72	9.7	4	.82	4.9	22	10.2
Nine Mile Point 2	0	.00	---	0	.00	---	0	.00	---	5	.19	26.3	10	.34	29.4	15	28.3
Palo Verde 1	0	.00	---	5	.28	17.8	6	.58	10.3	2	.52	3.8	4	.66	6.1	17	8.3
Palo Verde 2	0	.00	---	0	.00	---	6	.25	24.0	2	.80	2.5	1	.65	1.5	9	5.3
Palo Verde 3	0	.00	---	0	.00	---	0	.00	---	0	.08	0.0	0	.93	0.0	0	0.0
Perry 1	0	.00	---	0	.00	---	1	.00	---	8	.09	88.9	4	.79	5.1	13	14.8
River Bend 1	0	.00	---	3	.00	---	13	.55	23.6	2	.68	2.9	4	.94	4.3	22	10.1
St Lucie 2	7	.84	8.3	5	.85	5.9	3	.84	3.6	5	.84	6.0	0	1.00	[illegible]	20	4.6
San Onofre 3	3	.50	6.0	2	.55	3.6	3	.84	3.6	2	.81	2.5	0	.68	[illegible]	10	2.9
Shearon Harris	0	.00	---	0	.00	---	[illegible]	.00	---	18	.51	35.3	3	.75	4.0	21	16.7
Shoreham	0	.00	---	1	.00	---	[illegible]	.00	---	0	.00	---	0	.00	---	1	---
South Texas 1	0	.00	---	0	.00	---	0	.00	---	0	.00	---	3	.28	10.7	3	10.7
Susquehanna 2	4	.24	16.7	3	.81	3.7	2	.68	2.9	1	.97	1.0	0	.70	0.0	10	2.9
Vogtle 1	0	.00	---	0	.00	---	0	.00	---	14	.46	30.4	7	.78	9.0	21	16.9
WPPSS 2	20	.05	400.	3	.79	3.8	5	.73	6.8	2	.71	2.8	0	.72	0.0	30	10.0
Waterford 3	0	.00	---	19	.21	90.4	2	.80	2.5	5	.82	6.1	1	.75	1.3	27	10.4
Wolf Creek	0	.00	---	12	.32	37.5	6	.74	8.1	6	.70	8.6	0	.70	0.0	24	9.7

APPENDIX D

BOP Trips per Critical Year by Plant

(Cumulative Average, 1984-1988)

Table D.1

List of BOP Trips by Plant per Critical Year: Cumulative Average for B&W Units

Plant Name	OL Date	Critical Years	BOP Trips 1/84 - 12/88	5-Yr Average BOP Trips per Critical Year
Arkansas Nuclear One - 1	12/01/74	3.70	14	3.78
Crystal River 3	12/03/76	3.33	14	4.20
Davis Besse	04/22/77	2.06	13	6.31
Oconee 1	02/06/73	4.28	9	2.10
Oconee 2	10/06/73	4.38	9	2.05
Oconee 3	07/19/74	3.86	9	2.33
Rancho Seco	08/16/74	1.57	11	7.01
Three Mile Island 1	04/19/74	2.46	10	4.07
*** Subtotal ***	8 Units		89	31.86

All plants*: Average= 3.98 Sigma= 1.7548

* All B&W plants are in the "Mature" category, OL before 1/1/83

Table D.2

List of BOP Trips by Plant per Critical Year: Cumulative Average for CE Units

Plant Name	OL Date	Critical Years	BOP Trips 1/84 - 12/88	5-Yr Average BOP Trips per Critical Year
Arkansas Nuclear One - 2	07/18/78	3.89	12	3.08
Calvert Cliffs 1	07/31/74	3.74	20	5.35
Calvert Cliffs 2	08/13/76	4.[illegible]	11	2.70
Fort Calhoun 1	05/24/73	3.82	1	0.26
Maine Yankee	06/01/73	3.90	25	6.41
Millstone 2	08/01/75	3.97	11	2.77
Palisades	10/01/72	2.25	9	4.00
Palo Verde 1	12/31/84	2.04	17	8.33
Palo Verde 2	12/09/85	1.71	9	5.26
San Onofre 2	09/07/82	3.59	11	3.06
San Onofre 3	09/16/83	3.38	10	2.96
St. Lucie 1	03/01/76	4.06	12	2.[illegible]
St. Lucie 2	04/06/83	4.37	20	4.58
Waterford 3	12/18/84	2.59	27	10.42
*** Subtotal ***	14 Units		195	62.15

All Plants:	Average=	4.44	Sigma=	2.5069
Mature Plants*:	Average=	3.40	Sigma=	1.64
New Plants**:	Average=	6.31	Sigma=	2 69

* 9 plants with OL before 1/183

** 5 plants with OL after 1/1/83

Table D.3

BOP Trips by Plant per Critical Year: Cumulative Average for GE Units

Plant Name	OL Date	Critical Years	BOP Trips 1/84 - 12/88	5-Yr Average BOP Trips per Critical Year
Big Rock Point	08/30/62	3.94	6	1.52
Browns Ferry 1	12/20/73	1.11	4	3.60
Browns Ferry 2	08/02/74	0.67	1	1.49
Browns Ferry 3	08/18/76	0.25	3	12.00
Brunswick 1	11/12/76	3.56	11	3.09
Brunswick 2	12/27/74	3.19	8	2.51
Clinton 1	04/17/87	0.94	9	9.57
Cooper	01/18/74	3.30	13	3.94
Dresden 2	12/22/69	3.57	14	3.92
Dresden 3	01/12/71	3.07	22	7.17
Duane Arnold	02/22/74	3.53	5	1.42
Fermi 2	03/20/85	1.26	20	15.87
Fitzpatrick	10/17/74	3.78	13	3.44
Grand Gulf 1	07/01/82	2.88	27	9.38
Hatch 1	08/06/74	3.57	15	4.20
Hatch 2	06/13/78	3.63	18	4.96
Hope Creek 1	07/26/86	1.90	16	8.42
LaSalle 1	04/17/82	2.96	16	5.41
LaSalle 2	12/16/83	2.67	13	4.87
Limerick	10/26/84	2.82	5	1.77
Millstone 1	10/07/70	4.36	9	2.06
Monticello	09/08/70	3.64	8	2.20
Nine Mile Point 1	08/22/69	3.30	10	3.03
Nine Mile Point 2	10/31/86	0.53	15	28.30
Oyster Creek	04/09/69	2.55	11	4.31
Peach Bottom 2	08/08/73	1.65	8	4.85
Peach Bottom 3	07/02/74	2.23	13	5.83
Perry 1	03/18/86	0.88	13	14.77
Pilgrim	06/08/72	1.15	6	5.22
Quad Cities 1	10/31/71	3.88	7	1.80
Quad Cities 2	04/06/72	3.77	11	2.92
River Bend 1	11/20/85	2.17	22	10.14
Susquehanna 1	07/17/82	3.77	9	2.39
Susquehanna 2	03/23/84	3.41	10	2.93
Vermont Yankee	02/28/73	3.83	8	2.09
WPPSS 2	12/20/83	2.99	30	10.03
*** Subtotal ***	36 Units		429	211.43

All plants:	Average=	5.87	Sigma=	5.2893
Mature plants*:	Average=	4.02	Sigma=	2.44
New plants**:	Average=	10.67	Sigma=	7.31

* 26 plants with OL before 1/1/83

** 10 plants with OL after 1/1/83

Table D.4

BOP Trips by Plant per Critical Year: Cumulative Average for Westinghouse Units

Plant Name	OL Date	Critical Years	BOP Trips 1/84 - 12/88	5-Yr Average BOP Trips per Critical Year
Beaver Valley 1	01/30/76	4.03	14	3.47
Beaver Valley 2	08/01/87	1.05	12	11.43
Braidwood 1	05/21/87	0.75	9	12.00
Braidwood 2	12/18/87	0.17	10	58.82
Byron 1	02/14/85	2.49	20	8.03
Byron 2	01/30/87	1.25	13	10.40
Callaway 1	06/11/84	3.44	33	9.59
Catawba 1	06/01/85	2.53	18	7.11
Catawba 2	05/01/86	1.72	21	12.21
Connecticut Yankee	06/30/67	3.55	10	2.82
Cook 1	10/25/74	3.72	10	2.69
Cook 2	12/23/77	2.94	18	6.12
Diablo Canyon 1	09/22/81	3.01	18	5.98
Diablo Canyon 2	08/26/85	2.33	21	9.01
Farley 1	06/25/77	4.28	10	2.34
Farley 2	10/23/80	4.35	7	1.61
Ginna	09/19/69	4.34	11	2.53
Indian Point 2	09/28/73	3.67	23	6.2[illegible]
Indian Point 3	12/12/75	3.68	26	7.07
Kewaunee	12/21/73	4.34	16	3.69
McGuire 1	06/29/81	3.60	11	3.06
McGuire 2	03/01/83	3.62	27	7.46
Millstone 3	01/31/86	2.16	22	10.19
North Anna 1	11/26/77	3.63	16	4.41
North Anna 2	08/21/80	4.28	7	1.64
Point Beach 1	10/05/70	4.17	3	0.72
Point Beach 2	05/25/72	4.30	3	0.70
Prairie Island 1	08/09/73	4.41	5	1.13
Prairie Island 2	10/29/74	4.54	1	0.22
Robinson 2	09/23/70	3.16	15	4.75
Salem 1	04/06/77	3.59	19	5.29
Salem 2	08/18/81	3.04	31	10.20
San Onofre 1	03/27/67	2.49	3	1.20
Sequoyah 1	09/17/80	1.18	6	5.08
Sequoyah 2	09/15/81	1.92	11	5.73
Shearon Harris 1	01/12/87	1.26	21	16.67
South Texas 1	03/22/88	0.28	3	10.71
Summer 1	08/06/82	3.73	19	5.09
Surry 1	05/25/72	3.35	10	2.99
Surry 2	01/29/73	3.55	17	4.79
Trojan	11/21/75	3.36	12	3.57
Turkey Point 3	07/19/72	3.09	17	5.50
Turkey Point 4	04/10/73	2.92	13	4.45
Vogtle 1	01/16/87	1.24	21	16.94

Table D.4

BOP Trips by Plant per Critical Year: Cumulative Average for Westinghouse Units

(Continued)

Plant Name	OL Date	Critical Years	BOP Trips 1/84 - 12/88	5-Yr Average BOP Trips per Critical Year
Wolf Creek 1	03/11/85	2.46	24	9.76
Yankee-Rowe	07/09/60	4.23	9	2.13
Zion 1	04/06/73	3.50	14	4.00
Zion 2	11/14/73	3.71	11	2.96
*** Subtotal ***	48 Units		691	334.52

All plants:	Average=	6.97	Sigma=	8.5381
Mature plants*:	Average=	3.76	Sigma=	2.12
New plants**:	Average=	14.02	Sigma=	12.29

* 33 plants with OL before 1/1/83

** 15 plants with OL after 1/1/83

APPENDIX E

BOP Trips by Systems and Subsystems

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Susbsystem	No. of Trips
AC Power		4
AC Power	High Voltage	77
AC Power	High Voltage Offsite	3
AC Power	Low Voltage	6
AC Power	Medium Voltage	31
AC Power	Vital AC (120v)	47
Subtotal:		168
AC Power, Feedwater	Low Voltage	1
Subtotal:		1
Air		42
Air	Compressor	1
Air	Prefilter System	1
Subtotal:		44
Air, Feedwater		1
Subtotal:		1
Auxiliary Feedwater		2
Auxiliary Feedwater	AFW Initiation and Control	2
Subtotal:		4
Circulating Water		24
Circulating Water	Lube Oil	2
Circulating Water	Lube Oil Cooling Water	1
Subtotal:		27
Communications		1
Subtotal:		1
Component Cooling Water		1
Subtotal:		1

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Subsystem	No. of Trips
Computer		1
Subtotal:		1
Condensate Storage		1
Subtotal:		1
Containment Isolation	Feedwater	2
Subtotal:		2
Cooling Water		1
Subtotal:		1
DC Power		16
DC Power	Inverter	2
Subtotal:		18
Drains		2
Subtotal:		2
Feedwater		135
Feedwater	Condensate	26
Feedwater	Condensate Polisher	1
Feedwater	Containment Isolation	1
Feedwater	Demineralized Water	10
Feedwater	Drain	3
Feedwater	Feedwater Control	344
Feedwater	Feedwater Drain	1
Feedwater	Feedwater Heater	23
Feedwater	Feedwater Indication	1
Feedwater	Feedwater Instrumentation	1
Feedwater	Feedwater Isolation	4
Feedwater	Feedwater Lube Oil	3
Feedwater	Heater Drain	2
Feedwater	Lube Oil	5
Feedwater	Steam Relief	1
Subtotal:		561

