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Survey and Evaluation of System Interaction Events and Sources

Main Report and Appendices A and B

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This report describes the first phase of an NRC-sponsored project that identified and evaluated system interaction events that have occurred at commercial nuclear power plants in the United States. The project included (1) an assessment of nuclear power plant operating experience data sources, (2) the development of search methods and event selection criteria for identifying system interaction events, (3) a review of possible events, and (4) a final evaluation and categorization of the events. The report, organized in two volumes, outlines each of these steps and presents the results of the project. Volume 1 contains an introduction to the project, describes the process by which the project identified and evaluated the system interaction events, and presents the results and recommendations from that evaluation. Volume 1 also contains appendixes that review the data sources used in identifying events and outlines the information collected for each event. Volume 2 provides a description of each adverse system interaction event and lists the references for the events.

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FOREWORD

The work reported here was undertaken by the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory on behalf of the Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC). The Technical monitor for the project was Dale F. Thatcher of the NRR Generic Issues Branch.

LIST OF TABLES

LIST OF ACRONYMS AND ABBREVIATIONS

ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AE	architect-engineer
AEOD	NRC Office for Analysis and Evaluation of Operational Data
AFW	auxiliary feedwater (system)
ANO 1	Arkansas Nuclear One Unit 1
AO	abnormal occurrence
ASI	adverse system interaction
ASP	Accident Sequence Precursor
BAST	boric acid storage tank
B&W	Babcock and Wilcox Company
BWR	boiling-water reactor
CCW	component cooling water
CE	Combustion Engineering
CER	Construction Event Report
CO_2	carbon dioxide
CR	control room
CRD	control rod drive
CST	condensate storage tank
CVCS	chemical and volume control system
DBA	design basis accident
dc	direct current
delta-P	differential pressure
delta-T	differential temperature
DG	diesel generator
ECCS	emergency core cooling system
ESF	engineered safety features
ESFAS	engineered safety features actuation system
FEF	Foreign Event File
FM	frequency modulation
FW	feedwater
GI	generic issue
HELB	high-energy line break
HPCI	high-pressure coolant injection (system)
HPSI	high-pressure safety injection (system)
HTGR	high-temperature gas-cooled reactor
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and controls
ICS	integrated control system
IE	Inspection & Enforcement
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
I/PP	current/pressure
IPRDS	In-Plant Reliability Data System
kV	kilovolt
LER	licensee event report
LOCA	loss-of-coolant accident
LOP	loss of (electric) power

LOSP	loss of offsite (electric) power
LPCI	low-pressure coolant injection (system)
MCC	motor control center
MG	motor-generator
MFLB	main feedwater line break
MFW	main feedwater
MSLB	main steam line break
NaOH	sodium hydroxide
NNI	nonnuclear instrumentation (system)
NOAC	Nuclear Operations Analysis Center
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
NSSS	nuclear steam supply system
ORNL	Oak Ridge National Laboratory
PORV	power-operated relief valve
PRA	Probabilistic Risk Assessment
psig	pounds per square inch gage (pressure)
PWR	pressurized-water reactor
RAS	recirculation actuation signal
RB	reactor building
RCIC	reactor core isolation cooling (system)
RCP	reactor coolant pump
RCS	reactor coolant system
Recirc	recirculation; also recirculating water system
RHR	residual heat removal (system)
RPS	reactor protection system
RWST	refueling/borated water storage tank
SCSS	Sequence Coding and Search System
SDV	scram discharge volume
sec	secondary
SFAS	safety features actuation system
SG	steam generator
SGTS	standby gas treatment system
SI	safety injection
SIAS	safety injection actuation signal
SW	service water (system)
TMI	Three Mile Island
USI	unresolved safety issue
V	volt
V ac	volts alternating current
VCT	volume control tank
V dc	volts direct current
WNP	Washington Nuclear Project
WPPSS	Washington Public Power Supply System

EXECUTIVE SUMMARY

This report describes a project sponsored by the Nuclear Regulatory Commission (NRC) Generic Issues Branch and performed at the Oak Ridge National Laboratory. The project, conducted by the Nuclear Operations Analysis Center, was structured to identify system interaction events that have occurred at commercial nuclear power plants in the United States. Although previous studies have examined system interaction events, the thrust of this study was to provide a broad review of nuclear power plant operating experience using a specific definition of a system interaction.

Initially, the project selected over 4000 events for review from reactor experience data. A detailed review and evaluation reduced the 4000 events to 235 events that were considered adverse system interaction events. For these 235 adverse system interaction events, enough information was collected to allow further analysis. This information included items such as date of event, systems and components involved, method of discovery, and corrective action. Statistics from these attributes for each event are presented and discussed in the report.

The 235 events were placed into 23 categories using the data collected on each event. These categories contain events that are similar in some aspect and provide insight into the kinds of system interaction events that have occurred. The report describes each of the categories and discusses their significance. Examples of the categories are listed below:

1. adverse interactions between normal or offsite electric power and emergency electric power systems,
2. degradation of safety systems by vapor or gas intrusion,
3. degradation of safety-related systems by fire protection systems, and
4. flooding of safety-related equipment through plant drain systems.

In addition to drawing attention to the specific categories of system interaction events, the project made two recommendations for continued effort:

1. The safety significance of each of the categories should be examined, with emphasis on the potential for continuing problems.
2. Current system interaction analysis methods should be evaluated to examine their effectiveness in identifying the kind of system interaction events reflected in the operating experience.

Detailed evaluation of safety significance is a complex problem and was not within the scope of the project. It will require (1) an examination of all of the industry and NRC actions that have occurred in response to the events and (2) an assessment of how effective these actions have been. The second recommendation is being addressed in phase II of this project. That phase will assess system interaction analysis techniques, using in part the adverse system interaction events and categories discussed here.

SURVEY AND EVALUATION OF SYSTEM INTERACTION EVENTS AND SOURCES

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ABSTRACT

This report describes the first phase of an NRC-sponsored project that identified and evaluated system interaction events that have occurred at commercial nuclear power plants in the United States. The project included (1) an assessment of nuclear power plant operating experience data sources, (2) the development of search methods and event selection criteria for identifying system interaction events, (3) a review of possible events, and (4) a final evaluation and categorization of the events. The report, organized in two volumes, outlines each of these steps and presents the results of the project. Volume 1 contains an introduction to the project, describes the process by which the project identified and evaluated the system interaction events, and presents the results and recommendations from that evaluation. Volume 1 also contains appendixes that review the data sources used in identifying events and outlines the information collected for each event. Volume 2 provides a description of each adverse system interaction event and lists the references for the events.

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1. INTRODUCTION

1.1 Background

Safety (with regard to radiological releases and exposure of the general public and plant personnel) is of great importance to the nuclear power industry. To ensure this safety objective is met, numerous specialized systems are included in the design of nuclear power plants. The purpose of these "safety" systems is to mitigate accidents and minimize their consequences. Therefore, these systems must be reliable. (Note: The terms *safety* system and *safety-related* system are used interchangeably in this report.)

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To ensure the reliability of the safety systems, thorough reviews and evaluations are performed on all facets of the systems' operation. However, experts often question the completeness of the current review process, for the following reasons:

1. The plant reviews are frequently done on a system-by-system basis rather than being integrated over the many systems that function together.
2. The complexity of the systems makes comprehensive reviews difficult.
3. System design may not always take into account all parameters needed for operation.
4. Good communication among the many different specialists (e.g.,chemical, mechanical, civil, and electrical engineers) involved in the design and construction of these systems is difficult to achieve and maintain.

These factors can lead to design flaws. A major area of concern is unidentified interactions and dependencies between systems, in particular, redundant safety systems.

In 1974, the ACRS identified a generic need to examine the matter of "system interactions" — the unidentified (and possibly unanalyzed) dependencies between systems. In 1978, the NRC began a system interaction program by defining USI A-17, "Systems Interaction in Nuclear Power Plants," and initiated several programs to investigate the issue.[1,2]

1.2 Purpose

The objective of this project was to identify and evaluate possible system interaction events that have occurred at commercial nuclear power plants in the United States. This work was performed in support of the Task Action Plan developed by the NRC to address USI A-17. The results from this review of operational experience include (1) insights into the system interaction issue, (2) categories of system interaction events, and (3) data for use in reviewing system interaction analysis methods. This information will be useful in regulatory decisions concerning threats to safety by unanalyzed system interactions.

Phase II of this project (to be completed in FY 1985) will evaluate current search methods that are used to find potential adverse system interaction events. This evaluation will consider the effectiveness of the methods for finding adverse system interaction events and an estimate of costs involved. The results will then be used in the development of guidelines for search methods.

1.3 System Interaction Definition

In establishing this project, the NRC Generic Issues Branch provided the following system interaction definition, which was used as the basis for all project activities:

A system interaction occurs when an event in one system, train, component or structure *propagates* through *unanticipated or inconspicuous dependencies* to cause an action or inaction in other systems, trains, components or structures.

The definition contains three major points used for identifying system interactions: (1) initiating event, (2) propagation, and (3) unanticipated or inconspicuous dependencies. The *initiating event* can be a failure, action, or inaction of a system, train, component, or structure. This initiating event then *propagates* through *unanticipated or inconspicuous dependencies* to adversely affect at least one other system, train, component, or structure.

Of the events that satisfied the system interaction definition, the project focused on a subset — "adverse system interactions." An adverse system interaction satisfies the above definition but also has one or more of the following undesirable results:

1. degradation of redundant portions of a safety system, including consideration of all auxiliary support functions (redundant portions are those considered to be independent in the design and analysis of the plant);
2. degradation of a safety system by a nonsafety system;
3. initiation of an "accident" (e.g., LOCA, MSLB) *and* (a) the degradation of *at least one* redundant portion of any one of the safety systems required to mitigate the event; *or* (b) the degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action;
4. initiation of a "transient" (including reactor trip), *and* (a) the degradation of *at least one* redundant portion of any one of the safety systems required to mitigate the event; *or* (b) the degradation of critical operator information sufficient to cause him to perform an unanalyzed, unassumed, or incorrect action;
5. initiation of an event that (a) requires actions by the plant operators in areas outside the control room *and* (b) disrupts the access to these areas.

The ASI events are divided into three classes.

1. Functionally coupled: Those ASI events that result from sharing of common systems or components; or physical connections between systems including electrical, hydraulic, pneumatic, or mechanical connections.
2. Spatially coupled: Those ASI events that result from sharing of common structures, locations, or spatial ties such as HVAC and drain systems.
3. Induced-human-intervention coupled: Those ASI events where (a) a plant malfunction (such as failed indication) inappropriately induces an operator action or (b) a plant malfunction requires an operator action, and inhibits the operator's ability to respond. (Induced-human-intervention coupled ASI events exclude random human errors and acts of sabotage.)

1.4 Organization of Text

Chapter 2 contains a summary of the results of this project. Sources of event information and the process used in examining events are described in Chap. 3. In Chap. 4, the events chosen as adverse system interaction events are reviewed.

Appendix A lists the sources of events used by the project and gives an evaluation of each source. Event attributes are defined in Appendix B. In Appendix C, events chosen as adverse system interaction events are listed. Appendix D contains a list of references for the events in Appendix C.

2. SUMMARY OF RESULTS AND RECOMMENDATIONS

2.1 Summary of Results

The project surveyed and assessed relevant sources of operating event information and developed screening methods and criteria to identify system interaction events (as defined in the Task Action Plan for USI A-17). Over 4000 events were initially screened; of these, 235 events were selected as adverse system interaction events. Data were collected for each event for further analysis. A review of the characteristics of the ASI events revealed the following:

1. Sixty percent of the 235 events were reported as potential problems rather than actual operating experience events.
2. Over half (57%) of the ASI events involved functional dependencies. Most of these were between systems that normally interact with one another. However, the events considered ASIs in this study represent unanticipated dependencies for these systems.
3. Over half of the spatial events (41% of the total ASI events) were caused by harsh environmental conditions (high humidity, high temperature, and flooding). These include both actual and potential events.
4. The number of ASI events reported per year (both actual and potential) peaked in 1980. This is most likely a result of post-TMI modifications, requirement changes, and increased design reviews.
5. One-third of the ASI events involved degradation of safety-related equipment by non-safety-related equipment.

These observations provide general information about the types of adverse system interactions identified by this project. The 235 events do not represent all ASI events that have occurred but are the product of a systematic examination of operating experience. As such, the trends above are useful in evaluating system interaction problems.

As part of the data evaluation effort, the project staff also compared the 235 ASI events for commonalities. This allowed grouping of the events into 23 categories (Table 2.1). The number of events in each category is given in the table; no event was placed in more than one category. Each category contains events that share a predominant trait.

Evaluating the safety significance of each category of events, or of individual events, was not included in the scope of this project. However, future work will address qualitative and quantitative assessments of the safety significance of each of the categories.

Each category represents sources of intersystem dependencies that have degraded the level of redundancy required for safety systems.

Certain categories (categories 5, 7, 8, 10, 11, and 17) represent generic problems because they involve specific design problems that were reported for a number of plants. Also, some of the categories parallel areas of concern identified in certain unresolved safety issues and generic issues. Section 4.2 discusses each category in detail.

Table 2.1. Event categories

Category No.	Title	Number of events
1	Adverse interactions between normal or offsite power systems and emergency power systems	34
2	Degradation of safety-related systems by vapor or gas intrusion	15
3	Degradation of safety-related components by fire protection systems	10
4	Plant drain systems that allow flooding of safety-related equipment	8
5	Loss of charging pumps due to volume control tank level instrumentation failures	6
6	Inadvertent ECCS/RHR pump suction transfer	4
7	HPSI/charging pumps that overheat on low flow during safety injection	6
8	Level instrumentation degraded by high energy line break (HELB) conditions	21
9	Loss of containment integrity due to LOCA conditions during purge operations	10
10	HELB conditions degrading control systems	3
11	Auxiliary feedwater pump runout under steam line break conditions	2
12	Water hammer events	4
13	Common support systems or cross-connects	18
14	Instrument power failures affecting safety systems	5
15	Inadequate cable separation	8
16	Safety-related cables unprotected from missiles generated from HVAC fans	3
17	Suppression pool swell	3
18	Scram discharge volume degradation	2
19	Induced human interactions	4
20	Functional dependencies due to failures during seismic events	5
21	Spatial dependencies due to failures during seismic events	13
22	Other functional dependencies	21
23	Other spatial dependencies	30

2.2 Recommendations

The project recommends that the categories identified here be used for two purposes:

1. evaluation of the safety significance of system interactions that have been reported, and
2. examination of system interaction analysis methods to determine their effectiveness.

Evaluating the safety significance of the categories should focus on the potential for the problems to continue to occur. It was recognized that in many instances, both the affected plant and the other licensee plants have already made design changes. In general, these changes were initiated by individual licensee programs, industry working group actions, or NRC licensing actions. The project collected information about such activities (primarily NRC documents) pertaining to each category. This information is presented in Sect. 4.2 where each category is discussed. Assessing the corrective actions in response to these activities is a major part of evaluating the safety impact of each category.

Phase II of this project will address the second recommendation. The categories of events provided by Phase I of this work will be used in evaluating system interaction analysis methods. Each method will be examined to determine if its approach (scope, level of detail, assumptions, etc.) is consistent with identifying the types of system interactions found in operating experience. The project staff recognizes, however, that focusing on events that have occurred, or have been postulated to occur, may not adequately address all types of system interactions.

3. EVENT SELECTION

3.1. Data Source Evaluation

Numerous sources of operating experience data exist, including individual event reports sent to the NRC, component failure data collected by other agencies, topical reports, etc. To begin the process of selecting events, the project staff examined several data bases and reviewed a number of documents that contained operating experience data. Specific sources examined were: (1) operating experience data bases, (2) system interaction methodology reports, (3) system interaction application reports, and (4) other reports describing significant operating events. Some of these documents did not contain specific operating experience data; however, their system interaction definitions, screening criteria, and lessons learned were helpful.

Each source was evaluated on its accessibility, completeness, type of data contained, and usefulness of the data. After evaluating these data sources, the following were selected:

1. Licensee Event Report (LER) file;
2. Sequence Coding and Search System (SCSS) file;
3. Foreign Event File (FEF);
4. Construction Event Report (CER) file;
5. bulletins, notices, and circulars issued by the NRC Office of Inspection and Enforcement (IE);
6. analysis reports on special operating events performed by the NRC Office for Analysis and Evaluation of Operational Data (AEOD);
7. other reports providing reviews of significant operating events; and
8. NRC reports to Congress on abnormal occurrences (NUREG-0090 series).

A summary description of the sources is given below. A detailed assessment of each source is given in Appendix A.

3.1.1 Operating experience data bases

A number of data bases contain nuclear power plant operating experience information. The project examined six such data bases:

1. the Licensee Event Report (LER) file, which contains abstracts of all LERs for U.S. nuclear power plants;[3]
2. the Sequence Coding and Search System (SCSS) file, which also contains LER data but with much more detail for events from January 1981 to date;[4]
3. the Foreign Event file, which contains abstracts of selected events from foreign nuclear power plants that are considered proprietary;[5]
4. the Construction Event Report (CER) file, which contains construction deficiency reports filed by nuclear power plants that do not have operating licenses;[6]

5. the Nuclear Plant Reliability Data System (NPRDS), which contains component failure data supplied by utilities with operating nuclear power plants;[7] and
6. the Inplant Reliability Data System (IPRDS), which contains maintenance and repair data for specific equipment from six selected nuclear power plants.[8]

Of these six operating experience data bases, the LER, SCSS, and CER files were used as sources of operating events. The LER and SCSS files were chosen because they contain the most detailed event descriptions and cover the largest number of events. The CER file was chosen because it contains construction deficiency reports. Using these three files, data for plants in both the construction and operation phases were collected. Because the foreign event file is proprietary and none of the data can be released (without the consent of the NRC Office of International Programs), it was of limited use. However, a review of its significant events helped focus the search effort for similar events in domestic experience. These four data bases were readily accessible to the project staff. The NPRDS and IPRDS files were not utilized because they did not provide the information necessary for this project (i.e., plant name, event date, system information, etc.).

Sections A.1 through A.5 in Appendix A contain a detailed assessment of each of these data bases.

3.1.2 System interaction methodology reports

Several reports written in the past 10 years have proposed and reviewed methods for analyzing systems to identify system interactions. Evaluation of these reports found that the methods were directed toward analysis of plant systems in conjunction with a detailed systems analysis, such as probabilistic risk assessment. Thus, they are not easily adapted for the analysis of *event* data. However, these reports provided excellent background material and were helpful guides during the development of screening criteria for events. Section A.6 in Appendix A lists the methodology reports that were evaluated and provides remarks regarding each.

3.1.3 System interaction analysis application reports and related material

The project staff reviewed a number of reports documenting system interaction analyses performed on commercial nuclear power plants. The staff also reviewed letters and related documents issued by the ACRS pertaining to the system interaction issue. System interaction has been a major concern of the ACRS. These reports, letters, and related documents provided some event data but were of more use during the development of screening criteria for event selection. Section A.7 in Appendix A lists the material reviewed.

3.1.4 Reports describing significant events

Several sources of information on significant events were examined. Although these sources are not specifically concerned with system interactions, some of the documented events were considered significant to this effort because, in many cases, they involved intersystem dependencies. The sources that the project found useful included (1) NRC Office of Inspection and Enforcement bulletins, circulars, and information notices; (2) AEOD case studies and engineering evaluations of significant events; (3) reports evaluating selected events, published by the ORNL NSIC; and (4) reports to Congress on abnormal occurrences (NUREG-0090 series). Sections A.8 through A.13 in Appendix A provide more information about each of the significant event sources evaluated.

3.2. Selection of Events

Of the data sources selected, the LER file was the largest and most extensive source of operating experience data (data are available from 1969 through the present). Therefore, it provided the bulk of the data selected for this project. This file contains abstracts for each LER (and any subsequent updates) sent to the NRC by the utilities. The primary method for selecting data from this file is through "keyword" searches. Keywords are predefined attributes that are assigned to each LER when it is added to the LER file. The time period for the searches was restricted to events prior to January 1, 1984. Because all 1984 LERs were not yet available during the project, it was decided to defer examination of the 1984 events until a later phase of the project.

Screening efforts for identifying events focused primarily on events that involved common-cause failures, reactor transients, safety injections, and other complex events. All events identified as potentially significant by previous studies of operating event data were reviewed; for example, all events reviewed in the Accident Sequence Precursor Program were assessed from a system interaction standpoint. By focusing the screening efforts in these areas, there was a greater chance of finding system interaction events that involved safety-related systems. Table 3.1 summarizes the specific searches and the number of events selected by each. The strategies for the searches were structured so that events selected by a specific search would not appear in any other search and would prevent duplicate review efforts.

In addition to the LER file, the SCSS and CER data bases were also searched. The SCSS file contains LER data from 1981 to the present. Because all LERs contained in the SCSS file are also contained in the LER file, only events designated as "significant" or "complex" in SCSS (a total of 231 events) were selected for review. The CER file contains construction deficiency reports. A total of 254 data records (the entire file that was available at the time) were reviewed.

In addition to the data sources addressing individual events, the project reviewed each of the reports describing significant events (Sect. 3.1.4). If an IE bulletin, IE information notice, IE circular, AEOD report, NSIC report, or Abnormal Occurrence report described a system

Table 3.1. Searches conducted on LER abstract file using keywords

Subject defined by keywords	Number of abstracts reviewed
Common-mode failures	287
Events involving failures of redundant equipment	288
Events evaluated in the Accident Sequence Precursor Program[a]	938
Events involving accidents (loss of cooling accidents, control rod ejection accidents, and other design basis accidents)	828
Transients	675
External events (fire, flood, severe weather, earthquake, and explosions)	813
Other potentially significant events from previous NSIC studies	121
Additional events from supplemental searching after initial category identification	500

[a]This includes the ASP events from 1969 through 1981.

interaction event, the project staff added the event to the file for further evaluation. Multiple sources for an event were collected to provide additional information.

In total, the project staff initially reviewed more than 4000 events from the sources discussed above. From these events, ~400 events were selected for detailed review. The detailed review (described in Sect. 3.3) reduced this group to 204 events considered ASI events. To provide a more thorough search for ASI events, the project used the characteristics of the 204 events as a guide in searching the data bases a second time. This effort netted another 500 events for detailed review. The detailed review resulted in identification of an additional 31 ASI events, for a total of 235 events.

3.3 Screening and Processing of Events

After searching the data sources and identifying possible system interactions, the project staff then thoroughly reviewed the events to identify system interaction events. The following criteria were used:

1. Did the event involve, or have the potential to involve, a safety system?
2. Did the event involve combinations (two or more) of systems, trains, components, or structures?
3. Did a propagation of actions or inactions occur?
4. Were any of the interactions or dependencies that occurred unanticipated in that the plant design or plant procedures did not compensate for them?

Those events that satisfied all of the above criteria were deemed ASI events if they caused an undesirable result to occur. (Definitions of types of undesirable results are given in Sect. 1.3).

Events selected as adverse system interactions were closely reviewed and information for further analysis was collected. The data collected for each event included:

1. plant name;
2. date of occurrence;
3. unit's operating mode at the time of the event, if applicable;
4. how the event was discovered;
5. whether the event was an actual occurrence or a postulated occurrence;
6. a description of the initiating event including the system and component involved;
7. a description of how the initiating event propagated to affect other systems;
8. a description of the unanticipated system dependencies including the systems and components between which dependencies existed;
9. a description of the undesirable result caused by the system interdependencies and identification of the safety-related systems and components affected;
10. the plant building in which the event occurred (for spatial system interaction events only);
11. corrective action; and
12. references to all documents that describe the event.

Appendix B contains an example event and a further description of each data item.

4. REVIEW OF EVENTS

The review and screening of operating experience data identified a total of 235 events as adverse system interaction events. The task of evaluating these events was divided into two parts:

1. study of the events with identification of trends or significant characteristics, and
2. grouping of the events (by similarities) into categories with identification of areas of potential concern (with regard to their impact on safety system operability).

The results of the event evaluations (1) provide insights into system interactions and (2) identify areas of potential concern for future study. Both results will aid efforts to resolve USI A-17.

4.1 Descriptive Statistics of Event Attributes

The first task in evaluating the ASI events was to generate statistics for several of the event attributes (Sect. 3.3 describes the event attributes). These statistics provided insights for identifying possible trends. The event attributes of interest were

1. type of ASI event (i.e., functional, spatial, or induced human);
2. method of discovery;
3. plant operating mode at the time of the event occurrence;
4. result of the event;
5. type of corrective action;
6. number of events by NSSS vendor; and
7. number of events by year of occurrence.

The review and screening of operating experience data identified a total of 235 adverse system interaction events. Of these, 95 were actual events — that is, an initiating event actually occurred and propagated through an unidentified dependency to adversely affect one or more safety-related systems. The remaining 140 events were potential events where a dependency existed but no initiating event occurred. These events are called "potential events" in this report. Although the number of potential events is greater than the number of actual events, this may be inflated because the group of potential events includes generic events. Generic events in this project apply to multiple plants because of a common vendor or design feature. When a generic event was identified as an adverse system interaction event, an event description was included for each plant affected. For example, Westinghouse identified the potential for air binding and damage to the centrifugal charging pumps as a result of level instrument failures for the volume control tank. Five plants reported this potential problem, and an event description for each was included in the list of ASI events.

A system interaction event (as defined in Sect. 1.3) can have one of three types of dependencies: functional, spatial, or induced human.

Each of the 235 ASI events was classified as one of these three types. The number of events of each type are functional dependency — 135, spatial dependency — 96, and induced human dependency — 4. A majority of the events (57%) were the result of functional dependencies. For most of these events, the functional dependency occurred between systems that normally interact with one another (e.g., a process system and a compressed gas system). However, under certain conditions these "interactions" are not desired. For example, nitrogen may be used as a cover gas for the pressurizer when the pressurizer is drained (plant is in cold shutdown or refueling mode). Nitrogen is also used as a cover gas for other equipment such as the safety injection accumulators. During shutdown conditions, the interaction between the pressurizer and the nitrogen system is desired, but during startup or power operations this interaction can lead to several failures: loss of reactor coolant through the nitrogen system or pressurization and possible rupture of equipment served by the nitrogen system. (This example is event 106 in Appendix C.)

About 41% (96) of the ASI events were the result of spatial dependencies. A closer look at the dependencies revealed the following specific types of spatial dependencies:

Water spray or flooding degrading safety-related equipment	27 events
Harsh environmental conditions (high-temperature and humidity) degrading safety-related equipment	33 events
Toppling or falling equipment or structures degrading safety-related equipment (due to seismic or other causes)	15 events
Inadequate cable separation	8 events
Miscellaneous causes (fire, electromagnetic interference, missiles, etc.) degrading safety-related equipment	13 events

Over half the events involving spatial dependencies were caused by harsh environment conditions. Generally, safety-related equipment is qualified for conditions expected during normal operation and design basis accidents.

Only four ASI events were found involving induced human dependencies. As discussed previously, this does not include random human errors. Although infrequently reported, induced human errors can have severe consequences — the Three Mile Island 2 accident involved an induced human dependency. Induced human ASIs involve operator errors or failures that are caused by system dependencies. These type of errors are more probable (because of high stress levels) and may have more serious consequences during severe transients or accidents. Because operating experience data bases contain very little data from such situations, this experience review is not an adequate indication of the potential or lack of potential for such problems.

Another event attribute of interest is how the events were discovered. Table 4.1 lists the method of discovery for both actual and potential ASI events. Almost all of the actual events were discovered through operational abnormalities (i.e., failures occurring during plant power operations). Few were discovered through test or maintenance activities. As expected, the potential events were discovered through either design verification studies or vendor studies (vendor notification).

Another attribute of interest is the plant operating mode. (This attribute was collected only for actual events.) Almost half of the 95 actual events occurred during steady state conditions (when the plant is producing power at any stable, nonzero load). Table 4.2 lists the number of events for each operating mode.

Each event classified as an adverse system interaction event has one or more types of undesirable results (Table 4.3) (the definition of an adverse system interaction event is included in Sect. 1.3). Analysis of the event attribute for type of undesirable results provides some interesting information (Table 4.4). (Note that an event may have more than one type of undesirable result.) Of the 235 ASI events, 77 (or 33%) included a type 2 undesirable result — degradation of safety-related equipment by non-safety-related equipment. When only the actual events are considered, about 50% have a type 2 undesirable result. These facts suggest that further study of the protection of safety-related equipment be considered.

Statistics for undesirable result type 1 — degradation of redundant portions of safety-related systems — show that 149 events (or 63%) of the total number of ASI events have this result type. Considering only the potential events, 79% have this undesirable result type. The majority of these events involved either a common dependency, a single failure, or a shared design problem for redundant equipment.

A review of the corrective actions taken by the plant shows that a design change was the most frequent corrective action (54% or 128 of the 235 ASI events). Administrative/procedure changes were the second most frequent corrective action (20% or 48 events). Table 4.5 lists the corrective actions.

A count of events by NSSS vendor shows that on the average each plant, regardless of NSSS vendor, had about three ASI events (this includes only plants that reported one or more ASI events and excludes General Atomic). Table 4.6 gives the number of plants and events by NSSS vendor. This does not imply that the systems where the ASI occured were necessarily supplied by the NSSS vendor. That information was not collected by the project staff.