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Subsystem	No. of Trips
Feedwater, Main Steam	Feedwater Control	1
Subtotal:		1
Feedwater, Steam Generator	Feedwater Control	1
Subtotal:		1
Fire Protection		4
Subtotal:		4
HVAC		3
HVAC	Battery Room Cooling	1
HVAC	Cabinet Cooling	1
Subtotal:		5
HVAC (Building)		2
Subtotal:		2
HVAC Turbine building		1
Subtotal:		1
Instrumentation and Control		7
Instrumentation and Control	Control Room Instrumentation	1
Instrumentation and Control	Feedwater Control	3
Instrumentation and Control	Nuclear Instrumentation	3
Instrumentation and Control	Power Range Instrumentation	3
Instrumentation and Control	RCP Trip Circuit	2
Instrumentation and Control	Radiation	2
Instrumentation and Control	Safeguards Logic	1
Instrumentation and Control	Steam Generator Control	2
Subtotal:		24
Main Steam		47
Main Steam	Drain	1
Main Steam	Excess Steam Vent	1
Main Steam	MSLB Logic	2

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Subsystem	No. of Trips
Main Steam	Main Steam Indication & Alarm	1
Main Steam	Main Steam Isolation	4
Main Steam	Main steam isolation valves	1
Main Steam	Moisture Separator Reheater	20
Main Steam	Offgas	2
Main Steam	Steam Bypass	1
Main Steam	Steam Isolation	1
Main Steam	Steam Jet Air Ejector	1
Main Steam	Steam Reheater	2
Main Steam	Steam Relief	6
Subtotal:		90
Non-condensible Gases Extract.		1
Subtotal:		1
Non-nuclear Instrumentation		5
Subtotal:		5
Nuclear Instrumentation		2
Subtotal:		2
Panels/Cabinets		1
Subtotal:		1
Power Conversion		1
Subtotal:		1
Primary System Drain		1
Subtotal:		1
RWCU Drains		1
Subtotal:		1

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Subsystem	No. of Trips
Reactor Coolant Pump I&C		1
Subtotal:		1
Reactor Coolant Pump Oil		1
Subtotal:		1
Seismic Trip		1
Subtotal:		1
Service Water		1
Subtotal:		1
Steam Generator		1
Steam Generator	Instrumentation	1
Steam Generator	SG Blowdown Drain	1
Steam Generator	SG Low Level Trip	1
Steam Generator	Steam Generator Ammonia Supply	1
Steam Generator	Steam Generator Control	2
Subtotal:		7
Turbine Generator		87
Turbine Generator	Condenser	33
Turbine Generator	Cooling Water	1
Turbine Generator	Drain	1
Turbine Generator	Exciter	1
Turbine Generator	Generator	9
Turbine Generator	Generator Cooling Water	2
Turbine Generator	Generator Hydrogen Control	1
Turbine Generator	Generator Hydrogen Seal Oil	2
Turbine Generator	Generator Stator Cooling	1
Turbine Generator	Lube Oil	8
Turbine Generator	Pressure Regulator	3
Turbine Generator	Steam Jet Air Ejector	1
Turbine Generator	Steam Relief	1
Turbine Generator	Steam Sealing	1
Turbine Generator	T/G Instrumentation & Control	250
Turbine Generator	Thrust Bearing Wear Detector	1
Turbine Generator	Turbine	3
Turbine Generator	Turbine Bypass	2

Reactor Trips Counted by BOP Systems and Subsystems

BOP System	BOP Subsystem	No. of Trips
Turbine Generator	Turbine Control Oil	1
Turbine Generator	Turbine Cooling Water	2
Turbine Generator	Turbine Drain	1
Turbine Generator	Turbine Lube Oil	3
Turbine Generator	Turbine Lube Oil Cooler	1
Turbine Generator	Turbine Steam Sealing	3
Subtotal:		419
Turbine Generator, Feedwater	T/G Instrumentation & Control	1
Subtotal:		1
Unknown		1
Subtotal:		1
Total No. of Trips:		1405

APPENDIX F

BOP Trips by Systems and Components

Reactor Trips Counted by BOP Components and Systems

BOP System	BOP Component	No. of Trips
AC Power		3
AC Power	345 Kv test block stud	1
AC Power	Auxiliary transformer	4
AC Power	Bus	5
AC Power	Bus duct	4
AC Power	Cable	1
AC Power	Capacitor	2
AC Power	Circuit breaker	10
AC Power	Circuit card	5
AC Power	Computer	1
AC Power	Conduit	1
AC Power	Connection	3
AC Power	Connector	1
AC Power	Control circuit	1
AC Power	Fuse	5
AC Power	Human	50
AC Power	Human (other unit)	1
AC Power	Input filter	1
AC Power	Insulation	6
AC Power	Inverter	7
AC Power	Inverter, Fuse	2
AC Power	Lightning arrester	1
AC Power	Main transformer	3
AC Power	Multiplexer	1
AC Power	Oscillator	1
AC Power	Oscillator, Voltage controller	1
AC Power	Rectifier, Fuse	1
AC Power	Relay	7
AC Power	Switch	1
AC Power	Switchgear cabinet	1
AC Power	Test switch	1
AC Power	Transformer	23
AC Power	Transmission line	2
AC Power	Unknown	8
AC Power	Wire	3
Subtotal:		168
AC Power, Feedwater	Circuit breaker, Pres regulator valve	1
Subtotal:		1
Air	Air control valve, Air line	1
Air	Air dryer	2
Air	Air line	14
Air	Air line moisture trap	1
Air	Compressor	2

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Air	Compressor, Pressure switch	1
Air	Fastener, Pressure transmitter	1
Air	Human	11
Air	Insulation	1
Air	Pipe	3
Air	Relay	1
Air	Solenoid valve	1
Air	Unknown	2
Air	Unknown (Contaminant)	2
Air	Valve	1
Subtotal:		44
Air, Feedwater	Air line, Feedback arm	1
Subtotal:		1
Auxiliary Feedwater	Human	2
Auxiliary Feedwater	Pipe	1
Auxiliary Feedwater	Transmitter/Receiver	1
Subtotal:		4
Circulating Water	Circuit card	1
Circulating Water	Human	5
Circulating Water	Isolation valve	1
Circulating Water	Jet pump	1
Circulating Water	Motor operated valve	1
Circulating Water	Multiplexer	1
Circulating Water	Pump	3
Circulating Water	Relay	1
Circulating Water	Screen	1
Circulating Water	Strainers	1
Circulating Water	Transmitter	1
Circulating Water	Traveling screen	5
Circulating Water	Unknown	1
Circulating Water	Valve	4
Subtotal:		27
Communications	Hand held radios	1
Subtotal:		1

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Component Cooling Water	Human	1
Subtotal:		1
Computer	Inverter	1
Subtotal:		1
Condensate Storage	Human	1
Subtotal:		1
Containment Isolation	Fuse	1
Containment Isolation	Solenoid valve	1
Subtotal:		2
Cooling Water	Human	1
Subtotal:		1
DC Power	Bus	2
DC Power	Circuit card	1
DC Power	Control circuit	1
DC Power	DC power source	1
DC Power	Human	10
DC Power	Interlock	1
DC Power	Unknown	1
DC Power	Wire	1
Subtotal:		18
Drains	Pipe	1
Drains	Weld	1
Subtotal:		2
Feedwater	Air operated valve	2
Feedwater	Air regulator	1
Feedwater	Bistable	1
Feedwater	Bypass valve	2
Feedwater	Capacitor	1

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Feedwater	Check valve	6
Feedwater	Check valve, Circuit breaker	1
Feedwater	Check valve, Pump	1
Feedwater	Circuit breaker	1
Feedwater	Circuit card	29
Feedwater	Circuit card, Valve	1
Feedwater	Computer	1
Feedwater	Condensate demineralizer	1
Feedwater	Condensate polisher programmer	1
Feedwater	Connection	1
Feedwater	Control circuit	10
Feedwater	Control oil	2
Feedwater	Control valve	11
Feedwater	Current-pressure converter	2
Feedwater	DC power source	1
Feedwater	DC power source, FW ht lev cntrl switch	1
Feedwater	Deaerator tank	1
Feedwater	Delta P controller	1
Feedwater	Diaphram	1
Feedwater	Drain tank	1
Feedwater	FWP delta P controller meter	1
Feedwater	Feed regulator valve, bypass	1
Feedwater	Feedwater control valve	3
Feedwater	Feedwater heater	1
Feedwater	Feedwater regulator valve	38
Feedwater	Feedwater regulator valve, Block valve	1
Feedwater	Feedwater square root extractor	1
Feedwater	Flow controller	3
Feedwater	Flow recorder	2
Feedwater	Flow transmitter	5
Feedwater	Flyball governor	1
Feedwater	Fuse	13
Feedwater	Heat exchanger	1
Feedwater	Heater Drain Tank	1
Feedwater	High signal selector	1
Feedwater	Human	213
Feedwater	Indicator	2
Feedwater	Level controller	6
Feedwater	Level recorder	1
Feedwater	Level sensor	2
Feedwater	Level switch	2
Feedwater	Level transmitter	1
Feedwater	Limit switch	4
Feedwater	Lube oil seperator	1
Feedwater	Manual isol valve, air operated valve	1
Feedwater	Manual valve	1
Feedwater	Motor operated valve	1
Feedwater	Nozzle	1
Feedwater	Oil filter	1

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Feedwater	Oil line	3
Feedwater	Oil line, Wire	1
Feedwater	Oil seal	1
Feedwater	Oil/water separator	1
Feedwater	Pipe	9
Feedwater	Pneumatic valve	1
Feedwater	Potentiometer	1
Feedwater	Power supply	1
Feedwater	Pressure Control	1
Feedwater	Pressure switch	3
Feedwater	Pressure transmitter	5
Feedwater	Pump	31
Feedwater	Recirculation valve	1
Feedwater	Regulator valve	2
Feedwater	Relay	10
Feedwater	Relay contacts	1
Feedwater	Relief valve	2
Feedwater	Rupture disk	1
Feedwater	Seal	2
Feedwater	Servo control motor	1
Feedwater	Solenoid valve	12
Feedwater	Solenoid valve, isolation valve	1
Feedwater	Speed controller	5
Feedwater	Speed indication	1
Feedwater	Steam/Feed mismatch summator	1
Feedwater	Strainers	2
Feedwater	Switch	5
Feedwater	Tachometer	[illegible]
Feedwater	Thrust bearing wear detector	1
Feedwater	Transmitter	1
Feedwater	Trip circuit	1
Feedwater	Trip switch	1
Feedwater	Tube	3
Feedwater	Turbine governor	2
Feedwater	Turbine pump	2
Feedwater	Unknown	21
Feedwater	Valve	23
Feedwater	Valve operators, Relays, Solenoid valve	1
Feedwater	Vent line	1
Feedwater	Vibration detector	1
Feedwater	Wire	6
Subtotal:		561
Feedwater, Main Steam	Human	1
Subtotal:		1

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Feedwater, Steam Generator	Level controller	1
Subtotal:		1
Fire Protection	Human	2
Fire Protection	Pressure regulator, Sensing header pipe	1
Fire Protection	Unknown	1
Subtotal:		4
HVAC	Fan	1
HVAC	Human	2
HVAC	Unknown	2
Subtotal:		5
HVAC (Building)	Fan	2
Subtotal:		2
HVAC Turbine Building	Human	1
Subtotal:		1
Instrumentation and Control	Amplifier	1
Instrumentation and Control	Capacitor	2
Instrumentation and Control	Circuit card	1
Instrumentation and Control	Connection	2
Instrumentation and Control	Human	7
Instrumentation and Control	Instrumentation	1
Instrumentation and Control	Inverter	2
Instrumentation and Control	Level transmitter pressurizing valve	1
Instrumentation and Control	Power supply	1
Instrumentation and Control	Radiation monitor	2
Instrumentation and Control	Relay	1
Instrumentation and Control	Static inverter, Switches	1
Instrumentation and Control	Transformer	1
Instrumentation and Control	Unknown	1
Subtotal:		24
Main Steam	Air operated check valve	3
Main Steam	Bypass valve	1