The final statistic generated for the event attributes is the number of events grouped by year of occurrence or report date for potential events (Fig. 4.1). This statistic shows that the number of ASI events per year has been increasing steadily but peaked in 1980. The steady increase in actual events (shaded areas) is most likely a result of the increase in the number of plants on-line and reflects changes in the number of reports each year and changes in the reporting requirements. A search of the NSIC file on the RECON data base revealed only 238 abnormal occurrence reports (predecessor to LERs) were reported in 1969. By 1975, the number of reports had increased to 2516, and in 1980, the number rose

Table 4.1. Method of discovery

Method of discovery	Number of ASI events		
	Actual	Potential	Total
Design verification	1	40	41
Installation		2	2
Maintenance	2	3	5
NRC notification		2	2
Operational abnormality	79	2	81
Routine testing	9	6	15
Special testing	1	10	11
Review of test results		4	4
Vendor notification		57	57
Other		2	2
Unknown	2	13	15

Table 4.2. Mode of operation for actual events

Operating mode	Number of ASI events
Cold shutdown	17
Construction	2
Hot shutdown	4
Initial plant startup	4
Load change	1
Refueling	4
Routine shutdown	7
Routine startup	4
Steady state operation	46
Other	1
Unknown	4

Table 4.3. Result types

Type	Description[a]
0	Insignificant degradation of a safety-related system
1	Degradation of redundant portions of a safety-related system, including consideration of all auxiliary support functions. Redundant portions are those considered to be independent in the design and analysis of the plant. This also includes redundant portions of two safety-related systems that can perform the same safety function
2	Degradation of a safety-related system by a non-safety-related system
3	Initiation of an "accident" (e.g., LOCA, MSLB) *and* (a) the degradation of *at least one* redundant portion of any one of the safety-related systems required to mitigate that event; *or* (b) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action
4	Initiation of a "transient" (including reactor trip), *and* (a) the degradation of *at least one* redundant portion of any one of the safety-related systems required to mitigate the event; *or* (b) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action
5	Initiation of an event that (a) requires actions by the plant operators in areas outside the control room area and (b) disrupts the access to these areas

[a]Note that in some cases, combinations of undesirable results occurred. For example, failure of a non-safety-related system that caused a transient and degraded a safety system would be a type 2 result and type 4 event (recorded as 2,4).

to 3837. The number of LERs written by plants has risen steadily over the years — in 1983 the total was 5657 reports.

The sharp increase and peaking of potential events up to 1980 is most likely a result of increased design reviews and regulatory requirement changes in the years immediately following the Three Mile Island 2 accident. This observation is supported by the fact that most potential events were discovered by design verifications or vendor notification (see Table 4.1).

Table 4.4. Undesirable result for ASI events

Undesirable result type[a]	Number of ASI events		
	Actual	Potential	Total
0	4	3	7
1	14	38	52
2	24	15	39
3	5	3	8
4	18	4	22
5		1	1
1, 2	19	8	27
1, 3	1	56	57
1, 4	4	7	11
2, 3	1		1
2, 4	3	4	7
1, 2, 3		2	2
2, 4, 5	1		1

[a]Defined in Table 4.3.

Table 4.5. Corrective action taken

Corrective action	Number of ASI events		
	Total	Actual	Potential
Design change	128	45	83
Administrative change	48	14	34
Repair	18	13	5
Other	41	22	19

Table 4.6. Plants by NSSS vendor

NSSS vendor	Number of plants	Total number of ASI events
Babcock and Wilcox	13	30
Combustion Engineering	11	35
General Atomic	1	1
General Electric	23	64
Westinghouse	31	105

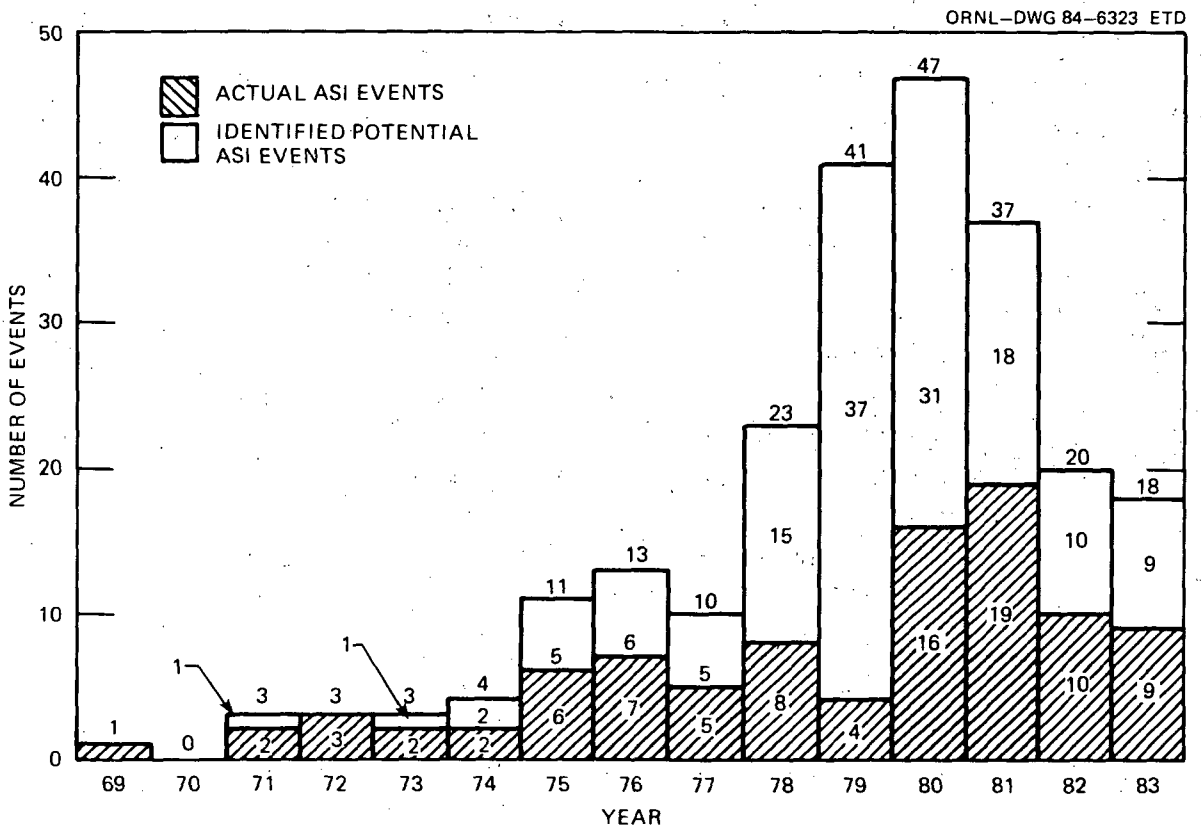

Fig. 4.1. Number of system interaction events by year.

4.2 Categories of System Interaction Events

As part of the data evaluation effort, the 235 ASI events were compared for commonalities. This allowed grouping of the events into 23 categories. (The number of events in each category varied. No event was placed in more than one category.) Each category contains events that share a predominant trait; for example, category 2 contains events where safety-related equipment was degraded by vapor or gas intrusion. Some of the categories parallel areas of concern identified in certain unresolved safety issues and CIs.[1,2] Others have been previously identified in AEOD reports and IE bulletins and notices; still others have received relatively little attention. Table 4.7 lists the categories and gives the subject and number of events for each.

Sections 4.2.1 through 4.2.23 discuss each category in greater detail. Specific details for each category include a general scenario of the events, the systems involved, and industry and regulatory responses. In this report, each of the 235 ASI events has been given a unique "event number." This event number refers to a more detailed description of the event, given in Appendix C.

Table 4.7. Event categories

Category No.	Title	Number of events
1	Adverse interactions between normal or offsite power systems and emergency power systems	34
2	Degradation of safety-related systems by vapor or gas intrusion	15
3	Degradation of safety-related components by fire protection systems	10
4	Plant drain systems that allow flooding of safety-related equipment	8
5	Loss of charging pumps due to volume control tank level instrumentation failures	6
6	Inadvertent ECCS/RHR pump suction transfer	4
7	HPSI/charging pumps that overheat on low flow during safety injection	6
8	Level instrumentation degraded by high energy line break (HELB) conditions	21
9	Loss of containment integrity due to LOCA conditions during purge operations	10
10	HELB conditions degrading control systems	3
11	Auxiliary feedwater pump runout under steam line break conditions	2
12	Water hammer events	4
13	Common support systems or cross-connects	18
14	Instrument power failures affecting safety systems	5
15	Inadequate cable separation	8
16	Safety-related cables unprotected from missiles generated from HVAC fans	3
17	Suppression pool swell	3
18	Scram discharge volume degradation	2
19	Induced human interactions	4
20	Functional dependencies due to failures during seismic events	5
21	Spatial dependencies due to failures during seismic events	13
22	Other functional dependencies	21
23	Other spatial dependencies	30

Evaluating the safety significance of each category of events, or of individual events, was not included in the scope of this project. However, future work will address quantitative and qualitative assessments of the safety significance of each of the categories.

Each category represents sources of intersystem dependencies that have degraded the level of redundancy required for safety systems. However, some of the categories represent problems that have already been resolved. Also, from a risk viewpoint, the problems may not be significant. Subsequent phases of this work will rank each category in terms of safety significance. The numbering of the categories in this report is simply for convenience and does not imply any prioritization.

4.2.1 Category 1 — Adverse interactions between normal or offsite power systems and emergency power systems

Thirty-four events were included in category 1 (Table 4.8). This category contains events that involved interactions between the normal power distribution systems (including offsite power) and emergency power systems. An evaluation of events within this category identified four specific problem areas that have been reported on several occasions. These areas include:

1. load sequencing/load shedding problems (11 events),
2. diesel generator (DG) failures caused by specific DG operating modes (6 events),
3. dc breaker failures due to loss of dc power (7 events), and
4. other failures that propagate from non-safety-related power systems (10 events).

Load sequencing/load shedding. Current nuclear power plants use electrical load sequencers to control the order and timing of startup of the large electrical loads required during accident conditions. The sequencers are designed to control these loads to ensure stable electrical distribution, with or without availability of offsite power. The signals pertinent to load sequencers occur under LOCA, ESFAS, LOSP or safety injection conditions.

Eight potential events were identified where electrical load sequencers could fail to operate properly. Six of the events describe instances where a sequencer could fail to start up loads. This occurs when: (1) an ESFAS is simultaneous with, or is followed by, a loss of power (events 50, 53, 171, and 172); (2) a LOCA and loss of power occurs after a DG has been manually stopped (event 49); or (3) a DG is supplying an essential bus and backfeeding the feeder bus, and subsequently a loss of power occurs (event 51). The remaining two potential events describe possible overloading of the DGs caused by: (1) the simultaneous sequencing of the loads on the buses (event 30) or (2) a loss of power followed by manual loading of the DG and then a LOCA (event 44).

Closely related to load sequencing is load shedding. During a loss of power, non-safety-related equipment loads are removed from the buses so that the buses and the diesel generators are not overloaded. Potential load shedding problems can occur when all nonessential loads are

Table 4.8. Category 1 — Adverse interactions between normal or offsite power systems and emergency power systems

Plant	Date	Event No.	Description
Arkansas Nuclear 2	09/16/78	4	Loss of electric power to both units due to overload of common transformer
Arkansas Nuclear 2	09/16/78	6	Potential loss of ESF equipment following loss of normal electric power and failure to transfer loads to diesel generators due to low voltage conditions
Brunswick 1	11/08/79	28	Loss of emergency bus due to lockout of DG output breaker (lockout caused by simultaneous open and close signals)
Brunswick 2	09/17/75	30	Potential overload of DGs due to simultaneous sequencing of loads on buses (occurs with loss of power and slow reduction of reactor pressure)
Connecticut Yankee	05/08/78	42	Potential overload of DGs due to presence of non-safety loads (occurs when LOCA with safety injection is coincident with loss of power)
Connecticut Yankee	01/29/80	44	Potential overload of DGs due to presence of non-safety loads (occurs when loss of power occurs followed by manual DG loading and then a LOCA)
Davis-Besse 1	12/23/76	49	Potential failure of SFAS sequence to restart safety system pump (occurs when LOCA and loss of power occur after DG has been manually stopped)
Davis-Besse 1	2/24/77	50	Potential overload of DGs due to SFAS sequencer failures (occurs when SFAS is manually initiated and then a loss of power occurs)
Davis-Besse 1	06/07/78	51	Potential failure of SFAS sequencer (occurs when DG is supplying essential bus and backfeeding a feeder bus prior to loss of power)
Davis-Besse 1	07/09/80	53	Potential overload of DGs due to ESFAS sequencer failures (occurs when ESFAS actuation is coincident with loss of power)
Fort St. Vrain	05/17/83	71	Potential overload of one DG (occurs when one DG is inoperable and other DG is paralleled to offsite power and loss of power occurs)
Hatch 1	03/30/78	77	Potential loss of two DGs (occurs during LOCA when DG battery fails followed by loss of power)
Hatch 1	01/29/80	79	Potential loss of DGs due to loss of SW (occurs following loss of ac power required to transfer power supply breakers)
Hatch 2	03/30/78	83	Same as event 77, but reported for Unit 2
Millstone 1	09/14/79	101	Potential loss of power to ECC due to loss-of-normal-power logic errors
Millstone 1	04/03/81	102	Potential loss of normal and emergency power to safety systems due to time delay relay failure (breakers do not get reclose signal)
Millstone 2	07/05/76	103	ESFAS equipment operability not assured under degraded grid voltage
Millstone 2	07/21/76	104	ESFAS loads shed from emergency buses due to improper undervoltage relay set points (changed due to a previous event)
Millstone 2	01/02/81	105	Failure of turbine trip and loss of power to auxiliary loads due to loss of dc power

Table 4.8 (continued)

Plant	Date	Event No.	Description
Millstone 2	01/02/81	105	Failure of turbine trip and loss of power to auxiliary loads due to loss of dc power
North Anna 1	11/14/80	117	Potential damage to safety equipment due to out-of-phase transfer of DGs to buses (occurs when ESFAS actuation is followed by loss of power)
Oyster Creek	09/18/73	125	Failure of two DGs to restart after lockout (occurs when DG is secured after a fast start)
Palisades	03/12/72	135	Loss of one-half of ESF systems due to failure of unit protection logic to transfer loads to alternate power source when reactor is manually scrammed
Quad Cities 1	06/22/82	152	Loss of required redundancy in electric power sources
Rancho Seco	11/01/79	156	Potential inadequacy of onsite power due to degraded grid voltage
San Onofre 1	09/02/80	171	Potential failure of SFAS sequencer to reload safety equipment to buses (occurs when SFAS actuation is followed by block and loss of power)
San Onofre 1	01/16/81	172	Potential loss of power to safety equipment due to sequencer failure (occurs when an SFAS actuation is followed by loss of power and oscillation of SFAS parameters)
Sequoyah 1	12/12/80	182	Potential overload of DG due to sequencer failure (loads not stripped from bus if DG is running before loss of power occurs)
St. Lucie 1	11/25/75	186	Unable to repower bus (following loss of power) because load shed relays required power to actuate
Surry 1	03/23/79	193	Potential overload of emergency buses following a LOCA unless loads are manually shed
Surry 1	11/14/80	199	Potential damage to DGs due to out-of-phase transfer of DGs to buses (occurs when ESFAS actuation starts DG prior to loss of power)
TMI-1	01/23/80	209	Potential loss of CCW to safety systems due to overload of an electric power bus (occurs when opposite train bus fails, SFAS actuates, and offsite power is available)
TMI-1	01/17/81	210	Potential damage to DGs due to out-of-phase transfer of DGs to buses (occurs when ESFAS actuation starts DG prior to loss of power)
Zion 2	09/19/76	232	Overload caused damage of DG while paralleled to grid (occurred when loss of dc power caused some trips but prevented other breaker transfers)
Zion 2	09/19/76	233	Severe MFW overfeeding caused a safety injection and relief to containment (occurred when loss of dc power prevented certain breaker transfers)

not stripped from the buses. The potential for failure to strip loads that could result in overloading and damaging the diesel generators was reported by three plants (events 42, 182, and 193).