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Main Steam	Check valve	1
Main Steam	Circuit card	3
Main Steam	Control circuit	3
Main Steam	Drain tank	1
Main Steam	Drain valve	1
Main Steam	Gasket	3
Main Steam	Human	33
Main Steam	Human, Unknown	1
Main Steam	Insulation	1
Main Steam	Level controller	2
Main Steam	Level switch	2
Main Steam	Limit switch	1
Main Steam	Main Steam Isolation Valves	2
Main Steam	Pipe	2
Main Steam	Pneumatic valve	1
Main Steam	Pressure control valve	1
Main Steam	Pressure switch	2
Main Steam	Pressure transducer	1
Main Steam	Pressure transmitter	3
Main Steam	Pump	1
Main Steam	Relay	2
Main Steam	Relief valve	3
Main Steam	Solenoid valve	3
Main Steam	Trip circuitry	1
Main Steam	Trip switch	1
Main Steam	Trip valve	1
Main Steam	Unknown	3
Main Steam	Valve	6
Main Steam	Vent line	1
Subtotal:		90
Non-condensible Gases Extract.	Human	1
Subtotal:		1
Non-nuclear Instrumentation		1
Non-nuclear Instrumentation	Flow switch	1
Non-nuclear Instrumentation	Fuse holders	1
Non-nuclear Instrumentation	Human	1
Non-nuclear Instrumentation	Valve	1
Subtotal:		5
Nuclear Instrumentation	Human	2
Subtotal:		2

Reactor Trips Counter by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Panels/Cabinets	Human	1
Subtotal:		1
Power Conversion	Pipe	1
Subtotal:		1
Primary System Drain	Valve	1
Subtotal:		1
RWCU Drains	Pipe	1
Subtotal:		1
Reactor Coolant Pump I&C	Relay	1
Subtotal:		1
Reactor Coolant Pump Oil	Level switch	1
Subtotal:		1
Seismic Trip	Coil, relay	1
Subtotal:		1
Service Water	Human	1
Subtotal:		1
Steam Generator	Bypass valve	1
Steam Generator	Control circuit	1
Steam Generator	Flow transmitter	1
Steam Generator	Human	2
Steam Generator	Relief valve	1
Steam Generator	Valve	1
Subtotal:		7

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Turbine Generator		2
Turbine Generator	Amplifier	1
Turbine Generator	Bearing	3
Turbine Generator	Bellows	3
Turbine Generator	Brush Assembly Enclosure	1
Turbine Generator	Brush collector ring	1
Turbine Generator	Bushing	2
Turbine Generator	Button	1
Turbine Generator	Bypass valve	4
Turbine Generator	Cable	1
Turbine Generator	Capacitor	1
Turbine Generator	Check valve	1
Turbine Generator	Circuit breaker	2
Turbine Generator	Circuit card	17
Turbine Generator	Condenser water box	1
Turbine Generator	Connection	3
Turbine Generator	Control circuit	4
Turbine Generator	Control oil	1
Turbine Generator	Control valve	10
Turbine Generator	Fan	1
Turbine Generator	Filter	2
Turbine Generator	Flow switch	1
Turbine Generator	Fluid cooler	1
Turbine Generator	Fluid filter	1
Turbine Generator	Fuse	2
Turbine Generator	Gasket	4
Turbine Generator	Generator conformable layer	1
Turbine Generator	Generator exciter	8
Turbine Generator	Generator stator coils	1
Turbine Generator	Governor	1
Turbine Generator	Hatch	1
Turbine Generator	Human	128
Turbine Generator	Hydraulic control system	2
Turbine Generator	Impeller	1
Turbine Generator	Indicator	2
Turbine Generator	Instrumentation	1
Turbine Generator	Insulation	1
Turbine Generator	Level controller	1
Turbine Generator	Level switch	1
Turbine Generator	Level transmitter	3
Turbine Generator	Limit switch	4
Turbine Generator	Main steam line instrument rack	1
Turbine Generator	Manual valve	1
Turbine Generator	Motor operated disconnect	1
Turbine Generator	Oil line	1
Turbine Generator	Orifice	4
Turbine Generator	Pilot valve	1
Turbine Generator	Pipe	9
Turbine Generator	Potentiometer	4

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Turbine Generator	Power supply	3
Turbine Generator	Power supply, Circuit card	1
Turbine Generator	Pressure control valve	1
Turbine Generator	Pressure limiter	1
Turbine Generator	Pressure regulator	4
Turbine Generator	Pressure regulator valve	1
Turbine Generator	Pressure sensor	1
Turbine Generator	Pressure switch	4
Turbine Generator	Pressure transducer	1
Turbine Generator	Pressure transmitter	4
Turbine Generator	Pump	7
Turbine Generator	Recirculation valve	1
Turbine Generator	Rectifier banks cooler	1
Turbine Generator	Regulator valve	1
Turbine Generator	Relay	9
Turbine Generator	Relay contacts	4
Turbine Generator	Relay contacts, Roller wheel	1
Turbine Generator	Relays, Circuit card	1
Turbine Generator	Relief device	1
Turbine Generator	Relief valve	4
Turbine Generator	Rotor collector ring	1
Turbine Generator	Rubber expansion joint	1
Turbine Generator	Seal	2
Turbine Generator	Sensor	2
Turbine Generator	Solenoid valve	4
Turbine Generator	Speed controller	1
Turbine Generator	Stator coil	1
Turbine Generator	Stop valve	2
Turbine Generator	Strainer, Orifice	1
Turbine Generator	Switch	3
Turbine Generator	Switch, Gauge	1
Turbine Generator	Temperature controller	1
Turbine Generator	Temperature sensor	1
Turbine Generator	Temperature switch	1
Turbine Generator	Test switch	1
Turbine Generator	Thrust bearing wear detector	3
Turbine Generator	Transducer	2
Turbine Generator	Transfer valve	1
Turbine Generator	Transformer	6
Turbine Generator	Trip Latch	1
Turbine Generator	Trip circuit	1
Turbine Generator	Trip switch	1
Turbine Generator	Tube	9
Turbine Generator	Turbine	5
Turbine Generator	Turbine bearing	1
Turbine Generator	Turbine bearing wear detector	1
Turbine Generator	Turbine bypass valve	5
Turbine Generator	Turbine control valve	1
Turbine Generator	Turbine governor	4

Reactor Trips Counted by BOP Component and Systems

BOP System	BOP Component	No. of Trips
Turbine Generator	Turbine governor valve	9
Turbine Generator	Turbine governor valve, bypass valve	1
Turbine Generator	Turbine header press. hand/auto station	1
Turbine Generator	Turbine stop valve	3
Turbine Generator	Unknown	25
Turbine Generator	Valve	7
Turbine Generator	Vibration detector	1
Turbine Generator	Vibration indicator	1
Turbine Generator	Voltage regulator	2
Turbine Generator	Weld	1
Turbine Generator	Wire	4
Subtotal:		419
Turbine Generator, Feedwater	Relay, Turbine feedpump coupling	1
Subtotal:		1
Unknown	Unknown	1
Subtotal:		1
Total No. of Trips:		1405

APPENDIX G

Mature Plants' Feedwater Pumping Capacity

MATURE PLANTS' FEEDWATER PUMPING CAPACITY

Plant	"N-1" Capacity (%)	Number of FW Pumps (M=motor;T=turbine)	Trips per Critical Year Jan. 1, 1984 - early 1988		
			BOP	FW	FW Control
Dresden 2	100	3M	5.0	2.1	2.1
Dresden 3	100	3M	8.9	3.4	3.0
Ft. Calhoun	100	3M	0.3	0	0
Nine Mile 1	100	3(M/T)	3.0	0.9	0.9
North Anna 1	100	3M	4.4	2.2	1.5
North Anna 2	100	3M	1.8	0	0
Zion 1	100	3(M/T)	3.6	1.1	0.7
Zion 2	100	3(M/T)	3.1	1.4	1.0
Millstone 1	87	3M	2.4	0.6	0.3
ANO-1	80	2T	4.6	3.0	3.0
Rancho Seco	80	2T	8.5	4.3	3.2
Sequoyah 1	78	2T	3.5	0.9	0.9
Sequoyah 2	78	2T	5.3	3.8	1.5
Peach Bottom 2	72	3T	4.2	1.8	1.2
Peach Bottom 3	72	3T	5.8	2.7	1.3
DC Cook 1	70	2T	3.6	2.2	1.8
DC Cook 2	70	2T	6.8	2.3	1.9
Trojan	70	2T	3.6	1.5	0.9
Duane Arnold	68	2M	1.4	0.4	0
Grand Gulf	67	3T	12.6	5.2	1.0
V.C. Summer	67	3T	6.3	2.7	0.7
Susquehanna 1	67	3T	2.5	0.4	0.4
Prairie Island 1	65	2M	1.1	0.8	0.6
Prairie Island 2	65	2M	0.3	0	0
ANO 2	60	2T	4.6	1.2	1.2
Robinson 2	60	2M	4.8	2.0	2.0
TMI 1	60	2T	4.7	1.2	1.2
Turkey Point 3	60	2M	6.9	0.8	0.7
Turkey Point 4	60	2M	5.1	0.8	0.4
Indian Point 2	55	2T	6.8	3.9	3.6
Millstone 2	55	2T	3.5	1.9	1.3
St. Lucie 1	55	2M	2.5	1.9	1.6
Surry 1	55	2M	3.4	1.4	0
Surry 2	55	2M	5.2	2.6	1.5
Beaver Valley 1	50	2M	3.7	0.6	0.6
Conn. Yankee	50	2M	3.2	1.8	0.7
Davis-Besse	50	2T	4.9	3.8	2.7
Diablo Canyon 1	50	2T	6.4	2.1	1.3
Farley 1	50	2T	2.6	1.4	1.2
Farley 2	50	2T	2.1	1.5	0.9
Fitzpatrick	50	2T	4.2	1.6	0.3
Ginna	50	2M	2.6	0.3	0.3
Indian Point 3	50	2T	7.7	4.9	3.5

MATURE PLANTS FEEDWATER PUMPING CAPACITY (Continued)

Plant	"N-1" Capacity (%)	Number of FW Pumps (M=motor;T=turbine)	Trips per Critical Year Jan. 1, 1984 - early 1988 BOP	FW	FW Control
Kewaunee	50	2M	4.1	1.8	1.8
Palisades	50	2T	5.4	0.6	0
Point Beach 1	50	2M	0.7	0.0	0.0
Point Beach 2	50	2M	0.7	0.0	0.0
Salem 1	50	2T	5.8	3.6	2.2
Salem 2	50	2T	12.3	5.5	4.7
San Onofre 1	50	2M	1.5	0	0
San Onofre 2	50	2M	3.8	2.3	0.4

APPENDIX H

BOP Trips by Plant with NSSS Vendor and A/E

Reactor Trips Counted by Plant
(with Associated NSSS Vendors
and by A/E Firms

Plant Name	NSSS Vendors	A/E Firms	No. of Trips
Arkansas Nuclear One - 1	B & W	Bechtel	14
Arkansas Nuclear One - 2	CE	Bechtel	12
Beaver Valley 1	W	Duquesne Light/Stone & Webster	14
Beaver Valley 2	W	Duquesne Light/Stone & Webster	12
Big Rock Point	GE	Bechtel	6
Braidwood 1	W	Sargent & Lundy	9
Braidwood 2	W	Sargent & Lundy	10
Browns Ferry 1	GE	TVA	4
Browns Ferry 2	GE	TVA	1
Browns Ferry 3	GE	TVA	3
Brunswick 1	GE	United Engineers	11
Brunswick 2	GE	United Engineers	8
Byron 1	W	Sargent & Lundy	20
Byron 2	W	Sargent & Lundy	13
Callaway 1	W	Bechtel	33
Calvert Cliffs 1	CE	Bechtel	20
Calvert Cliffs 2	CE	Bechtel	11
Catawba 1	W	Duke Power Company	18
Catawba 2	W	Duke Power Company	21
Clinton 1	GE	Sargent & Lundy	9
Connecticut Yankee	W	Stone & Webster	10
Cook 1	W	American Electric Power	10
Cook 2	W	American Electric Power	18
Cooper	GE	Burns & Roe	13
Crystal River 3	B & W	Gilbert	14
Davis Besse	B & W	Bechtel	13
Diablo Canyon 1	W	Pacific Gas & Electric	18
Dirblo Canyon 2	W	Pacific Gas & Electric	21
Dresden 2	GE	Sargent & Lundy	14
Dresden 3	GE	Sargent & Lundy	22
Duane Arnold	GE	Bechtel	5
Farley 1	W	Southern Company and Bechtel	10
Farley 2	W	Southern Company and Bechtel	7
Fermi 2	GE	Detroit Edison and S & L	20
Fitzpatrick	GE	Stone & Webster	13
Fort Calhoun 1	CE	Gibbs and Hill	1
Ginna	W	Gilbert	11
Grand Gulf 1	GE	Bechtel	27
Hatch 1	GE	Southern Company and Bechtel	15
Hatch 2	GE	Southern Company and Bechtel	18
Hope Creek 1	GE	Bechtel	16
Indian Point 2	W	United Engineers	23
Indian Point 3	W	United Engineers	26
Kewaunee	W	Fluor Pioneer	16
LaSalle 1	GE	Sargent & Lundy	16
LaSalle 2	GE	Sargent & Lundy	13
Limerick	GE	Bechtel	5

Reactor Trips Counted by Plant
(with Associated NSSS Vendors
and by A/E Firms

Plant Name	NSSS Vendors	A/E Firms	No. of Trips
Maine Yankee	CE	Stone & Webster	25
McGuire 1	W	Duke Power Company	11
McGuire 2	W	Duke Power Company	27
Millstone 1	GE	Ebasco	9
Millstone 2	CE	Bechtel	11
Millstone 3	W	Stone & Webster	22
Monticello	GE	Bechtel	8
Nine Mile Point 1	GE	Niagara Mohawk Power Corp.	10
Nine Mile Point 2	GE	Stone & Webster	15
North Anna 1	W	Stone & Webster	16
North Anna 2	W	Stone & Webster	7
Oconee 1	B & W	Duke and Bechtel	9
Oconee 2	B & W	Duke and Bechtel	9
Oconee 3	B & W	Duke and Bechtel	9
Oyster Creek	GE	Burns & Roe and GE (Turnkey!)	11
Palisades	CE	Bechtel	9
Palo Verde 1	CE	Bechtel	17
Palo Verde 2	CE	Bechtel	9
Peach Bottom 2	GE	Bechtel	8
Peach Bottom 3	GE	Bechtel	13
Perry 1	GE	Gilbert	13
Pilgrim	GE	Bechtel	6
Point Beach 1	W	Bechtel	3
Point Beach 2	W	Bechtel	3
Prairie Island 1	W	Fluor Pioneer	5
Prairie Island 2	W	Fluor Pioneer	1
Quad Cities 1	GE	Sargent & Lundy	7
Quad Cities 2	GE	Sargent & Lundy	11
Rancho Seco	B & W	Bechtel	11
River Bend 1	GE	Stone & Webster	22
Robinson 2	W	Ebasco	15
Salem 1	W	Public Service Electric & Gas	19
Salem 2	W	Public Service Electric & Gas	31
San Onofre 1	W	Bechtel	3
San Onofre 2	CE	Bechtel	11
San Onofre 3	CE	Bechtel	10
Sequoyah 1	W	TVA	6
Sequoyah 2	W	TVA	11
Shearon Harris 1	W	Ebasco	21
Shoreham	GE	Stone & Webster	1
South Texas 1	W	Bechtel	3
St. Lucie 1	CE	Ebasco	12
St. Lucie 2	CE	Ebasco	20
Summer 1	W	Gilbert	19
Surry 1	W	Stone & Webster	10
Surry 2	W	Stone & Webster	17
Susquehanna 1	GE	Bechtel	9

Reactor Trips Counted by Plant
(with Associated NSSS Vendors
and by A/E Firms

Plant Name	NSSS Vendors	A/E Firms	No. of Trips
Susquehanna 2	GE	Bechtel	10
Three Mile Island 1	B & W	Gilbert	10
Trojan	W	Bechtel	12
Turkey Point 3	W	Bechtel	17
Turkey Point 4	W	Bechtel	13
Vermont Yankee	GE	Ebasco	8
Vogtle 1	W	Bechtel	21
WPPSS 2	GE	Burns & Roe	30
Waterford 3	CE	Ebasco	27
Wolf Creek 1	W	Bechtel and Sargent & Lundy	24
Yankee-Rowe	W	Stone & Webster	9
Zion 1	W	Sargent & Lundy	14
Zion 2	W	Sargent & Lundy	11
	Total No. of Trips:		1405

APPENDIX I

BOP Trips as a Function of Power Level

Listing of Reactor Trips by Power Level

Plant Name	0-5%	5-25%	25-50%	50-75%	75-100%	Total
Arkansas Nuclear One - 1	0	3	2	1	8	14
Arkansas Nuclear One - 2	2	1	0	1	8	12
Beaver Valley 1	0	4	2	0	8	14
Beaver Valley 2	0	2	1	3	6	12
Big Rock Point	1	4	0	0	1	6
Braidwood 1	3	1	2	0	3	9
Braidwood 2	2	3	2	1	2	10
Browns Ferry 1	1	1	0	0	2	4
Browns Ferry 2	0	0	0	1	0	1
Browns Ferry 3	1	0	1	0	1	3
Brunswick 1	1	0	3	1	6	11
Brunswick 2	1	0	0	2	5	8
Byron 1	0	7	4	0	9	20
Byron 2	1	5	1	0	6	13
Callaway 1	7	4	4	3	15	33
Calvert Cliffs 1	1	2	2	1	14	20
Calvert Cliffs 2	0	1	1	0	9	11
Catawba 1	0	5	0	3	10	18
Catawba 2	1	7	4	2	7	21
Clinton 1	1	2	0	3	3	9
Connecticut Yankee	1	0	1	1	7	10
Cook 1	1	2	0	2	5	10
Cook 2	5	3	0	1	9	18
Cooper	1	2	3	3	4	13
Crystal River 3	1	4	0	3	6	14
Davis Besse	2	1	4	1	5	13
Diablo Canyon 1	1	6	5	1	5	18
Diablo Canyon 2	0	5	7	1	8	21
Dresden 2	3	0	2	2	7	14
Dresden 3	2	4	3	1	12	22
Duane Arnold	1	0	0	1	3	5
Farley 1	0	1	1	0	8	10
Farley 2	0	2	1	1	3	7
Fermi 2	8	7	3	1	1	20
Fitzpatrick	0	2	1	2	8	13
Fort Calhoun 1	0	0	0	0	1	1
Ginna	2	1	1	0	7	11
Grand Gulf 1	3	4	1	8	11	27
Hatch 1	0	1	1	3	10	15
Hatch 2	1	1	3	0	13	18
Hope Creek 1	3	1	1	1	10	16
Indian Point 2	5	4	3	1	10	23
Indian Point 3	2	5	1	3	15	26
Kewaunee	4	4	0	2	6	16
LaSalle 1	1	0	1	6	8	16
LaSalle 2	1	2	1	3	6	13
Limerick	1	0	1	0	3	5
Maine Yankee	1	6	0	5	13	25
McGuire 1	0	0	0	2	9	11

Listing of Reactor Trips by Power Level

Plant Name	0-5%	5-25%	25-50%	50-75%	75-100%	Total
McGuire 2	0	3	1	0	23	27
Millstone 1	1	1	1	1	5	9
Millstone 2	0	2	1	1	7	11
Millstone 3	0	10	2	1	9	22
Monticello	1	0	0	1	6	8
Nine Mile Point 1	3	1	0	0	6	10
Nine Mile Point 2	3	1	6	2	3	15
North Anna 1	0	4	0	0	12	16
North Anna 2	1	0	0	1	5	7
Oconee 1	0	1	2	1	5	9
Oconee 2	0	0	1	1	7	9
Oconee 3	0	2	1	2	4	9
Oyster Creek	3	1	1	0	6	11
Palisades	0	0	3	3	3	9
Palo Verde 1	1	3	2	4	7	17
Palo Verde 2	0	2	2	1	4	9
Peach Bottom 2	1	0	2	1	4	8
Peach Bottom 3	3	0	2	1	7	13
Perry 1	3	1	3	4	2	13
Pilgrim	1	2	1	0	2	6
Point Beach 1	0	0	0	0	3	3
Point Beach 2	1	0	0	0	2	3
Prairie Island 1	1	2	0	0	2	5
Prairie Island 2	0	0	0	0	1	1
Quad Cities 1	1	1	0	2	3	7
Quad Cities 2	1	0	1	0	9	11
Rancho Seco	0	3	1	3	4	11
River Bend 1	6	3	2	6	5	22
Robinson 2	1	4	0	4	6	15
Salem 1	0	3	0	3	13	19
Salem 2	3	7	2	6	13	31
San Onofre 1	0	0	0	0	3	3
San Onofre 2	1	0	0	1	9	11
San Onofre 3	0	2	1	0	7	10
Sequoyah 1	0	1	0	1	4	6
Sequoyah 2	0	4	1	1	5	11
Shearon Harris 1	1	3	5	3	9	21
Shoreham	1	0	0	0	0	1
South Texas 1	0	1	0	0	2	3
St. Lucie 1	1	2	1	0	8	12
St. Lucie 2	0	6	3	1	10	20
Summer 1	0	6	1	1	11	19
Surry 1	0	4	1	0	5	10
Surry 2	1	8	1	0	7	17
Susquehanna 1	0	1	1	2	5	9
Susquehanna 2	2	0	2	1	5	10
Three Mile Island 1	1	3	0	1	5	10
Trojan	0	1	2	1	8	12
Turkey Point 3	1	2	4	1	9	17

Listing of Reactor Trips by Power Level

Plant Name	0-5%	5-25%	25-50%	50-75%	75-100%	Total
Turkey Point 4	0	1	1	0	11	13
Vermont Yankee	1	2	0	0	5	8
Vogtle 1	2	5	1	0	13	21
WPPSS 2	3	13	6	2	6	30
Waterford 3	2	8	0	3	14	27
Wolf Creek 1	3	4	4	1	12	24
Yankee-Rowe	2	0	0	1	6	9
Zion 1	3	4	2	1	4	14
Zion 2	4	3	0	0	4	11
Total No. of Trips:	137	261	148	148	711	1405

APPENDIX J

Supplemental Data Base
Plant Design Data

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Arkansas Nuclear One 1	12/01/74	836	B & W	Bechtel	W
Arkansas Nuclear One 2	07/18/78	858	CE	Bechtel	GE
Beaver Valley 1	01/30/76	810	W	Duquesne Light & Stone & Webst	W
Beaver Valley 2	08/01/87	830	W	Duquesne Light/Stone & Webster	W
Big Rock Point	08/30/62	69	GE	Bechtel	GE
Braidwood 1	05/21/87	1120	W	S & L	W
Braidwood 2	12/18/87	1120	W	Sargent & Lundy	W
Browns Ferry 1	12/20/73	1065	GE	TVA	GE
Browns Ferry 2	08/02/74	1065	GE	TVA	GE
Browns Ferry 3	08/18/76	1065	GE	TVA	GE
Brunswick 1	11/12/76	790	GE	United Engineers	GE
Brunswick 2	12/27/74	790	GE	United Engineers	GE
Byron 1	02/14/85	1120	W	S & L	W
Byron 2	01/30/87	1120	W	S & L	W
Callaway 1	06/11/84	1150	W	Bechtel	GE
Calvert Cliffs 1	07/31/74	845	CE	Bechtel	GE
Calvert Cliffs 2	08/13/76	845	CE	Bechtel	W

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Catawba 1	06/01/85	1145	W	Duke Power Company	GE
Catawba 2	05/01/86	1145	W	Duke Power Company	GE
Clinton 1	04/17/87	950	GE	Sargent & Lundy	GE
Connecticut Yankee	06/30/67	582	W	Stone & Webster	W
Cook 1	10/25/74	1030	W	American Electric Power	GE
Cook 2	12/23/77	1100	W	American Electric Power	Brown Boveri
Cooper	01/18/74	778	GE	Burns & Roe	W
Crystal River 3	12/03/76	837	B & W	Gilbert	W
Davis Besse	04/22/77	906	B & W	Bechtel	GE
Diablo Canyon 1	09/22/81	1086	W	Pacific Gas & Electric	W
Diablo Canyon 2	08/26/85	1119	W	Pacific Gas and Electric	W
Dresden 2	12/22/69	794	GE	Sargent & Lundy	GE
Dresden 3	01/12/71	794	GE	Sargent & Lundy	GE
Duane Arnold	02/22/74	538	GE	Bechtel	GE
Farley 1	06/25/77	829	W	Southern Company and Bechtel	W

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME CF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Farley 2	10/23/80	829	W	Southern Company and Bechtel	W
Fermi 2	03/20/85	1093	GE	Detroit Edison and S & L	English Electric
Fitzpatrick	10/17/74	816	GE	Stone & Webster	GE
Fort Calhoun	05/24/73	492	CE	Gibbs and Hill	GE
Ginna	09/19/69	470	W	Gilbert	W
Grand Gulf 1	07/01/82	1250	GE	Bechtel	Allis-Chalmers
Hatch 1	08/06/74	786	GE	Southern Company and Bechtel	GE
Hatch 2	06/13/78	795	GE	Southern Company and Bechtel	GE
Hope Creek 1	07/26/86	1067	GE	Bechtel	GE
Indian Point 2	09/28/73	873	W	United Engineers	W
Indian Point 3	12/12/75	965	W	United Engineers	W
Kewaunee	12/21/73	535	W	Fluor Pioneer	W
La Crosse	11/01/69	50	Allis Chalmers	Sargent & Lundy	Allis Chalmers
La Salle 1	04/17/82	1078	GE	S & L	GE
La Salle 2	12/16/83	1078	GE	S & L	GE