Diesel generator failures due to specific DG operating modes. Six events in category 1 involved at least a partial failure of the emergency power system due to adverse system interactions involving the diesel generators. The potential for damage to the diesel generators when offsite power was lost was identified by three plants. In each instance, the diesel generator could be damaged if it was started prior to offsite power being lost. The damage could result from (1) overloading the diesel generator (event 71), (2) performing an out-of-phase transfer (event 199), or (3) transferring the diesel generator to a bus that contains a residual voltage (event 210). Event 71 was the subject of IE Notice 84-69, "Operation of Emergency Diesel Generators." The event occurred while the reactor was shut down and one of the two diesel generators was out of service for maintenance. As a result of high winds and snow, the offsite power system experienced problems. As a precautionary measure, the available diesel generator was started and tied to the associated safety bus that was in parallel with the offsite power source. Because of an overload, all offsite power to the plant was lost, and the output breaker of the operating diesel generator was tripped. As a result, the plant was without all ac power, except for the inverter ac power fed from the dc power system.

Three events (events 28, 117, and 125) involved design or operation errors (actual and potential) in which failure to consider certain diesel generator operating modes resulted in adverse interactions. In event 28, a degraded voltage condition caused the diesel generator output breaker to trip. By design, the trip (open) signal was applied to the breaker for 2 s by a time delay relay. However, once the breaker opened, the nonessential loads were stripped and the breaker immediately received a close signal. The resulting simultaneous "open" and "close" signals to the DG output breaker caused a lockout.

In event 125, power was lost to the station loads during an attempt to transfer power from an auxiliary transformer to a startup transformer. The diesel generator started when station power was lost. Station power was subsequently restored and the diesel generator was manually shutdown. Power was lost a second time; however, the diesel generator was in lockout. Because of a design deficiency, the diesel generator locks out when shut down after a fast start.

At North Anna 1 (event 117), personnel discovered that an out-of-phase transfer of the diesel generators to the buses could occur when a safety injection actuation was followed by a loss of offsite power. The out-of-phase transfer could damage numerous safety-related components. As a result of design errors, no logic existed to prevent the diesel generators from transferring to the buses before the residual voltage on the buses collapsed.

Direct current breaker failures caused by loss of dc power. Seven reports (events 77, 79, 83, 105, 186, 232, and 233) describe the inability to use dc power to trip breakers (open) following the loss of a dc power source. In event 77 and 83, the potential existed for the failure of a single battery system to cause redundant diesel generators to fail. This

could occur because the battery is required to operate one diesel generator, and loss of dc power causes the other DG to fail because of overloads from equipment that cannot be tripped off without dc power. In event 105, loss of a dc power bus produced a reactor scram. However, without the dc bus, the turbine could not be tripped and plant loads could not be transferred to an auxiliary power source.

In response to IE Bulletin 79-27, "Loss of Non-Class 1-E Instrumentation and Control Power Systems During Operation," personnel discovered a design error at Hatch 1. Certain supply breakers required ac power for motive force (event 79). However, ac power is not available to the breakers following a loss of station power. The breakers were changed to operate on dc power. A similar event occurred at St. Lucie 1 (event 186). A bus failed to load following a loss of offsite power. A review showed that the load shed relay received only nonessential (offsite) power. Thus, the loss of station power deenergized the load shed relay and prevented load shedding. (Proper load shedding is required before the bus can be reenergized.)

Two different system dependencies occurred in a single event at Zion 2 (events 232 and 233). A loss of dc power resulted in a turbine trip, but the main generator output breaker could not open because of the loss of dc power. Because the main feedwater pumps tripped when the generator output breaker opened, the pumps continued to run, resulting in an overfeeding transient. Concurrently, a diesel generator caught fire as a result of an overload caused by the failure of certain circuit breakers to trip without dc power.

All of the dc breaker events were reported as LERs, but no IE notices or bulletins were found that specifically addressed this area.

Other failures that propagate from non-safety-related power systems. The final concern for this category is events involving the propagation of failures from non-safety-related power systems. The variety of different ways that failures can propagate in electrical power systems is reflected in this group of diverse failures. They reflect the complexity of the electric power supply and distribution systems at nuclear power plants.

At Millstone 2 (event 103), a low grid voltage degraded several safety systems, while at the same time prevented a transfer to emergency power because power was not totally lost. Two months later at the same plant (event 104), the undervoltage protection modification made in response to the first event caused an inability to energize the ESF buses. Based on a review of these events, the NRC required changes in undervoltage protection.

A review of undervoltage protection design at Rancho Seco revealed that the plant undervoltage trip setpoints could cause a loss of power to onsite buses (event 156). Low voltage on the grid would cause the undervoltage protection relays to operate; however, this would not actuate the loss-of-normal-power logic.

Other failures reported involving normal station power include: potential failure to transfer loads to the diesel generator due to low voltage conditions (event 6); an error in the loss-of-normal-power logic (event 101); a relay error preventing energizing of the emergency buses (event 102); failure to change transformer set points to support two units at one location (event 4); failure to have a manually initiated scram

transfer power from the generator to the startup transformer (event 135); potential to overload an electric power bus when the opposite train bus failed (event 209); and loss of redundancy in electric power sources as a result of LOSP at unit 2 with no diesel generators available at unit 1 (event 152).

The large number of system interactions involving electric power systems is cause for concern. Because of the diversity of the events, further effort will be required to determine the extent to which industry and NRC actions have adequately addressed system interactions — actual or potential — in this key area. The NRC has recognized the safety significance associated with electric power systems and currently has identified seven unresolved safety issues and five generic issues concerning electric power. The unresolved safety issues are listed below:[1,2]

1. A-24, "Qualification of Class IE Safety-Related Equipment";
2. A-25, "Nonsafety Loads on Class IE Power Sources";
3. A-30, "Adequacy of Safety-Related DC Power Supplies";
4. A-35, "Adequacy of Offsite Power Systems";
5. A-44, "Station Blackout";
6. B-57, "Station Blackout"; and
7. B-70, "Power Grid Frequency Degradation and Effect on Primary Coolant Pumps."

The generic issues are listed below:

1. GI-17, "Loss of Offsite Power Subsequent to LOCA";
2. GI-26, "Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power";
3. GI-46, "Loss of 125 Volt DC Bus";
4. GI-47, "Loss of Off-Site Power"; and
5. GI-55, "Failure of Class IE Safety-Related Switchgear Circuit Breakers to Close on Demand."

4.2.2 Category 2 — Degradation of safety-related systems by vapor or gas intrusion

Category 2 contains 15 events that involved the unanticipated failure of safety-related equipment due to vapor or gas intrusion (Table 4.9). The safety-related equipment was affected in several ways:

1. steam binding of auxiliary feedwater pumps,
2. loss of pump suction head, and
3. interactions with compressed gas systems.

Steam binding of auxiliary feedwater pumps. Three events (events 7, 163, and 206) involved vapor binding of the auxiliary feedwater (AFW) pumps as a result of hot water or steam from the main feedwater system being introduced into the AFW system. At Arkansas Nuclear 2, operators left an isolation valve open in the steam generator blowdown system. This allowed hot water from the main feedwater system to enter the startup and blowdown tanks and demineralizers. The AFW pumps, which were taking

Table 4.9. Category 2 — Degradation of safety-related systems by vapor or gas intrusion

Plant	Date	Event No.	Description
Arkansas Nuclear 2	04/07/80	7	Loss of AFW due to steam binding of AFW pumps (hot water from SG blowdown system flashed)
Beaver Valley 1	01/17/80	11	Loss of RHR due to air binding of RHR pumps
Beaver Valley 1	04/11/80	12	Same as event 11
Bellefonte 1	11/12/82	18	Potential loss of one makeup train due to gas binding of makeup pumps
Calvert Cliffs 1	05/20/80	34	Loss of plant SW due to air binding of pumps (air entered system via air compressor aftercooler leak)
Calvert Cliffs 1	08/12/80	35	Same as event 34
Calvert Cliffs 2	10/17/78	37	Loss of RHR due to air binding of RHR pumps (air leaked into RHR from purification system via cross-connect)
McGuire 1	02/12/82	96	Loss of HPSI/charging due to gas binding of pumps (hydrogen from leaking pulsation dampener entered common suction line)
Millstone 2	01/08/81	106	Over-pressurization of ECCS accumulators by steam intrusion from pressurizer (via nitrogen system)
Quad Cities 1	12/30/76	151	Potential loss of all SW pumps due to air intrusion (air leaked into common SW header via air system connection to valve)
Robinson 2	06/13/77	159	Loss of two charging pumps due to air binding (air entered system via ruptured valve diaphragm)

Table 4.9 (continued)

Plant	Date	Event No.	Description
Robinson 2	04/19/83	163	Loss of AFW due to steam binding of AFW pumps (hot feedwater flashed to steam in AFW pumps)
San Onofre 1	07/17/81	173	Waste gas recombiner exploded when instrument air entered a recombiner via the nitrogen system
San Onofre 2	03/14/82	175	Loss of RHR due to gas binding of RHR pumps (nitrogen leaked into RHR suction line from purification system via cross-connect)
Surry 2	11/18/83	206	Loss of two AFW pumps due to steam binding (hot feedwater backflowed through leaky header check valves)

suction from both the condensate tanks and the startup and blow-down demineralizers, became vapor bound as the hot water flashed to steam. This event prompted the NRC to issue IE Notice 80-23, "Loss of Suction to Emergency Feedwater Pumps," alerting licensees of the potential loss of suction to AFW pumps.

At Robinson 2 and Surry 2 (events 163 and 206, respectively), the AFW system discharges to the steam generators via the main feedwater header. Leaking check valves and isolation valves, which separate the two systems, allowed hot water from the feedwater lines to seep into the AFW pumps. The hot water flashed to vapor, binding the pumps. The event at Robinson 2 prompted the NRC to issue IE Notice 84-06, "Steam Binding of Auxiliary Feedwater Pumps," to alert licensees of this problem. A recent AEOD case study (AEOD/C404, "Steam Binding of Auxiliary Feedwater Pumps") reviewed the problems of backleakage from the main feedwater system to the auxiliary feedwater system. AEOD recommended that licensees monitor the AFW system for backleakage and maintain fluid conditions in the system below saturation conditions.

Loss of pump suction head. Two reports from one plant (events 11 and 12) described the loss of residual heat removal (RHR) flow caused by air binding of the RHR pumps. The events occurred during refueling when the reactor coolant system (RCS) was at a low water level (only a small heat load was present). At low RCS water levels (1) sufficient head may not be provided or (2) air can be drawn from the steam generator allowing

the pumps to become air bound. These events can only occur during refueling (when the RCS water level is low).

Interactions with compressed gas systems. Ten events (events 18, 34, 35, 37, 96, 106, 151, 159, 173, and 175) involved the loss of safety-related equipment because of the intrusion of compressed gases. For example, at Calvert Cliffs 1 (event 34 and 35) a tube failure in an (instrument) air compressor aftercooler allowed compressed air to enter one train of the service water system. Because of a common header, air entered the redundant train and all service water flow was lost.

At McGuire 1 (event 96), the reciprocating pump in the chemical and volume control system (CVCS) had a dampener in its suction line. As a result of instrument failures, the water level in the dampener became low and the hydrogen cover gas entered the pump's suction line. Because of the common suction header, the potential existed for all of the charging pumps to become gas bound.

At Millstone 2 (event 106), personnel left two nitrogen isolation valves to the pressurizer open (nitrogen was used as a cover gas while the pressurizer was drained). During plant heatup, steam from the pressurizer entered the nitrogen system and subsequently overpressurized a core flooding accumulator (which uses nitrogen as a cover gas).

These three events are typical examples of adverse interactions between safety-related systems and compressed gas systems. IE Notices 81-27, "Flamable Gas Mixtures in Waste Gas Decay Tanks in PWR Plants"; 82-19, "Loss of High Head Safety Injection, Emergency Boration and Reactor Coolant Makeup Capability"; and 83-77, "Air/Gas Entrainment Events Resulting in System Failures," alerted licensees of possible failures of safety-related equipment caused by gas entrainment. AEOD has performed an engineering evaluation (AEOD/E317, "Loss of High Pressure Injection") on loss of the high pressure injection system (HPSI) due to hydrogen entrainment. [For some plant designs, the charging pumps in the CVCS double as HPSI pumps. Also, hydrogen is used in the CVCS as a cover gas for tanks and dampeners. Because of the shared suction header of the charging/HPSI pumps, hydrogen entrainment in one train can affect redundant trains (e.g., event 96).] AEOD recommended that at future plants all charging/HPSI pumps have separate suction lines. Category 5 contains similar events involving air binding and damage of the charging/HPSI pumps.

4.2.3 Category 3 — Degradation of safety-related components by fire protection systems

Category 3 contains ten events in which automatic actuation of the fire protection systems degraded or could potentially degrade safety-related equipment (Table 4.10). The safety-related equipment was affected in three ways: (1) water intrusion in electrical components, (2) water contamination of lube oil or fuel oil systems, and (3) overpressurization of safety-related structures.

Water intrusion in electrical components. Six events (events 8, 74, 128, 129, 184, and 218) in this category involved damage (both actual and potential) to electrical equipment caused by the actuation of sprinklers.

Table 4.10. Category 3 — Degradation of safety-related components by fire protection systems

Plant	Date	Event No.	Description
Arkansas Nuclear 2	08/03/83	8	Potential flooding of cable spreading room by fire suppression system (in auxiliary building)
Dresden 2	12/23/81	58	Loss of HPCI due to actuation of fire suppression system (actuation caused by high room temperature)
Dresden 3	11/30/81	60	Same as event 58
Ginna	11/14/81	74	Wetting of RPS motor generator switchgear and CRD power supply by fire suppression system (inadvertently actuated)
Grand Gulf 1	07/14/82	75	Repeated inadvertent actuation of ECCS penetration room CO_2 fire suppression system blew off locked door to auxiliary building
Oyster Creek	09/30/80	128	Loss of core spray system due to actuation of fire suppression system causing water damage to pumps
Oyster Creek	02/18/82	129	Water damage to RPS and core spray instruments due to actuation of fire suppression system
Sequoyah 1	12/01/83	184	Potential loss of control room HVAC chillers due to electrical equipment damaged by water spray from fire suppression system
Surry 2	05/28/81	204	Loss of diesel generator when water from foam distributor (fire protection) entered fuel tank
Trojan	07/28/81	218	Loss of hydrogen recombiner when fire suppression system wet control power transformer

This included both (1) water impingement directly on the electrical equipment and (2) flooding of areas containing electrical equipment due to prolonged operation of the sprinklers. For example, at Oyster Creek (event 128), a maintenance error inadvertently actuated sprinklers in plant areas that house the core spray system. The water spray from the sprinklers disabled the core spray pump motors. The core spray system was thought to be protected from water intrusion.