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Limerick	10/26/84	1055	GE	Bechtel	GE
Maine Yankee	06/01/73	825	CE	Stone & Webster	W
Mcguire 1	06/29/81	1180	W	Duke Power Company	W
Mcguire 2	03/01/83	1180	W	Duke Power Company	W
Millstone 1	10/07/70	660	GE	Ebasco	GE
Millstone 2	08/01/75	870	CE	Bechtel	GE
Millstone 3	01/31/86	1150	W	Stone & Webster	GE
Monticello	09/08/70	536	GE	Bechtel	GE
Nine Mile Point 1	08/22/69	610	GE	Niagara Mohawk Power Corp.	GE
Nine Mile Point 2	10/31/86	1080	GE	Stone & Webster	GE
North Anna 1	11/26/77	915	W	Stone & Webster	W
North Anna 2	08/21/80	915	W	Stone & Webster	W
Oconee 1	02/06/73	860	B & W	Duke and Bechtel	GE
Oconee 2	10/06/73	860	B & W	Duke and Bechtel	GE
Oconee 3	07/19/74	860	B & W	Duke and Bechtel	GE
Oyster Creek	04/09/69	620	GE	Burns & Roe and GE - TURNKEY !	GE

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Palisades	10/01/72	777	CE	Bechtel	W
Palo Verde 1	12/31/84	1270	CE	Bechtel	GE
Palo Verde 2	12/09/85	1270	CE	Bechtel	GE
Palo Verde 3	03/25/87	1270	CE	Bechtel	GE
Peach Bottom 2	08/08/73	1065	GE	Bechtel	GE
Peach Bottom 3	07/02/74	1065	GE	Bechtel	GE
Perry 1	03/18/86	1205	GE	Gilbert	GE
Pilgrim	06/08/72	670	GE	Bechtel	GE
Point Beach 1	10/05/70	497	W	Bechtel	W
Point Beach 2	05/25/72	497	W	Bechtel	W
Prairie Island 1	08/09/73	520	W	Fluor Pioneer	W
Prairie Island 2	10/29/74	520	W	Fluor Pioneer	W
Quad Cities 1	10/01/71	789	GE	S & L	GE
Quad Cities 2	04/06/72	789	GE	S & L	GE
Rancho Seco	08/16/74	916	B & W	Bechtel	W
River Bend 1	11/20/85	940	GE	Stone & Webster	GE
Robinson 2	09/23/70	665	W	Ebasco	W
Salem 1	04/06/77	1090	W	Public Service Electric & Gas	W
Salem 2	08/18/81	1115	W	Public Service	W

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
San Onofre 1	03/27/67	436	W	Bechtel	W
San Onofre 2	09/07/82	1070	CE	Bechtel	GEC Turbine Generators, Ltd.
San Onofre 3	09/16/83	1080	CE	Bechtel	GEC Turbine Generators, Ltd.
Sequoyah 1	09/17/80	1148	W	TVA	W
Sequoyah 2	09/15/81	1148	W	TVA	W
Shearon Harris 1	01/12/87	860	W	Ebasco	W
Shoreham	12/31/99	809	GE	Stone & Webster	GE
South Texas 1	03/22/88	1250	W	Bechtel	W
St Lucie 1	03/01/76	810	CE	Ebasco	W
St. Lucie 2	04/06/83	810	CE	Ebasco	W
Summer 1	08/06/82	900	W	Gilbert	GE
Surry 1	05/25/72	781	W	Stone & Webster	W
Surry 2	01/29/73	781	W	Stone & Webster	W
Susquehanna 1	07/17/82	1050	GE	Bechtel	GE
Susquehanna 2	03/23/84	1050	GE	Bechtel	GE
Three Mile Island 1	04/19/74	792	B & W	Gilbert	GE
Trojan	11/21/75	1130	W	Bechtel	GE
Turkey Point 3	07/19/72	728	W	Bechtel	W
Turkey Point 4	04/10/73	728	W	Bechtel	W

LISTING OF OPERATING NUCLEAR POWER PLANTS

NAME OF PLANT	OL DATE	POWER	NSSS	A/E	T/G MFG.
Vermont Yankee	02/28/73	514	GE	Ebasco	GE
Vogtle 1	01/16/87	1160	W	Bechtel	GE
WPPSS 2	12/20/83	1150	GE	Burns & Roe	W
Waterford 3	12/18/84	1165	CE	Ebasco	W
Wolf Creek	03/11/85	1150	W	Bechtel and Sargent & Lundy	GE
Yankee-Rowe	07/09/60	175	W	Stone & Webster	W
Zion 1	04/06/73	1040	W	S & L	W
Zion 2	11/14/73	1040	W	S & L	W

APPENDIX K

Supplemental Data Base
Critical Hour Data

Listing of Critical Hours for Each Plant During 1984 - 1988 Period

Plant Name	Critical Hours 1984	Critical Hours 1985	Critical Hours 1986	Critical Hours 1987	Critical Hours 1988	5-Yr Total Critical Years
Arkansas Nuclear One - 1	6222.4	7005.4	5536.5	7855.7	6156.60	3.70
Arkansas Nuclear One - 2	7631.9	6377.4	6370.0	7715.4	6032.00	3.89
Beaver Valley 1	6476.3	8245.3	6243.8	7339.4	7066.70	4.03
Beaver Valley 2	0.0	0.0	0.0	965.5	8283.80	1.05
Big Rock Point	6981.9	6539.5	8387.3	6214.6	6394.20	3.94
Braidwood 1	0.0	0.0	0.0	3059.7	3510.40	0.75
Braidwood 2	0.0	0.0	0.0	0.0	1517.20	0.17
Browns Ferry 1	8067.4	1647.7	0.0	0.0	0.00	1.11
Browns Ferry 2	5895.7	0.0	0.0	0.0	0.00	0.67
Browns Ferry 3	700.7	1517.5	0.0	0.0	0.00	0.25
Brunswick 1	7023.8	3409.6	8317.6	5788.7	6660.70	3.56
Brunswick 2	2650.1	7134.8	4232.4	8328.4	5645.80	3.19
Byron 1	0.0	1281.0	7820.9	6210.3	6485.10	2.49
Byron 2	0.0	0.0	0.0	2327.2	8676.00	1.25
Callaway 1	302.5	8161.0	7306.6	6227.7	8202.10	3.44
Calvert Cliffs 1	7531.0	5367.6	6906.2	6615.5	6398.50	3.74
Calvert Cliffs 2	6630.2	6884.2	8443.0	5957.8	7827.10	4.08
Catawba 1	0.0	3612.4	5425.2	6076.4	7070.30	2.53
Catawba 2	0.0	0.0	1392.9	7212.8	6496.80	1.72
Clinton 1	0.0	0.0	0.0	898.3	7399.40	0.94
Connecticut Yankee	6515.6	8682.4	5060.9	4728.9	6177.00	3.55
Cook 1	8085.9	2595.6	7536.4	6000.6	8433.80	3.72
Cook 2	5294.8	5948.8	5560.5	6283.1	2715.50	2.94
Cooper	5952.6	2057.5	6570.1	8424.2	5967.90	3.30
Crystal River 3	8346.5	4385.3	3691.4	5333.6	7457.30	3.33
Davis Besse	5529.0	2846.6	178.0	7425.7	2126.70	2.06
Diablo Canyon 1	967.0	5295.6	5967.4	8475.7	5682.30	3.01
Diablo Canyon 2	0.0	1361.0	6857.0	6058.8	6190.70	2.33
Dresden 2	6511.4	4961.6	7110.1	5763.7	6974.90	3.57
Dresden 3	3889.0	6718.8	2756.8	7208.7	6346.30	3.07
Duane Arnold	6627.1	4733.2	7350.2	5668.3	6609.90	3.53
Farley 1	7005.8	7504.1	7276.4	8307.2	7423.30	4.28
Farley 2	8375.7	6888.1	7549.7	6537.7	8784.00	4.35
Fermi 2	0.0	0.0	869.8	5147.8	5022.50	1.26
Fitzpatrick	7087.2	5799.6	8075.8	6161.3	6060.60	3.78
Fort Calhoun 1	5386.3	6466.1	8485.2	6608.3	6510.00	3.82
Ginna	6848.7	7838.5	7716.7	8014.5	7679.20	4.34
Grand Gulf 1	1010.1	2883.4	5624.6	7203.3	8498.10	2.88
Hatch 1	5638.7	6907.5	5521.2	7191.7	6008.80	3.57
Hatch 2	3108.7	7373.1	6451.7	8519.6	6359.20	3.63
Hope Creek 1	0.0	0.0	2037.9	7570.1	7089.50	1.90
Indian Point 2	4718.4	8504.1	5101.7	6347.3	7492.10	3.67
Indian Point 3	6941.6	5901.1	6581.6	5496.5	7312.70	3.68
Kewaunee	7570.5	7266.5	7584[illegible]	7860.9	7755.60	4.34
LaSalle 1	6280.0	5757.5	239[illegible]	5609.1	5931.10	2.96
LaSalle 2	1611.8	3777.6	6613.9	4781.4	6648.20	2.67
Limerick	0.0	3420.1	6717.0	6127.0	8476.30	2.82
Maine Yankee	6688.8	7037.1	7790.8	5724.4	6949.70	3.90

Listing of Critical Hours for Each Plant
During 1984 - 1988 Period

Plant Name	Critical Hours 1984	Critical Hours 1985	Critical Hours 1986	Critical Hours 1987	Critical Hours 1988	5-Yr Total Critical Years
McGuire 1	6090.8	6842.4	5022.2	6835.7	6783.80	3.60
McGuire 2	6136.3	5490.5	5770.4	7046.9	7313.50	3.62
Millstone 1	6990.2	7324.4	8276.5	6970.7	8661.60	4.36
Millstone 2	8596.8	4460.7	6599.6	8242.0	6953.10	3.97
Millstone 3	0.0	0.0	5412.8	6350.7	7196.30	2.16
Monticello	810.6	8163.0	6984.9	7173.6	8768.70	3.64
Nine Mile Point 1	6414.0	8524.0	5823.5	8171.2	0.00	3.30
Nine Mile Point 2	0.0	0.0	0.0	1638.9	2982.30	0.53
North Anna 1	4759.9	6938.8	7560.0	4585.4	8019.50	3.63
North Anna 2	6136.0	8534.4	7301.3	6842.2	8734.90	4.28
Oconee 1	7452.4	8453.3	5948.7	6913.9	8769.00	4.28
Oconee 2	8784.0	6740.3	7253.7	8604.9	6989.20	4.38
Oconee 3	6520.7	6140.9	7835.4	6142.2	7229.70	3.86
Oyster Creek	1700.0	6818.5	2389.1	5620.0	5789.00	2.55
Palisades	1550.5	7490.2	1490.5	4226.6	4990.40	2.25
Palo Verde 1	0.0	2450.7	5112.5	4589.7	5762.90	2.04
Palo Verde 2	0.0	0.0	2217.9	6984.2	5750.00	1.71
Palo Verde 3	0.0	0.0	0.0	726.9	8201.70	1.02
Peach Bottom 2	2583.9	2910.6	7272.8	1729.8	0.00	1.65
Peach Bottom 3	7757.7	4055.7	5929.6	1823.2	0.00	2.23
Perry 1	0.0	0.0	0.0	811.3	6939.10	0.88
Pilgrim	170.3	8159.0	1715.5	0.0	0.30	1.15
Point Beach 1	6420.1	6974.4	7905.4	7389.4	7847.70	4.17
Point Beach 2	7544.2	7576.2	7262.7	7583.1	7707.80	4.30
Prairie Island 1	8321.3	7363.2	7898.1	7287.6	7835.60	4.41
Prairie Island 2	7844.0	7408.6	7972.1	8760.0	7813.90	4.54
Quad Cities 1	4766.9	8339.0	6151.3	6251.6	8477.90	3.88
Quad Cities 2	6988.6	6361.8	6448.0	6941.4	6292.80	3.77
Rancho Seco	5338.8	2874.6	0.0	0.0	5543.80	1.57
River Bend 1	0.0	0.0	4777.5	5995.1	8279.80	2.17
Robinson 2	616.1	7859.6	7118.3	6354.3	5791.40	3.16
Salem 1	2672.3	8361.9	7097.2	6412.5	6937.10	3.59
Salem 2	3386.0	5231.2	5629.4	6423.0	5992.80	3.04
San Onofre 1	888.6	6783.8	2975.3	7382.9	3817.70	2.49
San Onofre 2	5272.4	5235.8	6479.1	6192.5	8286.30	3.59
San Onofre 3	4395.2	4789.9	7402.2	7135.2	5930.80	3.38
Sequoyah 1	6206.1	3797.2	0.0	0.0	379.50	1.18
Sequoyah 2	6334.0	5289.4	0.0	0.0	5202.10	1.92
Shearon Harris 1	0.0	0.0	0.0	4449.9	6585.10	1.26
South Texas 1	0.0	0.0	0.0	0.0	2496.90	0.28
St. Lucie 1	5555.2	7134.7	8424.0	6971.6	7554.30	4.06
St. Lucie 2	7379.2	7442.7	7326.7	7382.3	8784.00	4.37
Summer 1	5553.4	6439.9	8453.2	6222.4	6067.70	3.73
Surry 1	5293.7	7935.4	6233.2	6178.3	3755.20	3.35
Surry 2	7435.3	5936.5	6171.1	6555.2	5028.30	3.55
Susquehanna 1	6549.3	5598.5	6196.3	6464.6	8289.70	3.77
Susquehanna 2	2145.9	7121.2	5946.6	8484.0	6156.90	3.41
Three Mile Island 1	0.0	2084.8	6268.6	6435.2	6760.90	2.46

Listing of Critical Hours for Each Plant
During 1984 - 1988 Period

Plant Name	Critical Hours 1984	Critical Hours 1985	Critical Hours 1986	Critical Hours 1987	Critical Hours 1988	5-Yr Total Critical Years
Trojan	4895.4	6804.7	7064.1	4730.5	5925.30	3.36
Turkey Point 3	7366.6	5405.0	6988.1	1909.7	5408.10	3.09
Turkey Point 4	5079.8	7916.8	3048.1	4503.2	5050.10	2.92
Vermont Yankee	7115.2	6297.2	4359.6	7374.6	8404.40	3.83
Vogtle 1	0.0	0.0	0.0	4048.1	6822.30	1.24
WPPSS 2	416.5	6899.7	6391.5	6199.4	6310.90	2.99
Waterford 3	0.0	1868.7	7011.6	7224.3	6624.50	2.59
Wolf Creek 1	0.0	2790.3	6523.6	6152.6	6117.60	2.46
Yankee-Rowe	6398.6	7598.3	8343.5	7248.2	7486.70	4.23
Zion 1	6319.8	5321.2	5491.0	6877.3	6723.90	3.50
Zion 2	6285.2	5909.2	7783.5	5569.7	7004.60	3.71
*** Total ***						315.17

APPENDIX L

Sources of Information

APPENDIX L
SOURCES OF INFORMATION

This study of the safety significance of balance-of-plant systems failures drew upon a broad range of information sources. The primary source for quantitative analyses was the Licensee Event Report (LER) data base sponsored by NRC and currently maintained by Oak Ridge National Laboratory. The LER data base was used to develop a study data base of BOP-related reactor trips. The development and use of the BOP data base are described, respectively, in Sections 2 and 3 of the report.

In addition, the study included an extensive review of other studies and activities by the NRC, its contractors, and nuclear industry organizations. In this appendix, synopses are provided of the documents reviewed for the study. Summaries of activities that also served as information sources are also included.

The material presented in this appendix is organized in eight subsections:

1. NUREG Reports and Inspection Reports
2. Information Resulting from NRC Requirements and Requests
3. Unresolved Safety Issues and Generic Issues
4. Maintenance Rulemaking Activities
5. The Precursor Identification Program
6. ACRS Information and Meetings
7. AEOD Activities
8. Efforts by Utilities and Industry Groups.

L.1 NUREG Reports and Inspection Reports

Following are synopses of the NUREGs and Inspection Reports reviewed in preparation for the analyses conducted in this study.

NUREG-1115, "Categorization of Reactor Safety Issues from a Risk Perspective," March 1985

NUREG-1115 reports on the results of a categorization and ranking of reactor safety issues based on risk considerations. With regard to the portions of the program relevant to BOP systems, the risk-based importance ranking was generally consistent with importance ranking of data derived from Licensee Event Reports (LERs), i.e., hardware failures and human errors were highly ranked. Other issue areas with high rankings were initiating events, system responses, and accident sequence analysis.

NUREG-1206, "Analysis of French (Paluel) Pressurized Water Reactor Design Differences Compared to Current U.S. PWR Designs," June 1986

The NRC staff identified 25 differences between the French P4 design and the U.S. SNUPPS plant, of which four to six issues are perceived as BOP-related. Three issues have a "moderate" impact on safety significance: the capability to resupply the Condensate Storage Tank, the use of self-cooled safety-related pumps, and the improvement in the DC power supply system.

NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants; Technical Findings Related to Unresolved Safety Issue A-47," Draft Report for Comment, April 1988

NUREG-1217 reports the technical findings of an evaluation of Unresolved Safety Issue (USI) A-47 concerning the safety implications of control system failures in nuclear power plants. The report concludes that with the exception of the specific events stated below, transients or accidents resulting from or caused by control system failures are less severe than, and therefore bounded by, the transients and accidents identified in the FSAR. The exceptions are reactor vessel (BWR) or steam generator (PWR) overfill events, core overheat events, and primary system overpressure events.

NRC recommendations for actions to deal with these events are given in a companion document, NUREG-1218, "Regulatory Analysis for Proposed Resolution of USI A-47, Safety Implications of Control Systems," April 1988. An evaluation of the risk implications of control system failures should therefore focus on the adequacy and implementation effectiveness of the actions recommended in NUREG-1218.

NUREG-1218, "Regulatory Analysis for Proposed Resolution of USI A-47, Safety Implications of Control Systems," Draft report for Comment, April 1988

NUREG-1218 presents the regulatory analysis related to the proposed resolution of USI A-47, "Safety Implications of Control Systems." Technical findings regarding USI A-47 are given in NUREG-1217. Although the scope of USI A-47 is quite large, the proposed resolution is quite limited, addressing primarily the need to improve protection against overfill events (steam generators for PWRs, reactor vessel for BWRs) for selected types of reactors.

Many of the events of interest to the BOP project relate to feedwater system failures, but most of these are not overfill events. Thus, the resolution of USI A-47 is of limited interest in the larger context of the risk implications of BOP failures.

NUREG-1272, "Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data - 1986," May 1987

NUREG-1272 is the 1986 annual report of the NRC's Office of Analysis and Evaluation of Operational Data (AEOD). This NUREG covers a wide range of activities, some of which parallel the BOP project. These activities include the evaluation of initiating systems of plant trips that occurred during 1984-1986, which could be used to check the methods employed for the BOP study.

NUREG-1275, "Operating Experience Feedback Report - New Plants," July 1987

Newly licensed commercial reactors have always exhibited a higher operational event frequency than mature plants. NUREG-1275 concludes that this behavior should not be accepted as inherent in the process of debugging a new plant. Early increased attention to operations, aggressive root cause analysis, enhanced training, and emphasis on BOP systems that have historically caused many events will significantly reduce new plant trips, Emergency Safety Features actuations, and violations of Technical Specifications.

NUREG/CR-3541, "Measures of the Risk Impacts of Testing and Maintenance Activities," November 1983

No information of direct applicability to the BOP study was found.

NUREG/CR-3568, "Handbook for Value-Impact Assessment," December 1983

No information of direct applicability to the BOP study was found.

NUREG/CR-3591, "Precursors to Potential Severe Core Damage Accidents: 1980-1981, A Status Report," July 1984

An evaluation of about 8400 Licensee Event Reports for 1980-1981 was performed to evaluate precursors to potential severe core damage accidents. In general, reductions in the frequency and safety significance of initiating events were observed when compared to the 1969-1979 period which was previously analyzed. A significant number of events were initiated or exacerbated by failures of BOP systems that could have resulted in severe core damage. This substantiates the importance of BOP systems in nuclear power plant safety.

NUREG/CR-3762, "Identification of Equipment and Components Predicted as Significant Contributors to Severe Core Damage," May 1984

NUREG/CR-3762 describes work performed to identify equipment and components whose failure would make a significant contribution to severe core damage probabilities, based on predictive methods (probabilistic risk assessment) and performance data (Licensee Event Reports). The results are qualitative and not directly useful in developing quantitative data on the impact of BOP failures on safety systems.

NUREG/CR-3922, "Survey and Evaluation of System Interaction Events and Sources," January 1985

NUREG/CR-3922 identifies adverse system interactions from the body of documentation available from the NRC and industry. From some 4000 events during the years 1969 to 1983, 235 were identified as adverse system interactions; these were put in 23 categories.

This document was prepared as the first phase of a project to identify and evaluate adverse system interactions for the Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants." The document draws some conclusions about the characteristics of the resultant data, but provides no conclusions about systems interactions on the whole.

NUREG/CR-3958, "Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a Combustion Engineering Pressurized Water Reactor," March 1986

Pacific Northwest Laboratory performed a study of the dominant control system failure scenarios defined by Oak Ridge National Laboratory for the Calvert Cliffs-1 nuclear power plant. This study used an existing probabilistic risk assessment to evaluate the value or impact of proposed corrective actions in reducing public risk from these postulated events. Two of the three postulated events involve failures in the main feedwater system, a BOP system. These events result in overfilling the steam generator with a potential main steam line break and steam generator tube rupture as consequences. The most promising corrective action to mitigate these events is a high steam generator level trip of the main feedwater pumps and/or feedwater block valves.

NUREG/CR-4103, "Uses of Human Reliability Analysis Probabilistic Risk Assessment Results to Resolve Personnel Performance Issues That Could Affect Safety," October 1985

No information provided in this report had direct relevance to the BOP study.

NUREG/CR-4281, "An Empirical Analysis of Selected Nuclear Power Plant Maintenance Factors and Plant Safety," July 1985

NUREG/CR-4281 examines the relationship between five maintenance program attributes and the intermediate and final safety indicators.

The five program attributes are related to the size and organization of the maintenance program and the experience of the top-level managers in the maintenance programs. The intermediate safety indicators included the number of maintenance-related LERs, Systematic Analysis of Licensee Performance (SALP) ratings, and the number of maintenance-related instances of noncompliance. The final safety indicators were all related to radiological releases and occupational exposures.

The study found a relationship between the maintenance program resources and the safety indicators. There was some indication that smaller, less hierarchical maintenance programs, with separate units for mechanical, electrical, and instrumentation and control maintenance, result in better performance as rated by the intermediate safety indicators. However, programs with combined mechanical, electrical, and instrumentation and control units tended to perform better when the final safety indicators were used.

The correlations found were not always as expected and in some cases were not in the direction expected. A possible explanation suggested by the authors is that the safety indicators may not have been complete. Suggested additions included maintenance-related trips and outages.

NUREG/CR-4314, "Brief Survey and Comparison of Common Cause Failure Analysis," June 1985

NUREG/CR-4314 presents a summary of the methods and models available for the evaluation of common cause and common mode failures. This report provides information on the general approaches to modeling common cause events, but does not deal with the causes of such events, nor does it address possible solutions. The methods identified for common cause analysis include: bounding techniques, a Beta-factor model, a Binomial Failure Rate model, a C-factor model, and common load models. Computer codes that can be used as aids to common cause analysis were also discussed.

Recommendations for the improvement of common cause analysis techniques were presented:

- Develop a standard terminology.

- Develop criteria for comparative assessments of proposed methodologies.

- Develop credible data bases designed to answer the relevant estimation questions raised by system designers, performance analysts, and decision makers.

None of the methods summarized in NUREG/CR-4314 was found to include all three of these features.

NUREG/CR-4372, "Probabilistic Risk Assessment (PRA) Applications," January 1986

NUREG/CR-4372 reports the results of a study to correlate system reliability insights from a specific PRA (Limerick Generating Station) with utility surveillance programs and test procedures, and with NRC inspection procedures. A similar program could be performed to correlate PRA risk and reliability insights with utility maintenance and surveillance testing programs for BOP systems and components. Such a study would be one way of distinguishing BOP-related risk concerns from BOP-related reliability concerns. The results could also help provide assurance that the utility maintenance and surveillance testing programs contain sufficient detail to cover the more common failure modes expected in BOP components and systems.