Water contamination of lube oil or fuel oil systems. Three events in this category (events 58, 60, and 204) involved degradation of mechanical equipment as a result of water contaminating its lube oil or fuel oil system. At Dresden 2 and 3 (events 58 and 60, respectively), water from sprinklers entered the lube oil system for the high pressure coolant injection system. At Surry 2 (event 204), water entered several fuel oil storage tanks (for the diesel generators) through a foam distributor. The foam distributor was connected to the fire suppression system water main.

Overpressurization of safety-related structures. One event in this category involved the overpressurization of a safety-related structure by a fire protection system. At Grand Gulf 1 (event 75), a ground in the initiation circuit caused repeated actuation of the carbon dioxide fire suppression system, pressurizing a penetration room (for ECCS penetrations). The design of the room did not allow adequate venting of the excess carbon dioxide, and the pressure buildup blew off the penetration room door.

In 1982, AEOD issued a report (AEOD/E204, "Effects of Fire Protection System Actuation on Safety-Related Equipment") documenting several instances where actuation of fire suppression systems adversely affected safety-related equipment. The NRC has also (1) issued IE Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," to alert licensees of the potential degradation of safety-related equipment by fire suppression systems and (2) created Generic Issue 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment,"[1] for further study of this problem.

Most of the events in this category were initiated by inadvertent actuation of the fire protection system. Several were caused by the use of high temperature or high-humidity detectors in fire detection roles. Because rooms containing safety-related mechanical equipment would most likely be the hottest during an accident, this application of sensors should be examined.

4.2.4 Category 4 — Plant drain systems allow flooding of safety-related equipment

Category 4 contains eight events in which safety-related equipment was degraded or could be degraded because of inadequate plant drains (Table 4.11). For these events, the plant drains were inadequate in one of two ways: (1) drains were not isolatable, or (2) drains were inadequately sized.

Drains were not isolatable. Six events (events 36, 38, 85, 148, 164, and 219) in this category involved degradation of safety-related equipment caused by water or steam backflowing through plant drains. For

Table 4.11. Category 4 — Plant drain systems allow flooding of safety-related equipment

Plant	Date	Event No.	Description
Calvert Cliffs 1	11/05/81	36	Potential loss of SW due to flooding of pumps (from main condenser leak) via unisolatable drains
Calvert Cliffs 2	11/05/81	38	Same as event 36
Calvert Cliffs 2	10/19/83	40	Control rod dropped when water (from toilet) dripped on CR cabinet shorting equipment
Hatch 1	08/25/82	85	Loss of RCIC and electrical equipment due to high ambient temperature when steam (from SDV leak) traveled through drains
North Anna 2	07/03/81	120	Spread of oil from transformer fire when deluge water overflowed pit (drains too small)
Prairie Island 2	08/30/75	148	Potential loss of both RHR trains due to water inleakage from redundant RHR pit or containment spray pit
Salem 1	02/06/75	164	Damage to vital bus and two 4-kV breakers when water flowed (via construction blockout) into auxiliary building
Turkey Point 3	11/17/72	219	Flooding of 4160-V switchgear rooms when water from yard catch basin backflowed through floor drains

example, at Hatch 1 (event 85), a valve on the scram discharge volume leaked. The floor drains collected the hot water and steam flow from the leak. Because of the lack of check valves in the floor drain system, hot water and steam backflowed into other areas of the reactor building including the room housing the reactor core isolation cooling (RCIC) system. This resulted in high ambient temperatures and actuation of the fire protection system. The ambient temperatures were also above the limits for electrical equipment located in the area.

Drains were inadequately sized. Two events (events 40 and 120) in this category involved degradation of safety-related equipment as a result of flooding (the drain could not adequately handle flows). For example, at North Anna 2, the B-phase main transformer caught fire and later ruptured, spilling oil into the surrounding pit. The sprinklers actuated automatically to control the fire. Because the floor drain in the pit was too small to accommodate water flow from the sprinklers, the pit overflowed. The flaming oil spilled into the surrounding areas and hampered fire fighting efforts.

The Calvert Cliffs 1 and 2 events (events 36 and 38) prompted AEOD to evaluate the generic implications of backflow flooding of safety-related equipment through drain lines. AEOD concluded (in report AEOD/E304, "Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Components") that backflow flooding protection had not been adequately addressed. The NRC issued IE Circular 78-06, "Potential Common Mode Flooding of ECCS Equipment Rooms in BWRs," and IE Notice 83-44, "Potential Damage to Redundant Safety Equipment from Backflow Through Equipment and Floor Drain System," to alert licensees of this problem. The NRC also created Generic Issue 77, "Flooding of Safety-Related Equipment Compartments by Back-flow Through Floor Drains," for further study of this problem.[1] This issue has a high-priority rank.

4.2.5 Category 5 — Loss of charging pumps due to volume control tank level instrumentation failures

Six events were classified as category 5 (Table 4.12). Five of the events (events 16, 66, 118, 168, and 200) involved the loss of charging/makeup pumps (while in the makeup mode) because of interactions with the volume control tank (VCT) level instrumentation. For some PWR designs, the pumps that provide charging or makeup to the RCS also serve as high pressure safety injection (HPSI) pumps. This problem is generic to certain plants of Westinghouse design. The postulated event sequence is as follows:

1. A single level transmitter on the VCT fails, giving a false "high" level reading.
2. This false signal causes the control system to stop letdown flow to the tank.
3. The operating charging pumps eventually drain the VCT and fail due to a loss of suction.
4. When an operating charging pump fails or trips off, a standby pump starts.
5. The standby pump will also fail on loss of suction because the level control circuitry will prevent switchover to the alternate water source — the refueling water storage tank.

The vendor identified this potential interaction in 1981. The five utilities that had a potential for this problem made procedural changes to outline corrective actions should it occur.

Table 4.12. Category 5 — Loss of charging pumps due to volume control tank level instrumentation failures

Plant	Date	Event No.	Description
Beaver Valley 1	05/21/81	16	Potential loss of all HPSI/ charging pumps due to loss of suction (VCT is pumped dry)
Farley 1	05/22/81	66	Same as event 16
North Anna 1	05/22/81	118	Same as event 16
Salem 1	05/21/81	168	Same as event 16
St. Lucie 1	10/23/82	189	Loss of all charging pumps (became vapor bound) due to loss of suction (VCT was pumped dry)
Surry 1	05/22/81	200	Same as event 16

In 1982, St. Lucie (event 189) lost all charging pumps because of gas binding when an empty reference leg caused a VCT level instrument to fail. The NRC performed two engineering evaluations addressing the problem (AEOD reports E314, "Loss of All 3 Charging Pumps Due to Empty Common Reference Leg in the Liquid Level Tranducers for the Volume Control Tank," and E317, "Loss of High Pressure Injection") and issued IE Notice 83-77, "Air/Gas Entrainment Events Resulting in System Failures." Category 2 contains some similar events involving gas binding of the charging pumps (at plants not of Westinghouse design).

4.2.6 Category 6 — Inadvertent ECCS/RHR pump suction transfer

The emergency core cooling systems for PWRs are designed to operate in two phases:

1. injection phase — the ECCS pumps take suction from the borated/ refueling water storage tank (RWST) and inject it into the RCS for initial cooling, and
2. recirculation phase — the ECCS pumps take suction (via the RHR) from the containment sump and inject it into the RCS for long-term cooling.

For these systems to change from injection phase to recirculation phase, certain valves must switch positions. Several of the NSSS vendors provide automatic switching logic in the engineered safety features actuation system (ESFAS). Category 6 involves the inadvertent (automatic) transfer of ECCS pump suction to the containment sump.

Four events were identified (Table 4.13) in which various failures initiated an inappropriate (and undesired) recirculation actuation signal (RAS) by the ESFAS; two of the four events were caused by loss of power (LOP). At Arkansas Nuclear 2 (event 5), a transformer failure caused an LOP. Because of incorrect set points and associated problems, the inverters failed, and all vital ac instrument power was lost. This caused a full safety injection actuation signal (SIAS) and an undesired RAS. While valves in ECCS were changing positions, borated water from the RWST was gravity fed to the containment sump. The ECCS pumps could have been damaged because their suction was transferred to an empty sump.

Table 4.13. Category 6 — Inadvertent ECCS/RHR pump suction transfer

Plant	Date	Event No.	Description
Arkansas Nuclear 2	09/16/78	5	Premature transfer of ECCS pump suction to containment sump (following a safety injection) due to loss of power
Davis-Besse 1	04/19/80	52	Premature transfer of RHR pump suction to containment sump (following a safety injection signal) due to loss of power to two ESF buses
San Onofre 3	12/17/82	177	Potential premature transfer of ECCS pump suction to containment sump (following a safety injection signal) due to single RPS cable failure
Sequoyah 2	08/06/81	185	RCS pressure boundary breached after an RHR sump isolation valve opened creating a leak path from the RCS to the containment sump

At Davis-Besse 1 (event 52), power was lost to two essential buses that were sharing a power supply (for maintenance work). The loss of these buses caused a full SIAS with RAS. During automatic valve realignment, borated water was gravity fed to the containment sump. Davis-Besse 1 experienced several other inadvertent RASs that have caused transfer of ECCS suction to an empty containment sump.[1]

Sequoyah 2 (event 185) experienced a similar event when a testing error initiated an RAS. This opened the isolation valves between the

containment sump and one train of RHR. The unit was in cold shutdown with both trains of RHR in operation. As certain valves were changing position (in response to the RAS), ~7800 gal of primary coolant was lost to the sump via the open RHR recirculation line.

San Onofre 3 (event 177) experienced an inadvertent RAS following the loss of two independent power supplies to the plant protection system. Investigation of this event identified a single 40-pin amphenol connector in the plant protection system that, if disconnected, would deenergize the bistable relay matrix and initiate an SIAS and RAS.

The Davis-Besse 1 event prompted the NRC to write IE Bulletin 80-12, "Decay Heat Removal System Operability," and IE Notice 80-20, "Loss of Decay Heat Removal Capability at Davis-Besse Unit 1 While in a Refueling Mode." Neither of these addressed the problem of an inadvertent RAS. The NRC is aware of this problem (inadvertent RAS) and has created Generic Issue 24, "Automatic Emergency Core Cooling System Switch to Recirculation."[1]

4.2.7 Category 7 — HPSI/charging pumps overheat on low flow during safety injection

The events grouped in category 7 involve the potential failure of the charging/HPSI pumps caused by low flow conditions through the pumps (Table 4.14). The postulated event sequence is as follows:

1. A feedwater or main steam line break inside containment produces high ambient temperatures and humidity.
2. Engineered safety features instrumentation senses this accident and initiates appropriate safety system response.

Table 4.14. Category 7 — HPSI/charging pumps overheat on low flow during safety injection

Plant	Date	Event No.	Description
Beaver Valley 1	08/27/80	13	Potential loss of all HPSI/charging pumps due to low pump flow (high RCS pressure at discharge and closure of minimum flow line)
Farley 1	06/13/80	63	Same as event 13
North Anna 1	05/09/80	116	Same as event 13
Sequoyah 1	06/13/80	180	Same as event 13
Surry 1	06/11/80	197	Same as event 13
Zion 1	05/23/80	230	Same as event 13

3. As part of this response, the charging/HPSI pumps switch from charging mode to high pressure safety injection mode and the recirculation flow paths (miniflow lines) for the pumps isolate.
4. Because of high ambient temperatures, the control circuitry for the pressurizer power-operated relief valve (PORV) fails.
5. The PORV cannot be opened by the plant's operators, and the reactor coolant system (RCS) pressure increases.
6. With their recirculation paths isolated, the charging/HPSI pumps overheat and fail when the RCS pressure becomes greater than the pumps' maximum discharge pressure. (The pressurizer safety valve will prevent overpressurization of the RCS.)

This accident sequence is of interest because the failed charging/HPSI pumps might require repair to restore them to operation and could be unavailable for the duration of the accident. The NRC addressed this problem in IE Bulletin 80-18, "Maintenance of Adequate Minimum Flow Through Centrifugal Charging Pumps Following Secondary Side HELB." A total of six plants have notified the NRC via LERs of the potential for this problem at their facility. Two corrective actions were implemented: (1) the isolation valves on the recirculation lines no longer close on a safety injection signal and (2) manipulation of these valves is controlled by procedures.

4.2.8 Category 8 — Level instrumentation degraded by high energy line break conditions

This category includes potential failures of engineered safety features (ESF) level instrumentation. The postulated event sequence is as follows:

1. A high energy line break occurs inside the primary containment, resulting in high ambient temperatures inside containment.
2. The level sensors (steam generator or containment sump level sensors in PWRs or the reactor vessel level sensors in BWRs) do not sense a level change and continue to give a false level reading. (The level instruments in question have a water-filled reference leg that can boil dry or rupture if subjected to high temperatures.)
3. Because of the false level readings, a delay occurs in actuating safety systems needed to mitigate the accident.

Although numerous instruments are used by the Engineered Safety Features Actuation System (ESFAS) to sense accidents and initiate appropriate safety system response, these level sensors are among the most important. The unanticipated dependency for the events in this category is the susceptibility of the level instrument to fail because of HELB accident conditions, delaying actuation of ESF systems needed to mitigate the accident.

The vendors, Westinghouse and General Electric, informed the plants of this problem in 1979. Later that year, the NRC issued IE Bulletin 79-21, "Temperature Effects on Level Measurements." In 1982, AEOD published a report entitled "Safety Concern Associated with Reactor Vessel

Level Instrumentation in Boiling Water Reactors." This report and Generic Issue 50, "Reactor Vessel Level Instruments in BWRs,"[1] are both concerned with degradation of safety functions through failure of reactor vessel level instruments. The AEOD report identified the potential for delayed actuation of safety systems as a result of level instrument failures (particularly failures involving the instruments' reference leg). The report did not address failures caused by HELBs.

A total of 21 plants notified the NRC via LERs of the potential for adverse environmental conditions failing important level instruments. Table 4.15 lists these plants. Corrective actions included set point changes, modifications to the instruments, and procedure changes.