NUREG/CR-4385, "Effects of Control System Failures on Transients, Accidents and Core-Melt Frequencies at a Westinghouse PWR," November 1985

Although a number of control system failures (some involving the BOP) can lead to previously unanalyzed events with a risk of core-melt, the magnitude of this risk is small as compared to the overall plant risk. The relative magnitude of this risk is exaggerated due to the inherent conservatisms used for those areas in which there is considerable phenomenological uncertainty. The relative benefit of proposed corrective actions cannot be justified solely on value impact (i.e., cost-benefit). The role of the operator is crucial in reducing both the frequency and consequences of these events. The BOP components/systems were involved in steam generator overfilling and reactor coolant system (RCS) overcooling scenarios, but not in RCS overpressurization and steam generator tube rupture scenarios. The conclusions of the study pertain specifically to the Westinghouse PWR design and were obtained from modeling of the H.B. Robinson 2 power plant by the Idaho National Engineering Laboratory.

NUREG/CR-4386, Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a Babcock and Wilcox Pressurized Water Reactor," December 1985

The relevance to the BOP project is in the analysis of the plant risk caused by failures of BOP systems and components such as main feedwater (MFW) pumps and valves, or the integrated control system (ICS).

For the steam generator overfill scenarios, the initiating event is postulated as a combination of an ICS failure that causes a feedwater increase and an undetected failure of the high-level MFW pump trip. The accident is then postulated to progress to a transient or a main steam line break event, which then results in a core-melt accident.

For the ICS-related power failures which lead to overfill and undercool events, two cases were identified: (1) loss of ICS hand power, and (2) loss of ICS auto power. For case 1, it is assumed that continuous MFW pump operation at minimum speed would prevent operation of the emergency feedwater (EFW) system, since no trip signal would be generated. Steam generator dryout would occur unless the operator manually initiated the EFW system within 30 minutes or high pressure injection within 60 minutes.

For case 2, the outcome depends on whether the operator detects the ICS auto power failure before an upset condition develops. If the operator detects the failure early enough, he or she will be able to control the event before the plant trips. If the condition is not detected, the plant will eventually trip due to perturbations in the system caused by the ICS power failure.

NUREG/CR-4387, "Effects of Control System Failures on Transients, Accidents, and Core-Melt Frequencies at a General Electric Boiling Water Reactor," December 1985

The relevance to the BOP project is in the analysis of the risk to the plant caused by BOP system failures such as condensate booster pump failures that cause reactor vessel overfill, and failures that initiate reactor vessel overfill and also defeat the high level feedwater trip.

The four failure modes leading to failure of level indication and the high level trip are all related to the water level sensors or sensor circuitry.

The failure modes associated with the condensate booster pump may be summarized as follows: any of the feedwater pump discharge valves, or their bypass valves, fails open; or the condenser bypass valve used to recirculate excess condensate flow back to the condenser fails closed.

NUREG/CR-4611, "Trends and Patterns in Maintenance Performance in the US Nuclear Power Industry 1980-1985," October 1986

NUREG/CR-4611 presents an analysis of maintenance performance in the US nuclear power industry for the years 1980 through 1985. The analysis addressed the impact of maintenance practices, not the specifics of the programs that may cause the trends and patterns identified. The analysis focused on five performance categories that are directly influenced by the maintenance function: (1) overall system/component reliability, (2) overall safety system reliability, (3) challenges to safety systems, (4) radiological exposure, and (5) regulatory assessment. Trends and patterns over the 6-year period were explored. The most significant finding was that, although overall plant performance improved, the number and proportion of maintenance-related events increased. This was attributed to either a decrease in attention to this type of event by both the NRC and the nuclear industry or an actual decline in maintenance program effectiveness.

The effects of BOP systems were not explicitly identified in this report. From the information provided in the report it is not possible to separate the impact of BOP maintenance programs from the impact of safety-related system maintenance programs.

NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: 1985 A Status Report," December 1986

Of the 10 most serious severe accident precursor events identified in this study, several involved or were initiated by BOP systems/components:

- Failure of an electric pressure regulator resulted in the closure of a main steam isolation valve and reactor trip. Subsequent multiple failures resulted in equipment overheating, high reactor vessel water level, and loss of the isolation condenser function.

- Following a loss of power to safety-related buses, the failure of five check valves in the main feedwater system prevented auxiliary feedwater flow to the steam generators, caused a damaging water hammer in the feedwater piping system, and resulted in an unisolatable leak in that system.

- An auxiliary transformer cooling system-initiated trip resulted in the temporary loss of all auxiliary feedwater.

- Several reactor trips initiated by loss of main feedwater were followed by further degradation due to failures in the auxiliary feedwater system, the reactor core isolation cooling system, or the high pressure cooling system.

NUREG/CR-4783, "Analysis of Balance of Plant Regulatory Issues, January 1987"

The information in this report was used widely in the BOP project, and is reflected throughout this report.

The following NRC Inspection Reports and documents regarding inspections provided background information for the BOP study:

- Letter from R.M. Gallo, USNRC Region I, to C.A. McNeill, Jr., Senior Vice President - Nuclear, Public Service Electric and Gas Company, "Combined Inspection 50-272/87-18 and 50-311/87-20." (Salem 1 and 2), August 13, 1987.

- SECY-86-349, from V. Stello, Jr., to the Commissioners, "Balance of Plant," November 21, 1986.

- Inspection and Enforcement Manual, Temporary Instruction 2515/83, "Balance of Plant Trial Inspection Program (Feedwater System)," February 26, 1987.

- Internal NRC memo from R.P. Correia to distribution, "BOP Initiated Trips Data," (undated) CIRCA late March 1987.

- Letter from A.R. Herdt, USNRC Region II, to J.P. O'Reilly, Senior Vice President - Nuclear Operations, Georgia Power Company, "NRC Special Inspection Team Reports Nos. 50-321/87-17 and 50-366/87-17," (Hatch 1 and 2), September 2, 1987.

- Letter from J.J. Harrison, USNRC Region III, to C. Reed, Senior Vice President, Commonwealth Edison Company, (No Subject), (Re: Zion 1 and 2 BOP Inspection) November 25, 1987.

- Letter from A.R. Herdt, USNRC Region II, to W.L. Stewart, Vice President, Virginia Power Company, "NRC Special Inspection Team Reports Nos. 50-280/88-02 and 50-281/88-02" (Surry 1 and 2), March 15, 1988.

L.2 Information Resulting From NRC Requirements and Requests

Information sources in this category are Generic Letters, Bulletins, and Notices. NRC Circulars were not investigated in detail because they generally covered less significant issues and events, and the issuance of circulars was terminated in 1981.

Titles of Generic Letters, Bulletins, and Notices were reviewed to identify those with generic implications and potential relevance to the BOP study. Documents describing failures of specific pieces of BOP equipment or specific plant events were generally not selected for further review. Selected Generic Letters, Bulletins, and Notices were reviewed for input into the BOP project. Relevant documents are grouped by subject and summarized below. (Note: for Notices and Bulletins, the first two digits of the identification numbers indicate the year of issue.)

Electrical Systems - Failures/Problems (Information Notices 83-80, 84-76, 84-80, 85-28, 86-70, and 87-24)

Electrical system problems were expected to be a significant contributor to BOP challenges to safety systems. The Information Notices on this topic gave information on problems with inverters, lead acid batteries, non-nuclear instrument power, and elevated DC control voltage. As a result of these notices, some of the electrical problems have been resolved.

Instrument Air System Failures/Problems (Information Notices 81-38, 87-28, and 87-28 Supplement 1; Generic Letter 88-14)

Because of the number and persistence of instrument air problems, and because of their potential effects on safety systems, instrument air failures leading to reactor trips were flagged as an item of particular interest in the analysis of Licensee Event Reports performed during this study.

Human Error (Information Notices 84-58, 87-25)

Information Notices 84-58 and 87-25 are devoted to human error as it appears in the so-called wrong unit, wrong train, or wrong component events. Between the two notices, 15 events are described, and reference is made to an AEOD report which identified some 200 such events. Clearly, wrong unit, wrong train, and wrong component errors are relatively frequent, apply to both BOP and safety systems, and are a problem which could have serious ramifications. Human errors have been a recognized concern for some time, as evidenced by the two Generic Safety Issues on human factors.

Instrumentation and Controls (Information Notices 84-86, 85-51, and 85-89)

Information Notices 84-86, 85-51, and 85-89 cover three independent problems in the area of instrumentation and control: inadequate signal isolation, detrimental removal of fuses, and total loss of control room cooling. Of the three, only detrimental removal of fuses seems to have precedents and might be considered a generic problem. The other two incidents appear to have been isolated cases.

Six events of detrimental removal of fuses were reported to have occurred between 1981 and 1984. However, only one of the six of these human errors was on BOP equipment. This area does not seem to be a significant contributor to BOP challenges to safety systems.

Fire Protection (Information Notice 83-41)

Information Notice 83-41 describes 11 cases of fire suppression actuation causing inoperability of safety-related equipment and indicates that many other cases were reported. Additionally, the Notice extrapolates some of these events to more serious situations.

In spite of the fact that only one Information Notice has been issued on the subject, actuation of fire suppression systems can pose a serious and unpredictable challenge to safety systems. Because the systems interactions from such events are sometimes hard to identify before they occur, there are potentially many such problems existent yet undetected, many of which could challenge safety systems.

The thrust of the BOP study, however, was not to find system interactions; so, although the information in notice 83-41 is relevant, it is not applicable within the study scope.

Flooding (Information Notice 83-44)

Information Notice 83-44, titled "Potential Damage to Redundant Safety Equipment as a Result of Backflow Through the Equipment and Floor Drain System," later became part of Unresolved Safety Issue A-17, "Systems Interaction." This issue was reviewed for the BOP study and is summarized in Section 2.3 of this report.

Auxiliary Equipment (Information Notice 83-56)

Information Notice 83-56 gives the specifics of one case where the auxiliary equipment required to support operation of the Emergency Core Cooling System was too narrowly defined. This is an isolated case and has no generic conclusion applicable to the BOP study.

Service Water (Bulletin 81-03)

The increasingly wide distribution of Asiatic clams and their ability to live in freshwater piping systems, as well as the growth of mussels in saltwater systems was the topic of Bulletin 81-03. The Bulletin required licensees to look for clams and mussels and to set up ongoing programs to detect their establishment, and to eliminate them if detected.

Although this problem does have some safety significance, it is difficult to see how it could create a challenge to the safety systems. In addition, the programs required by the Bulletin should have mitigated the problem. It was concluded that this issue required no follow-up in this study.

Gas Entry into Solid Systems (Information Notice 83-77)

Entry of gas into normally solid systems has caused multiple incidents of system failures, as presented by Information Notice 83-77. Gas-bound pumps in the BOP could cause safety problems, and some situations could result in challenges to the safety systems. The Information Notice gives details on four such gas entrainment events, one of which was on a BOP system (service water). In spite of this, however, the root causes of these events seem scattered, with no apparent generic lesson to be addressed by the BOP project. Therefore, due to the limited number of reported events, and the variable root causes, there was no purpose in considering this issue further. Note, however, that gas (steam) binding of auxiliary feedwater pumps was dealt with elsewhere (Notice 84-06 and Bulletin 85-01).

Main and Auxiliary Feedwater System Problems (Information Notices 84-06, 86-14, 86-14 Supplement 1, 87-34, and 87-53; Information Bulletins 85-01 and 85-03)

These Information Notices and Bulletins point to some repetitive problems experienced in the feedwater systems, especially in auxiliary feedwater systems. The role of these systems in the BOP-related reactor trips was addressed in the analysis performed in this study.

L.3 Unresolved Safety Issues and Nuclear Generic Issues

Two Unresolved Safety Issues (USIs) and seven Nuclear Generic Issues (GIs) were identified as being related to the BOP study. These nine issues were reviewed in detail and applicable information was factored into the BOP study. The nine issues are listed below, each with a summary of how it relates to the BOP study.

USI A-17, "Systems Interactions in Nuclear Power Plants"

The class of adverse systems interactions that is the subject of USI A-17 includes, as a subset, the BOP-related plant trips or safety system degradations that are under investigation in this study. The resolution of the narrowly-defined USI A-17 is directed toward the risk implications of specific external events, i.e., earthquakes and floods. The documentation generated during the NRC/industry investigation of USI A-17 did not provide significant BOP-related insights beyond those obtained from examination of the LERs that reported on significant systems interaction events.

USI A-47, "Safety Implications of Control Systems"

Documentation related to the investigation and resolution of USI A-47 included two NUREG reports (1217 and 1218) and four NUREG/CR reports (3958, 4385, 4386, and 4387). The resolution of USI A-47 focused almost exclusively on the adequacy of steam generator (PWR) and reactor vessel (BWR) overfill protection, with implications for reactor vessel damage, steamline break, or steam generator tube rupture events. The risk-related information in the USI A-47 documents reviewed was reviewed concerning the estimation of the effects on public risk of BOP challenges to safety systems.