Table 4.15. Category 8 — Level instrumentation degraded by high energy line break (HELB) conditions

Plant	Date	Event No.	Description
Beaver Valley 1	06/21/79	10	Potential failure of SG level instruments due to HELB conditions boiling the instrument reference legs dry
Big Rock Point	08/22/79	20	Potential failure of reactor level instruments due to HELB conditions boiling the instrument reference legs dry
Browns Ferry 1	08/09/79	22	Same as event 20
Brunswick 1	09/21/79	27	Same as event 20
Farley 1	06/22/79	62	Same as event 10
Indian Point 2	06/26/79	89	Same as event 10
Indian Point 3	06/21/79	91	Same as event 10
Kewaunee	06/26/79	93	Same as event 10
McGuire 1	06/22/79	95	Same as event 10
North Anna 1	06/21/79	114	Same as event 10
North Anna 2	06/27/79	119	Same as event 10
Robinson 2	06/25/79	160	Same as event 10
Salem 1	07/10/79	166	Same as event 10
Sequoyah 1	06/29/79	178	Same as event 10
Sequoyah 1	06/18/79	183	Potential failure of containment sump level instruments due to HELB conditions rupturing the sensor bellows
Surry 1	06/21/79	194	Same as event 10
Trojan	06/21/79	215	Same as event 10
Turkey Point 3	06/26/79	220	Same as event 10
Turkey Point 4	06/26/79	221	Same as event 10
Watts Bar 1	06/13/79	224	Same as event 10
Zion 1	07/13/79	229	Same as event 10

4.2.9 Category 9 — Loss of containment integrity due to LOCA conditions during purge operations

This category contains postulated events in which containment integrity was lost during a LOCA. The postulated event sequence is as follows:

1. A LOCA occurs while a containment purge is in progress.
2. The resulting high pressure inside containment places stress on the purge system in one of two ways: (a) purge (containment) isolation valves fail to close from their fully open position, or (b) damage occurs to purge system ducts, preventing containment isolation.

The review identified ten plants with this potential problem (Table 4.16). This problem was initially identified by several valve vendors.

Table 4.16. Category 9 — Loss of containment integrity due to LOCA conditions during purge operations

Plant	Date	Event No.	Description
Arkansas Nuclear 2	03/01/78	3	Potential loss of containment isolation capability if LOCA occurs during purge operations (purge valves would not close due to the high delta-P)
Arnold	03/06/79	9	Same as event 3
Browns Ferry 1	02/01/80	23	Potential loss of containment integrity if LOCA occurs during purge operations (pressure surge damages ducts and dampers)
Hatch 1	09/10/79	78	Same as event 3
Monticello	03/01/79	108	Same as event 23
Point Beach 1	03/27/79	144	Same as event 3
Point Beach 2	03/27/79	147	Same as event 3
San Onofre 2	01/16/78	174	Same as event 3
San Onofre 3	01/16/78	176	Same as event 3
Trojan	05/21/79	214	Potential loss of containment integrity if LOCA occurs during purge operations (pressure surge damages purge valves)

The NRC then notified all utilities of potential problems involving the containment isolation valves for the purge system.

Although this accident sequence is considered unlikely to occur, the utilities responded by implementing procedural and/or design changes. For the short term, purge operations were restricted to low pressure/low power conditions in the reactor coolant system. (It is interesting to note that this problem was reported for four General Electric BWRs, three Westinghouse PWRs, and three Combustion Engineering PWRs.)

4.2.10 Category 10 — High energy line break conditions degrading control systems

Category 10 events involved the potential failure of certain non-safety-related control systems that were caused by adverse environmental conditions created by an HELB. Because they were not safety related, these control systems are not required to function under adverse environmental conditions. However, certain failure modes of these controls could degrade the effectiveness of safety systems required to mitigate the HELB accident.

In response to IE Notice 79-22, "Qualification of Control Systems," Westinghouse identified four control systems that could possibly affect a protective function performed by a safety system. These control systems are as follows:

1. Steam generator PORV control system — A main feedwater line break (MFLB) adversely affects the steam generator PORV control system. The PORVs fail open, depressurizing the main steam lines. Thus, no steam is available for the turbine-driven auxiliary feedwater pump.
2. Pressurizer PORV control system — A main feedwater line break adversely affects the pressurizer PORV control system. The PORV fails open, possibly depressurizing the RCS.
3. Main feedwater control system — A small MSLB adversely affects the main feedwater control system. The control system fails such that the water mass in the steam generator is less than anticipated for this break.
4. Automatic rod control system — An intermediate size MSLB adversely affects the excore detectors. The automatic rod control system receives an erroneous signal from the excore detector and subsequently issues a rod withdrawal signal. This can occur before the reactor protection system senses the MSLB and initiates a reactor trip.

Three plants (Table 4.17) reported a susceptibility to one or more of these control failures. For two of the plants (Surry 1 and North Anna 1), the control system failures would create conditions that were less severe than those analyzed in the design basis accidents and, therefore, did not constitute a significant safety concern. For Salem 1 (event 167) no information was given about the severity of the potential problem.

Recently, the NRC has questioned the role of primary system PORVs and the lack of reliability and operability specifications for PORVs and their block valves. Generic Issues 70, "PORV and Block Valve Reliability,"

Table 4.17. Category 10 — HELB conditions degrading control systems

Plant	Date	Event No.	Description
North Anna 1	09/17/79	115	Potential failure of several control systems due to adverse conditions caused by HELB in containment
Salem 1	09/07/79	167	Potential failure of steam generator PORV control system due to adverse conditions caused by HELB in containment
Surry 1	08/29/79	195	Same as event 115

and 84, "CE PORV's," address these concerns.[1] Both Surry 1 and North Anna 1 made procedural changes instructing operators to close the PORV block valves in the event of an HELB.

4.2.11 Category 11 — Auxiliary feedwater pump runout under steam line break conditions

The events grouped in category 11 involved the potential failure of the auxiliary feedwater pumps because of pump runout. The postulated event sequence is as follows:

1. A rupture occurs in either the Decay Heat Removal system header or the steam supply header to the turbine-driven AFW pump.
2. The rupture initiates an uncontrolled blowdown of the secondary system that depressurizes all of the steam generators.
3. Without a supply of steam, the turbine-driven AFW pump is inoperable. Also, because the blowdown reduces the steam generator's back pressure, the motor-driven AFW pumps face low discharge pressures and trip upon reaching runout conditions.

This potential event was addressed in IE Bulletin 80-04, "Analysis of PWR Main Steam Line Break with Continued Feedwater Addition." Two plants (Table 4.18) reported having this potential problem. To correct the problem, both plants (1) installed flow orifices in the motor-driven pumps' discharge lines and (2) modified procedures and operator training for such an event.

Table 4.18. Category 11 — Auxiliary feedwater pump runout under steam line break conditions

Plant	Date	Event No.	Description
Beaver Valley 1	10/03/80	15	Potential loss of AFW due to pump runout under main steam line break conditions
Surry 1	10/16/80	198	Same as event 15

4.2.12 Category 12 — Water hammer events

Since 1969, over 150 incidents occurring at BWRs and PWRs involved water hammer.[9] The water hammer incidents generally involved steam generator feedrings and piping, the RHR system, emergency core cooling systems, and containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor and involved pipe hangers and restraints. However, there have been several incidents that resulted in piping and valve damage.

Unresolved Safety Issue A-1, "Water Hammer," addressed water hammer events; the NRC resolved this issue in March 1984 (Refs. 1 and 2). The results of USI A-1 are summarized below:

1. The total elimination of water hammer events is not feasible due to the possible coexistence of steam, water, and voids in various systems.
2. For the approximately 150 water hammer events that have occurred since 1969, damage has been limited primarily to pipe support systems. In addition, approximately half of these events have occurred either in the preoperational phase or the first year of commercial operation.
3. The frequency of water hammer events peaked in the mid-1970s — a time when the rate of new plants coming into commercial operation was at its highest. Experience also led to corrective design changes that reduced the frequency of occurrence.
4. Steam generator water hammer associated with top feedring steam generators appears to have been corrected through design changes.

Because water hammer concerns have been addressed by USI A-1, this project did not pursue the topic and did not attempt to record all water hammer events that have occurred. It should be recognized, however, that water hammer can represent an undesirable form of system interaction and

needs to be considered in hydraulic design. For example, Category 12 contains four water hammer events that are a result of system interaction (events 33, 94, 231, and 234). These events are listed in Table 4.19. All four events occurred during the initiation of auxiliary feedwater injection. The water hammer events were the result of steam in the feedwater line mixing with cold water from the auxiliary feedwater system.

Table 4.19. Category 12 — Water hammer events

Plant	Date	Event No.	Description
Calvert Cliffs 1	05/23/75	33	Damage to AFW system due to water hammer caused by steam in common feedwater header being quenched by cold auxiliary feedwater
Maine Yankee	01/25/83	94	Damage to AFW system due to water hammer caused by steam in feedwater lines condensing due to cold auxiliary feed water
Zion 2	05/25/76	231	Damage to AFW system due to a water hammer caused by water and steam mixing in the feedwater lines
Zion 2	09/03/80	234	Same as event 231

4.2.13 Category 13 — Common support systems or cross-connects

Eighteen events were assigned to category 13 (Table 4.20). These events resulted from redundant trains or systems failing (or potentially failing) because of (1) the loss of a common support system, (2) the loss of a common component, or (3) the existence of an unisolated piping cross-connection. Five events that also involved common support systems are discussed separately due to their uniqueness.

Common support systems. Support systems, which may be common to many systems, include service water, electric power, HVAC, instrument air supply, etc. These systems provide direct support to other systems. It is recognized and accepted that the loss of a support system can cause the failure of the components it supports. However, it is unacceptable for a single failure to cause redundant safety-related systems (or components) to fail. Therefore, redundant safety-related systems (and components) are designed to preclude single failures. This study identified

Table 4.20. Category 13 — Common support systems or cross-connects

Plant	Date	Event No.	Description
Arkansas Nuclear 1	01/18/73	1	Potential loss of both RB cooler trains due to the loss of a common SW train
Calvert Cliffs 2	07/20/82	39	Loss of both SW trains and one CCW train due to the loss of a common discharge header
Farley 1	11/21/78	61	Potential loss of both CCW trains due to the rupture of a cross-connect pipe at a charging pump
Farley 1	11/12/80	64	Potential loss of both SW trains due to any failure that could cause the loss of one SW train
Farley 2	11/12/80	67	Same as event 64
Hatch 1	05/24/80	80	Potential loss of two RHR trains and one core spray train due to a leaky RHR isolation valve plus DBA
Hatch 1	07/11/80	81	Loss of all LPCI due to loss of LPCI inverter room cooler common to each train
Indian Point 2	10/19/77	88	Potential loss of both containment isolation valves on air ejector diversion line due to the loss of a common electric power source
Midland 1	07/22/83	98	Potential loss of two steam supply valves to the AFW turbine due to loss of offsite power and no dc backup
Midland 2	07/22/83	99	Same as event 98
Monticello	03/03/81	109	Loss of two RHR SW trains due to the loss of a common seal water supply
Nine Mile Point 1	10/14/76	110	Potential loss of two containment spray pumps due to the actuation of a common lockout switch
North Anna 1	10/05/78	113	Loss of both containment atmosphere monitoring trains due to the loss of a common power supply
Oyster Creek	03/07/83	132	Potential loss of both SGTS trains due to backflow through a common discharge duct
Oyster Creek	04/06/83	133	Potential loss of both SGTS trains due to the loss of a common power source
Palisades	09/16/77	136	Potential loss of all six containment isolation valves on purge lines due to the loss of a common air supply
Surry 1	09/19/74	191	Potential loss of both AFW trains due to the rupture of cross-connect piping
Surry 2	09/19/74	202	Same as event 191

seven events where the loss of a single support system could cause redundant safety-related systems to fail (events 1, 81, 88, 109, 113, 133, and 136). For example, at ANO 1, all reactor building coolers could fail if a single train of service water was lost (event 1). That one train of service water provided coolant to both RB cooler trains.

Common components. In addition to identifying events involving shared support *systems*, this study also identified events involving shared *components*. In many cases, the systems were knowingly designed this way (e.g., some systems such as main feedwater and auxiliary feedwater share a common discharge header). This study identified two events where a failure in a common discharge line caused the failure of multiple trains. At Calvert Cliffs 2 (event 39), both service water trains and a component cooling water train were lost when a valve in the discharge line transferred closed. Oyster Creek identified a potential reduction of SGTS efficiency when discharge from the operating train backflowed into the redundant train. The redundant train was out of service, and its inlet and outlet valves had transferred open (event 132).

In some cases, redundant safety-related systems (or components) shared components unknowingly. For example, during a review of the plant design at Nine Mile Point 1, personnel discovered that the control switch for one containment spray pump locks out the sister pump (redundant pump in the same train), thus preventing it from automatically starting (event 110).

Unisolated piping cross-connection. Redundant systems can also be rendered inoperable because of unisolated cross-connect piping if a piping rupture occurs. The existence of a cross-connection may or may not be recognized by the operating staff at the time. For example, at Farley 1, it was recognized that both CCW trains are physically connected to all three charging pumps. However, on one occasion, both CCW trains were accidentally cross-connected because the operating procedures failed to specify that a charging pump should be supported by only one CCW train at a time (event 61).

Another example of an undesirable cross-connection was reported at Surry 1 and 2 in 1974. A cross-connection between AFW trains was installed during a design change. Checkout during installation discovered that no isolation valves were included. All auxiliary feedwater flow could be lost if piping in either one of the two trains ruptured (events 191 and 202).

Five other events that were placed in category 13 are unique events in that they are not similar to the groups above. At Farley 1 and 2 (events 64 and 67), personnel discovered that both service water trains would be lost if a failure rendered one train unavailable. The potential for a total loss of service water existed because train A cooled train B components and vice versa. At Midland 1 and 2 (events 98 and 99), personnel discovered that the two steam supply valves to the turbine-driven AFW pump would close on loss of offsite power. Consequently, the pump would be unavailable because there was no backup (dc) power supplied to the steam supply valves. In a potential problem found at Hatch 1 (event 80), Bechtel notified the utility that both RHR trains and one of the core spray trains could be disabled by a leaking RHR isolation valve. If a recirculation line broke and a certain motor control center failed while

the plant was in operation with the leaking isolation valve, an RHR heat exchanger would be pressurized, disabling both RHR trains.

The project did not identify any unresolved safety issue or generic issue that specifically evaluate common support systems or cross-connects. However, basic regulations prohibit such dependencies by requiring independent safety system trains.

4.2.14 Category 14 — Instrument power failures affecting safety systems

Five events were classified as category 14 (Table 4.21). In each instance, plant control was or could be adversely affected by instrument power failure (events 48, 72, 73, 123, and 155).

Table 4.21. Category 14 — Instrument power failures affecting safety systems

Plant	Date	Event No.	Description
Crystal River 3	02/26/80	48	Loss of multiple NNI instruments due to loss of 24-V dc power causing transient
Ginna	04/22/71	72	Potential loss of SI pumps due to the loss of BAST level channels (following loss of power to the instrument buses) that prevents the pump suction valves from opening
Ginna	10/21/73	73	Loss of power to instrument bus caused loss of level indication for BAST and premature safety injection pump switch to RWST
Oconee 3	11/10/79	123	Loss of indication for systems required for shutdown due to instruments being fed from non-Class 1E NNI inverter
Rancho Seco	03/20/78	155	Loss of NNI and lack of plant control due to loss of dc power causing transient

Because of the safety significance of this type of event, the NRC Office of Inspection and Enforcement has issued several IE notices, circulars, and bulletins. IE Bulletin 79-27, "Loss of Non-Class 1-E Instrumentation and Control Power System Bus During Operation," required the licensees to investigate the loss of individual power supplies as well as the total loss of an inverter or vital bus. The bulletin required licensees to review all Class 1E and non-Class 1E buses that supply power to safety-related and non-safety-related instrumentation and control systems whose failure could affect the ability to achieve cold shutdown condition. In addition, licensees were told to review their existing procedures (or to prepare emergency procedures) that are used to achieve a cold shutdown condition upon loss of power to each Class 1E and non-Class 1E bus that supplies power to safety-related and non-safety related instrumentation and control systems. Licensees were also required to review again IE Circular 79-02, "Failure of 120 Volt Vital AC Power Supplies" and to include in their review both Class IE and non-Class IE safety-related power supply inverters.

The implications of the loss of non-class IE power supply buses inhibiting the ability to achieve cold shutdown is of continuing concern to the NRC. The NRC is currently studying the safety implications of instrument power failures in Unresolved Safety Issue A-47, "Safety Implications of Control Systems" and in Generic Issues 19, "Safety Implications of Non-safety Instrument and Control Power Supply Bus," and 76, "Instrumentation and Control Power Interactions."[1,2]

4.2.15 Category 15 — Inadequate cable separation

Eight events (seven potential) were identified where redundant Class IE cables were not adequately separated (Table 4.22). Separation problems were created as a result of cables being routed through the same area or same cable tray (events 21, 46, 55, 56, 57, 142, 153, and 208). The potential loss of redundancy could occur because of a fire or some other event that damaged cables in a specific location.

An example of the problems that inadequate separation can pose occurred at Browns Ferry 1 on March 22, 1975 (event 21). A fire broke out in an electrical cable penetration between the cable spreading room and the reactor building. The fire spread horizontally and vertically to all ten cable trays within the penetration. The plant was shut down safely, but because of the fire, normal shutdown cooling systems were inoperable. In addition, part of the ECCS was degraded.

The basic cause of the fire was failure to recognize the significance of the flammability of the materials involved. The immediate cause of the fire was the ignition of polyurethane used for cable penetration sealing material. Construction workers checking for air leaks in penetration used a candle flame to detect air flow. The candle flame ignited the polyurethane.[10] Since this event, the seriousness of fires in nuclear plants has been realized by both the utilities and the NRC; fire prevention and protection have received additional attention.

Table 4.22. Category 15 — Inadequate cable separation

Plant	Date	Event No.	Description
Browns Ferry 1	03/22/75	21	RHR, ECCS, and auxiliary systems degraded due to fire in cable spreading room
Cooper	10/16/78	46	Potential loss of redundancy in safety systems due to a Division I cable for a HPCI valve being routed in a Division II riser
Diablo Canyon 1	10/06/78	55	Potential failure of multiple safety systems due to inadequate separation of Class I circuits
Diablo Canyon 2	10/06/78	56	Same as event 55
Dresden 2	02/02/78	57	Potential loss of dc power for safety systems due to inadequate separation of dc power sources
Pilgrim 1	01/21/80	142	Potential degradation of ECCS by fire due to common power cable locations
Quad Cities 2	05/21/79	153	Potential degradation of multiple safety systems (by fire, impact, etc.) due to common power cable location
Susquehanna 1	11/01/77	208	Potential degradation of ESF control cables (by fire, impact, etc.) due to inadequate cable separation

4.2.16 Category 16 — Safety-related cables unprotected from missiles generated from HVAC fans

Three events were classified as category 16 (events 41, 227, and 228). All three events were identified through the construction deficiency file and are listed in Table 4.23. The potential for this type event was identified by the Buffalo Forge Company (the fan vendor).

Table 4.23. Category 16 — Safety-related cables unprotected from missiles generated from HVAC fans

Plant	Date	Event No.	Description
Clinton 1	06/26/81	41	Potential damage to safety-related cables due to HVAC fan failure propelling missiles through fan housing
WPPSS 1	06/11/81	227	Same as event 41
WPPSS 4	06/11/81	228	Same as event 41

While recalculating the fan housing thickness, which is required to prevent a fan blade from penetrating the housing, Buffalo Forge determined that the fan housings were not of sufficient thickness to prevent penetration by the fan blades.

Unacceptable damage to essential systems caused by missiles can occur as a result of (1) ejection of an energetic missile, (2) a missile striking a critical component, and (3) unacceptable damage occurring to an essential system or component due to the missile strike.[1]

The three events identified in Table 4.23 satisfy the necessary conditions that could result in an essential system being damaged by a missile. The possibility of a fan blade penetrating the fan housing satisfies criterion (1) above. Criteria (2) and (3) could be satisfied because safety-related cables were in the vicinity of the fans. Corrective actions at each of the plants consisted of removing the possibility for the ejection of an energetic missile. Plant personnel welded reinforcing plates to each of the existing fan housings. This type of corrective action eliminates the need for analysis of the potential for fan blades impacting safety-related cables and the damage that could occur.

The NRC has addressed the problem of missiles generated from turbines (USI A-37, "Turbine Missiles"), tornados (USI A-38, "Tornado Missiles"), and BWR recirculation pumps or PWR main coolant pumps (USI B-68, "Pump Overspeed During LOCA").[1] The project staff found no unresolved safety issues or generic issues specifically addressing the evaluation of the probability of unacceptable damage to essential systems caused by missiles generated from fans. However, basic regulations require utilities to evalute all potential sources of missiles.

4.2.17 Category 17 — Suppression pool swell

In 1975, General Electric Company analyses indicated that the occurrence of a large LOCA could cause suppression pool swell. The problem concerned all Mark I and Mark II containment structures because certain

structural loadings were not considered in the original containment design calculations. For 19 operating facilities with Mark I containments, the design safety margins of the containment structure under LOCA conditions were not as large as originally planned (Table 4.24).[11] Eighteen of the facilities increased their margin of safety simply by instituting special operating procedures that reduced the pool dynamic loads. The nineteenth facility, Vermont Yankee (event 223), added structural supports and instituted a differential pressure mode of operation for the containment system. This reduced the potential accident loads to acceptable values.

Table 4.24. Category 17 — Suppression pool swell

Plant	Date	Event No.	Description
Oyster Creek	12/20/76	126	Potential torus damage due to stresses created by relief valve operation
Susquehanna 1	03/06/75	207	Potential torus damage due to suppression pool swell caused by LOCA or safety relief valve actuation
Vermont Yankee	01/30/76	223	Potential containment structure damage due to suppression pool swell created by LOCA forces (applies to 19 Mark I containments)

In addition to the 19 plants that identified the possibility of damage to the suppression pool as a result of LOCA forces, two plants identified the possibility of damage to the suppression pool because of the actuation of the safety relief valves. One of the plants has a Mark I containment (event 126); the other plant has a Mark II containment (event 208).

The pool swell phenomenon and the associated hydrodynamic loads have been a concern of the NRC. In fact, five unresolved safety issues, listed below, address this phenomenon.

1. USI A-6, "Mark I Short-Term Program";
2. USI A-7, "Mark I Long-Term Program";
3. USI A-8, "Mark II Containment Pool Dynamic Loads — Long Term Program";
4. USI A-39, "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits"; and
5. USI B-10, "Behavior of BWR Mark III Containments."

The technical resolutions for USIs A-6, A-7, A-8, and A-39 have been completed. For USI B-10, the Mark III suppression pool dynamic loads were reviewed by the NRC at the construction permit stage for Grand Gulf Nuclear Station Units 1 and 2. The NRC staff is currently reviewing GE's pool dynamic load calculations to arrive at a final hydrodynamic load definition that can be used by all Mark III containment applicants for operating licenses.

4.2.18 Category 18 — Scram discharge volume degradation

Two events were identified as category 18 (Table 4.25). In event 26, the ability to scram was lost when the reactor building equipment drain tank failed to allow water to drain from the scram discharge volume (SDV). In event 59, the potential for loss of the ability to scram (for the same reason as event 26) was discovered during a test.

Table 4.25. Category 18 — Scram discharge volume degradation

Plant	Date	Event No.	Description
Browns Ferry 3	06/28/80	26	Loss of ability to scram due to RB equipment drain tank not allowing water to drain from SDV
Dresden 3	07/19/80	59	Potential loss of ability to scram control rods due to RB equipment drain tank not allowing water to drain from SDV

When a BWR is scrammed, the scram inlet and outlet valves associated with each control rod drive are opened. This applies high-pressure water under the control rod drive piston and vents the upper side of the piston to the SDV (the SDV is normally at atmospheric pressure). This produces a large upward force on the piston that drives the control rod up into its fully inserted position. The SDV receives the "exhaust" water from all of the control rod drives during a scram. The SDV must be large enough to accommodate all of this water so that the scram motion is not impeded.[2]

Investigations of the Browns Ferry 3 event (event 26) determined that the loss of ability to scram was caused by water accumulation in the SDV header. At the time of the first scram ~40% of the control rods failed to insert. The water accumulation reduced the available free volume in the SDV for water discharge from a scram, thereby inhibiting the

insertion of the control rods. Water accumulated in the SDV because flow from the SDV into the reactor building equipment drain tank was restricted.

An NRC review performed after the Browns Ferry event determined that long-term hardware improvements in the isolation valve arrangements for the SDV system were needed. The NRC noted that the SDV vent and drain lines at several BWRs were normally equipped with a single isolation valve. However, an NRC safety criterion states that no single failure shall create an uncontrolled loss of reactor coolant. The failure of either a vent valve or drain valve could result in an uncontrolled loss of reactor coolant following a reactor scram. The NRC noted that an acceptable method of satisfying the single failure criterion would be to provide two isolation valves in series in all SDV vent and drain lines.

In a related NRC review, AEOD evaluated the added (temporary) SDV instrumentation arrangement at Browns Ferry 3 in terms of its acceptability for continued operation. Their review, reported as IE Notice 80-30, "Potential for Unacceptable Interaction Between the Control Rod Drive Scram Function and Non-Essential Control Air at Certain GE BWR Facilities," concluded that a thorough evaluation was needed of the potential for the unacceptable interaction between the control rod drive system and the nonessential (nonsafety) control air system. No positive position indication (other than full open) for the scram inlet and outlet valves existed, and potential problems existed if a partial loss of control air occurred. During a slow loss of control air pressure, the scram valves would drift open slowly without any position indication being given to the operator. The loss of air pressure would lead to a significant SDV in-leakage, but the control rods might not move until the pressure decreased substantially.

Approximately 2 years after the event at Browns Ferry 3, an SDV drain valve failed to close at Hatch 2 (event 86). (This event is a category 23 event but is also discussed in this category because of the SDV drain valve failure.) Because of the resulting blowdown, a "high dry-well pressure" scram signal occurred. The loss of reactor coolant through the drain valve could not be terminated because the high dry-well pressure scram signal could not be cleared or bypassed. (The high dry-well pressure could not be reduced by normal means because the dry-well chillers were unavailable. The load shedding logic that was initiated by the high dry-well pressure condition caused the dry-well chillers to trip.) The continuous scram signal prevented a routine reclosure of the upstream scram outlet valves via the reset of the reactor protection system (RPS). This incident could have been avoided had the required NRC surveillance requirements (that resulted from the Browns Ferry 3 incident) been in place and implemented.[12]

The NRC has evaluated BWR SDV problems in four generic issues, which are listed below:[1]

1. GI-25, "Automatic Air Header Dump on BWR Scram System";
2. GI-39, "Potential for Unacceptable Interaction Between the CRD System and Non-essential Control Air System";
3. GI-40, "Safety Concerns Associated with Pipe Breaks in the BWR Scram Systems"; and
4. GI-41, "BWR Scram Discharge Volume Systems."

4.2.19 Category 19 — Induced human interactions

Four events were classified as category 19 (Table 4.26). Incorrect procedures were the cause of the first event (event 112). The procedures required personnel to enter the reactor building following a LOCA to vent the primary containment. Because the reactor building may not be accessible following a LOCA, the containment may not be vented using that procedure. Consequently, the potential to overpressurize the containment existed. A procedural change was made that allowed venting without entering the reactor building. In addition, another procedural change was made that established a redundant purge path.

Table 4.26. Category 19 — Induced human interactions

Plant	Date	Event No.	Description
Nine Mile Point 1	01/29/82	112	Potential to overpressurize containment since procedures require access to RB during a LOCA to vent containment
Palisades	09/08/71	134	Loss of power to a relief valve's pilot valve solenoid control circuit (the technician was misled by plant drawings)
Point Beach 2	12/19/74	146	Dependency between RHR and RCS introduced by human error after two valves were left open during an SI pump test
TMI-2	03/28/79	213	Fuel damage resulted after operator shut off safety injection system (operator was unaware of true plant conditions due to inadequate instrumentation)

The second event (event 134) resulted from the use of a nonstandard contact designation in the plant drawings of the control circuit to the pressurizer PORVs. The nonstandard designation of the contacts led a technician to believe that the PORV would remain closed when the RPS breakers were deenergized. However, after the technician deenergized the RPS breakers, the solenoids on the pressurizer PORV deenergized. This caused the valve to open and in turn caused a primary system blowdown. The reactor pressure dropped to ~1280 psia in the 2 to 3 min before an

operator could close the motor-operated block valve. The drawings were corrected to show the as-built conditions and to conform with standard notation.

Procedural errors also caused the occurrence of event 146. During a refueling outage, two manually operated isolation valves were installed on the cross-connect between the two safety injection banks. However, the procedures were not reviewed after these valves were installed. Consequently, during a test of a safety injection pump, the RCS and RHR were momentarily pressurized to 1400 psig (the RHR design pressure is 600 psig). The dependency between the RCS and RHR was introduced because the modified valve lineup for the test did not include closing the two, newly installed valves.

As a result of the occurrence of the fourth event, the accident at TMI-2 (event 213), many new requirements for operating reactors were implemented. These requirements included more operator training, equipment and instrumentation modifications, control room design analyses, and human factors analyses. The event began when a pressurizer PORV stuck open causing a small LOCA. Because of the resulting loss of RCS inventory, the safety injection system actuated. However, due to inadequate instrumentation, the operator shut off the safety injection systems. The loss of RCS inventory and the shutting off of the safety injection system resulted in fuel damage. (The AFW also failed, but its failure was not caused by the interaction of systems.)

4.2.20 Category 20 — Functional dependencies due to failures during seismic events

Five potential events were classified as category 20 and are listed in Table 4.27. The events involved either mechanical failures (events 14 and 139) or electrical failures (events 141, 187, and 190).