GI-23, "Reactor Coolant Pump Seal Failure"

GI-23 considered the causes and effects of reactor coolant pump (RCP) seal failure and concluded that station blackout was the only probable event that could cause a RCP seal failure severe enough to result in leakage equivalent to a small-break loss of coolant accident. Hence, the resolution of this issue was tied to station blackout (Unresolved Safety Issue A-44).

The LER search done for the BOP study identified station blackout events, but they were not included in the BOP data base initiating event for the blackout was onsite, e.g., transformer failure.

GI-65, "Component Cooling Water System Failures"

Generic Issue 65 identified reactor coolant pump seal failure in PWRs as the primary safety concern resulting from total loss of component cooling. This issue was therefore absorbed by GI-23, "Reactor Coolant Pump Seal Failure." Consideration of GI-23 as part of this study covered all issues raised by GI-65.

GI-93, "Steam Binding of Auxiliary Feedwater Pumps"

GI-93 raised a potentially serious BOP issue. The problem was effectively solved in 1985 by increased operator surveillance of the auxiliary feedwater line temperature and is no longer a regulatory or technical issue. The BOP study therefore did not consider GI-93, since steam binding of the auxiliary feedwater pump should not be a future concern.

GI-122, "Miscellaneous Feedwater Issues"

GI-122 principally investigated the reliability of auxiliary feedwater systems. It was concluded that plants with a two-train auxiliary feedwater system were the most vulnerable. On a plant-by-plant basis, each of the seven two-train plants was evaluated and recommendations for changes were made and implemented.

Since the reliability of the auxiliary feedwater systems had recently been evaluated and the least reliable systems improved, the BOP study did not concentrate on the auxiliary feedwater system.

GI-130, "Essential Service Water Pump Failures at Multi-Plant Sites"

GI-130 deals with a narrow problem which can occur only in a small population of plants. The methods and ideas employed for the investigation and resolution of this issue were found to be of little relevance to the BOP study since the conditions analyzed are probabilistic. An event of the kind considered in GI-130 has never occurred.

GI-HF-01, "Human Factors Program Plan"

The Human Factors Program Plan provides a definition of perceived weaknesses in the human factors engineering of nuclear power plants, goals to correct those weaknesses, and outlines of how to achieve those goals. Reduction of human errors was of interest to the BOP study and was considered by evaluating trends in human-error-related trips.

GI-HF-02, "Maintenance and Surveillance Program Plan"

Investigations for the Maintenance and Surveillance Program Plan served as a source for the maintenance policy and the proposed rule on maintenance program effectiveness. Since the area of maintenance, as it applies to the BOP, is believed to be an important part of the problem of BOP challenges to safety systems, these efforts toward improving maintenance may be part of the overall solution to the problem at which the BOP study was aimed. Hence, this Generic Issue, the documents prepared for it, and, perhaps most importantly, the proposed rule on maintenance programs, were all considered in the study.

L.4 Maintenance Rulemaking Activities

The development of the Maintenance Rule was reviewed from its inception as the Maintenance and Surveillance Program Plan in 1985, through the proposed Rule on Maintenance (November 1988) and its subsequent deferral in May 1989.

As part of the monitoring of the maintenance rulemaking, BOP project personnel attended the NRC-sponsored workshop in November 1988, as well as a meeting of the Advisory Committee on Reactor Safeguards (March 1989) and an NRC meeting (May 1989) on this topic. The proceedings of the workshop were published as NUREG/CP-0099.

The maintenance rulemaking efforts are the result of NRC concern that inadequate maintenance on the part of some licensees is compromising safety. The industry has resisted NRC attempts at rulemaking in this area, citing industry improvements through self-regulation.

Action on the proposed rule was deferred as of May 1989 to allow for further study and monitoring of industry progress. Initially, the proposed maintenance rule was to cover essentially all BOP systems; the final rule, as proposed by the NRC staff, was somewhat restricted in scope, but still relevant in its implications for the BOP systems. All licensees were to have a maintenance program with certain broadly stated attributes.

L.5 Precursor Identification Program

The Accident Sequence Precursor Program was examined, principally as it was presented in NUREG/CR-3591 and NUREG/CR-4674. The results of this effort are discussed in Section 5.2 of this report.

L.6 ACRS Information and Meetings

The activities of the Advisory Committee for Reactor Safety (ACRS) and its subcommittees were monitored for BOP-related information, which was utilized in this study when it was applicable. The former ACRS subcommittee on BOP systems has been discontinued, with its functions picked up by the Subcommittee on Secondary Systems. Other subcommittees monitored were:

- AC/DC Power Systems Reliability
- Auxiliary Systems
- Human Factors
- Instrumentation and Control Systems
- Maintenance Practices and Procedures
- Reliability Assurance
- Systematic Assessment of Experience.

L.7 AEOD Activities

Relevant activities for the Office of Analysis and Evaluation of Operational Data (AEOD) were monitored, primarily by reviewing AEOD reports. Many of the applicable reports are periodic reports (quarterly, annual, etc). Two of these, AEOD/P503 and AEOD/P504, are summarized below as examples:

AEOD/P503, "Engineered Safety Feature Actuations At Commercial United States Nuclear Power Reactors January 1 Through June 30, 1984"

AEOD/P503 documents an analysis of Licensee Event Reports of ESF actuations. Many of the events reported involved or were influenced by failures in BOP systems. One of the four problem areas identified from this study that is of safety significance, involving the BOP, is component cooling water system interaction.

AEOD/P504, "Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1984"

This report presented findings that were very relevant to the BOP study. It indicated that in 1984, about 59 percent of reactor trips above 15 percent power were related to BOP systems: feedwater (27 percent), turbine (15 percent), condensate (6 percent), main generator (6 percent), main steam (5 percent). It was also identified that 71 percent of the trips between 2 percent and 15 percent power level were associated with BOP systems: Feedwater (40 percent), turbine (18 percent), and main steam (13 percent). Most of these trips were caused by hardware failure in BOP systems.

L.8 Efforts by Utilities and Industry Groups

Numerous utilities and industry groups have reactor trip reduction programs, almost all of which have a BOP component. Information about these programs was evaluated for the BOP study. Four categories of programs were evaluated: utility programs, NUMARC/INPO/EPRI programs, Owners Group programs, and international programs.

L.8.1 Utility Programs

NUREG/CR-4783, "Analysis of Balance of Plant Regulatory Issues," contains a detailed discussion and comparison of reactor trip reduction programs or performance/reliability improvement programs at six U.S. utilities. No additional information on individual utility programs was reviewed; rather, composite information from sources such as INPO, NUMARC, and the Owners Groups was emphasized.

L.8.2 NUMARC/INPO/EPRI

The Nuclear Management and Resource Council (NUMARC) has established quantitative scram reduction goals for the industry. The original goal was set in 1984 and there have been yearly revisions downward since. Establishing the methods to achieve the goals and tracking the results was left to INPO and the Owners Groups.

The Institute for Nuclear Power Operations (INPO) has defined areas for specific scram reduction efforts, as detailed in INPO-85-011, "Scram Reduction Practices," May 1985. The areas are: administrative, system/design, maintenance, surveillance testing, and operations.

In the administrative area, three efforts were urged: improved communication between plants; improved quality of root cause evaluations, and augmented sharing of ideas to reduce reactor trip frequency. In the system design/modification category, three goals were defined: identify common design problems that are reactor trip root causes, identify possible solutions to the design problems, and focus on feedwater-related trips. The maintenance goal is to initiate activities to reduce on-line and outage maintenance errors. Reducing surveillance test errors was the stated goal in the surveillance category, and for operations, two goals were set: identify human factor root causes of trips and identify possible solutions.

The involvement of the Electric Power Research Institute (EPRI) in the effort to reduce scram frequency has been primarily the production of a series of studies of the effects on scram frequency of trip setpoint modifications. Four reports were generated:

- "Reducing Scram Frequency by Modifying Reactor Setpoints for a Westinghouse 4-Loop Plant," NSAC/94, April 1986.

- "Reducing Scram Frequency by Modifying Reactor Setpoints for a Westinghouse 3-Loop Plant," NSAC/99, December 1988.

- "Scram Reduction by Relaxing Setpoints, An Analysis of C-E PWR's with Digital Controls Using RETRAN-02," NSAC/93, January 1986.

- "Scram Reduction by Relaxing Setpoints, An Analyses of C-E PWRs with Analog Controls Using RETRAN-02," NSAC/92, November 1985.

L.8.3 Owners Group Programs

The four nuclear steam supply system owners groups are very active in BOP-related activities. Babcock and Wilcox (B&W) Owners Group activities include the Safety and Performance Improvement Program (SPIP), the Trip Reduction and Transient Response Improvement Program, and the Comparative Study. Owners of B&W plants are implementing the recommendations of these programs through plant-specific modifications aimed at reducing the frequency of reactor trips.

The Combustion Engineering Owners Group (CEOG) consists of a steering group and several technical subcommittees. The Scram Reduction Program, initiated in May 1985, is being implemented through the Operations Subcommittee. The major thrust of this program is not in the balance of plant. The only significant BOP activity involves an open interchange of information between the member CE utilities at meetings in which BOP-related trips, root causes, and experience with corrective actions are discussed. The consensus among CE utilities is that the major contributor to BOP trips is feedwater system malfunction, caused by either equipment failure or human error. Florida Power and Light has installed the Combustion Engineering digital feedwater control system at St. Lucie and has reported excellent results in terms of low power feedwater control, which is one problem area causing frequent trips. Southern California Edison is looking into reducing the steam generator low-level trip setpoint by advanced analyses, anticipating that a setpoint reduction would help reduce feedwater-associated trips. Combustion Engineering is preparing specific proposals to the CEOG regarding reduction in the frequency of trips, which will include consideration of BOP systems and components.

The General Electric Boiling Water Reactor Owners Group (BWROG)is conducting a Scram Frequency Reduction Program (SFRP), which is subdivided into three areas -- operations, systems design, and maintenance. The operations group is establishing a data base on both automatic and manual reactor trips, with root cause information, and is examining the question of how best to perform effective root cause evaluations. The systems design group is responsible for maintaining the reactor trip data base, for suggesting improvements in root cause evaluations and related training, and for trend analysis of the data base. The maintenance group is examining maintenance-related contributions to reactor trips. Plans for the SFRP include identification of the most critical BOP components in terms of trips and investigating why some BWRs are more resistant to trips than others, given the same BOP component failure.

The Westinghouse Owners Group is conducting a Trip Reduction and Assessment Program (TRAP). Based on operating experience, the initial emphasis is being placed on the feedwater control system configurations at low power and on the steam generator low-level trip setpoints. Analyses indicated that a steam generator level trip modification that included sensor inputs for containment temperature and pressure could allow up to a 15 percent widening of the level trip band. Feedwater and steam generator level trip modifications have been made on several Westinghouse plants. The TRAP also includes examination of turbine-generator and control systems, electrical systems, maintenance issues, and detailed categorization of the root causes of trips.

L.8.4 International Programs

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA/OECD) conducted a symposium on scram reduction in Tokyo, Japan, in April 1986. The proceedings of this symposium, "Reducing the Frequency of Nuclear Reactor Scrams," were reviewed for the BOP study. Ten countries participated: Belgium, Canada, France, West Germany, Great Britain, Italy, Japan, Spain, Sweden, and the United States. All of the types of commercial nuclear power plants found among the OECD nations were represented. Each country offered its experience, analysis, and philosophy on scrams and scram reduction, including detailed scram statistics. Other papers were given which outlined the process of scram cause identification and correction, including the resulting design improvements. The Germans made a presentation on their instrumentation and control system which helps keep the German scram rate to around one per reactor year. The Japanese described improvements they have made to keep their scram rate to a similar low number, which was approximately one-fifth of the scram rate of the U.S. in 1985. Automatic testing devices, scram setpoint changes, operator and maintenance training, preventive maintenance, and improvements in design and construction were all discussed as methods to reduce scram frequency.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report documents the results of an analysis of balance-of-plant (BOP) related reactor trips at commercial U.S. nuclear power plants over a 5-year period, from January 1, 1984, through December 31, 1988. The study was performed for the Plant Systems Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The objectives of the study were:

1. to improve the level of understanding of BOP-related challenges to safety systems by identifying and categorizing such events;

2. to prepare a computerized data base of BOP-related reactor trip events and use the data base to identify trends and patterns in the population of these events:

3. to investigate the risk implications of BOP events that challenge safety systems;

4. to provide recommendations on how to address BOP-related concerns in a regulatory context.

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