In the two events involving mechanical failures, a potential interaction between seismic and nonseismic qualified components existed. Beaver Valley 1 (event 14) reported the potential loss of RHR cooling. The stresses from an earthquake could cause the failure of a nonseismic qualified branch line in the CCW system. To stop the resulting leak through the branch line, operators would have to isolate the entire CCW header. Peach Bottom 2 (event 139) reported the potential to lose emergency service water. Personnel discovered that an earthquake could damage the reactor building CCW heat exchanger. A seismic qualified valve isolates the service water system from the nonseismic qualified CCW heat exchanger. However, because the valve is normally aligned in the open position, a seismic event severe enough to damage the CCW heat exchanger could fail the emergency service water system.

Three events (events 141, 187, and 190) involved the potential failure of nonseismic qualified breakers. In each case, the nonqualified breakers had to successfully disconnect to allow reenergizing of the power system from qualified sources. The concern in each event was the potential for the breakers to fail to disconnect or to cause short circuits in the power system (during a seismic event).

Table 4.27. Category 20 — Functional dependencies due to failures during seismic events

Plant	Date	Event No.	Description
Beaver Valley 1	09/12/80	14	Potential loss of RHR due to earthquake stress breaking 2-in. branch line of 24-in. CCW line, which would require CCW line to be isolated
Peach Bottom 2	04/11/79	139	Potential loss of emergency SW due to a seismically qualified valve in the SW system being aligned in the open position to the non-seismically qualified CCW system
Pilgrim 1	08/16/79	141	Potential failure of DG output breakers to close (following a seismic event) because nonseismic auxiliary transformer breakers fail to trip
St. Lucie 1	03/31/78	187	Potential to lose emergency power during a seismic event because normal and emergency power share bus with non-Class 1E contacts
St. Lucie 2	01/24/78	190	Same as event 187

The NRC currently has four unresolved safety issues that deal with seismic events:[1,2]

1. USI A-40, "Seismic Design Criteria — Short Term Program";
2. USI A-41, "Long-Term Seismic Program";
3. USI A-46, "Seismic Qualification of Equipment in Operating Plants"; and
4. USI B-24, "Seismic Qualification of Electrical and Mechanical Equipment."

The main objectives of these issues are (1) to establish a set of guidelines to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants and (2) to better understand the inherent conservatisms in seismic design.

4.2.21 Category 21 — Spatial dependencies due to failures during seismic events

Thirteen events were identified as category 21 (Table 4.28). Each event involved the potential for the interaction of multiple systems or components during a seismic event because of spatial relationships (i.e., common locations). The events involved (1) the potential for masonry walls to collapse on safety-related components (events 65, 68, 107, 143, 145, 181, 201, 212, and 216); (2) the potential for a nonseismic qualified duct to fall on safety-related equipment (event 69); (3) the potential for the control room habitability to be lost because of the control room HVAC not being isolated (event 211); and (4) the potential for flooding of an HPSI pump room because of the rupture of an inadequately supported fire protection pipe (events 225 and 226).

IE Bulletin 80-11, "Masonry Wall Design," described the potential for masonry walls to collapse on safety-related equipment at Trojan (event 216). In addition to collapsing on other components, the failure of the walls could also degrade safety-related equipment that depends on the walls for support. Events 65, 68, 143, 145, 181, and 201 were all reported in response to IE Bulletin 80-11.

The NRC currently has four unresolved safety issues that deal with seismic events. These USIs are listed in category 20. In addition to IE Bulletin 80-11, the NRC Office of Inspection and Enforcement has issued the following relevant documents:

1. IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts";
2. IE Bulletin 79-14, "Seismic Analysis for As-Built Safety Related Piping Systems"; and
3. IE Notice 79-28, "Overloading of Structural Elements Due to Pipe Support Loads."

4.2.22 Category 22 — Other functional dependencies

Twenty-one events were placed in category 22 (Table 4.29). This category includes all of the functionally coupled events that were not assigned to any other category. There is no other apparent commonality among the events.

This category does not contain all of the functionally coupled ASIs identified by the project. Several other categories also contain events whose dependencies are primarily functional — categories 1, 2, 5, 6, 7, 9, 11, 12, 13, 14, 17, 18, and 20. However, events in those categories exhibited some other commonality and were categorized based on that aspect.

Several of the problems demonstrated by events in this category are the topics of generic issues. At Crystal River 3 and Surry 1 (events 47 and 196, respectively) the RCS boron concentration was inadvertently reduced. These events have been evaluated in Generic Issue 22, "Inadvertent Boron Dilution Events." The significance of boron dilution lies in the insertion of positive reactivity with the possibility of inadvertently achieving criticality.

Table 4.28. Category 21 — Spatial dependencies due to failures during seismic events

Plant	Date	Event No.	Description
Farley 1	12/09/80	65	Potential damage to multiple safety systems due to non-seismic equipment support walls falling during a seismic event
Farley 2	12/09/80	68	Same as event 65
Fermi 2	03/10/82	69	Potential loss of safety-related equipment due to nonseismic HVAC duct (over safety-related equipment) falling during a seismic event
Millstone 2	12/05/83	107	Potential radiological release due to nonseismic wall falling on nearby safety-related HVAC equipment during a seismic event
Pilgrim 1	10/08/81	143	Potential loss of several safety-related systems due to nearby masonry walls collapsing on equipment during a seismic event
Point Beach 1	07/14/81	145	Potential loss of safety equipment required for shutdown due to block walls in the control building falling on nearby equipment during a seismic event
Sequoyah 1	11/14/80	181	Potential loss of safety equipment required for shutdown due to block walls in the auxiliary building falling on nearby equipment during a seismic event
Surry 1	07/24/81	201	Potential loss of spent fuel pool integrity due to block walls in the fuel building falling into the pool during a seismic event
TMI-1	02/25/82	211	Potential loss of control room habitability due to control room HVAC not being isolated from control building (duct damaged during a seismic event)
TMI-2	08/28/75	212	Potential damage to safety systems due to hollow wall collapsing on nearby safety-related electrical cabling during a seismic event
Trojan	05/08/80	216	Potential damage to safety-related components due to collapse of masonry walls during a seismic event
Watts Bar 1	08/08/83	225	Potential flooding of HPSI pump (during seismic event) due to rupture of an overhead fire protection system pipe
Watts Bar 2	08/08/83	226	Same as event 225

Table 4.29. Category 22 — Other functional dependencies

Plant	Date	Event No.	Description
Big Rock Point	10/31/77	19	Loss of containment integrity due to connecting an external plant heating system to RCS
Browns Ferry 3	03/04/76	25	Potential damage to all RHR pumps under LPCI operation (during LOCA) due to runout flow being exceeded
Brunswick 2	02/27/75	29	Potential for seven ADS valves to fail open due to a trickle current holding their solenoids in the actuated position
Brunswick 2	01/05/76	31	Loss of HPCI due to the leak detection system improperly isolating the turbine steam supply line
Crystal River 3	02/07/77	47	RCS boron concentration was diluted after NaOH drained from the NaOH tank into the RHR system
Davis-Besse 1	07/30/81	54	Loss of secondary containment after a containment purge fan tripped causing an overpressure blow-out panel to open
Grand Gulf 1	08/04/83	76	Numerous instruments damaged due to a cable being connected between 125-V ac power and 125-V dc power
Hatch 2	01/28/80	84	Potential loss of multiple safety system during a LOCA due to torus water entering the CST via core spray suction piping
Kewaunee	11/05/75	92	Loss of AFW due to resin beads from make-up water demineralizers leaking into CSTs
North Anna 3	02/08/80	121	Potential overpressurization of containment following an MSLB due to AFW injection causing a long-term blowdown
Oyster Creek	05/02/79	127	Erroneous reactor water level indications (following a LOCA) due to inadequate flow from isolation condenser to vessel annulus via recirculation discharge valve bypass line
Palisades	08/19/82	137	Potential loss of all SW during a LOCA due to SW pump runout
Palisades	11/30/82	138	Potential loss of systems required to mitigate a LOCA due to two motor control centers becoming overloaded
Prairie Island 1	04/12/79	149	Potential failure of ESF systems to automatically start during a small LOCA (ESFAS logic not satisfied when pressurizer pressure decreases but level does not)
Rancho Seco	09/20/74	154	Loss of multiple control devices and operator displays due to a single power source being disconnected during maintenance
Robinson 2	05/01/75	158	Loss of all three RCPs due to a broken seal in one pump overpressurizing the common seal leakoff line and preventing seal leakoff flow
Robinson 2	01/13/81	161	Potential loss of containment integrity due to backflow through leaky SW line (following a LOCA)
Robinson 2	01/29/81	162	Reactor coolant released to containment after an SI actuation due to a CVCS letdown line end-cap being blown off
Sequoyah 1	05/25/80	179	Loss of one train (each) of RHR and containment spray due to an FW valve failing to actuate (the RHR valve was interlocked with the FW valve)
Surry 1	05/12/80	196	Dilution of RCS boron concentration due to water flowing from the RWST into the RCS
Zion 2	12/11/81	235	Failure of both motor-driven AFW pumps to auto start due to the simultaneous start of both pumps causing their sensed suction pressures to drop below the trip set point

Resin-bed-type demineralizer failures have occurred in both nuclear and other power plants. Generally, process systems that use these types of demineralizers do not perform any reactor protection or engineered safeguards functions, yet their failure may seriously impair the effectiveness of safety-related systems. At Kewaunee (event 92), the AFW system was lost when resin beads leaked from a make-up water demineralizer into the condensate storage tanks (CSTs) and clogged the AFW strainers. (The CSTs are the preferred source of water to the AFW system.) These type failures are under consideration in Generic Issue 71, "Failure of Resin Demineralizer Systems and Their Effects On Nuclear Power Plant Safety."[1]

4.2.23 Category 23 — Other spatial dependencies

Thirty events were placed in category 23 (Table 4.30). This category includes all of the spatially coupled ASIs that were not assigned to other categories. The other categories that also contain predominantly spatially coupled events are 3, 4, 8, 10, 15, 16, and 21. Those categories, however, have some other commonality that was considered significant enough to be addressed separately.

Although there is not a single commonality among the spatially coupled events in category 23, there are three specific (and one general) spatial problems that are represented. These problems are (1) flooding (5 events), (2) water leaking or splashing from one component onto another component (4 events), (3) excessive moisture in the containment atmosphere (5 events), and (4) other spatial dependencies (16 events).

Flooding. Four of the five events that involved flooding (events 90, 100, 150, and 192) occurred because of a leak in systems designed to supply large volumes of water (service water system and the circulating water system). The severe flooding event at Indian Point 2 (event 90) prompted the NRC to issue IE Bulletin 80-24, "Prevention of Damage Due to Water Leakage Inside Containment." In this event, almost 100,000 gal of service water flooded the reactor vessel pit. The bulletin requested a summary description of all open-loop cooling water systems inside containment. An open-loop water system is of interest because the system draws from an indefinite volume of water, such as a river. Consequently, leakage from the system cannot be detected by decreases in inventory. Also, the system may provide a direct pathway for radioactive releases to the outside environment should a LOCA occur simultaneously with a system leak inside containment. In addition, the NRC has studied the issue of flooding from these type systems in Generic Issue 58, "Containment Flooding."[1]

In the fifth flooding event (event 217), the containment sump was flooded and the valve operator to a containment isolation valve for the RCS drain tank line became submerged. Consequently, the RCS drain tank could not be isolated.

Leaks or splashing. Components do not necessarily have to be submerged to be affected by water. Water leaking, splashing, or spraying onto nearby components can cause those components to fail (events 130, 203, 205, and 222). For example, at Surry 2, a service water pump was lost when water from the other service water pump was splashed onto it

Table 4.30. Category 23 — Other spatial dependencies

Plant	Date	Event No.	Description
Arkansas Nuclear 1	09/06/77	2	Potential loss of safety-related equipment due to high temperatures following LOCA and loss of offsite power
Bellefonte 1	11/01/76	17	Potential loss of RB coolers due to LOCA conditions causing boiling in cooler tubes
Browns Ferry 1	02/10/80	24	Potential loss of dry-well isolation via CCW line if recirculation line break (LOCA) impinges on CCW line and isolation valve fails.
Brunswick 2	04/05/77	32	Loss of HPCI due to delta-T leak detection instruments falsely isolating steam supply to HPCI turbine
Connecticut Yankee	08/25/78	43	Erroneous turbine runback and automatic rod block given due to electromagnetic interference from radio transceiver
Cook 2	03/26/82	45	Potential degradation of ice condensers due to temperature-gradient-induced air currents (caused by heat conducted through crane wall) causing ice migration
Ft. Calhoun 1	05/19/82	70	Potential loss of AFW due to a steam supply line break since turbine-dirven and motor-driven pumps were in the same area
Hatch 1	11/05/81	82	Potential erroneous isolation of HPCI and RCIC steam supply lines on main steam line or scram discharge line break
Hatch 1	08/25/82	86	Loss of RCIC due to SDV valve leak that caused hot, humid atmosphere for electrical equipment
Hatch 2	02/03/84	87	Torus vent header cracked when liquid nitrogen entered purge line (due to vaporizer failure) and impinged on the header
Indian Point 2	10/17/80	90	SW leaks from containment fan coolers flooded reactor cavity wetting lower portion of reactor vessel
Midland 1	07/11/79	97	Potential loss of RB coolers due to LOCA conditions causing boiling in SW lines to air coolers
Millstone 1	05/01/71	100	Loss of dc MCC due to flooding by SW heat exchanger leak
Nine Mile Point 1	01/07/81	111	Loss of DG voltage regulator and trip of output breaker due to pieces of bailing wire (used to secure fire proofing frames) falling into control cabinet

Table 4.30 (continued)

Plant	Date	Event No.	Description
Oconee 3	12/07/78	122	Loss of both RB vent system filter trains due to FW valve leak
Oconee 3	03/03/81	124	Heat and moisture damage to emergency power switching cables due to FW valve leak
Oyster Creek	01/18/83	130	Loss of one core spray pump due to CRD pump vent line leakage.
Oyster Creek	03/06/83	131	Loss of one train SGTS when flow switch was damaged by nearby space heater
Peach Bottom 2	04/17/80	140	Potential loss of some ESF systems when a HPCI steam-line break causes a wall holding ESF cabling to fall
Quad Cities 1	06/10/72	150	Loss of DG cooling and SW due to circulating water pipe break and subsequent flooding of turbine building
Rancho Seco	02/19/80	157	Potential damage to reactor vessel, internals, control rod drives, and spent fuel rods to load from polar crane being dropped
Salem 1	11/07/78	165	Loss of all five RB coolers due to erroneous isolation of SW by radiation monitors
Salem 1	11/06/81	169	Trip of one vital power inverter (due to electromagnetic interference from cabinet fan) plus unit shutdown conditions gave reactor trip and safety injection
San Onofre 1	03/12/68	170	Fire in electrical penetration caused by overloaded pressurizer heater cables
St. Lucie 1	06/11/80	188	Reactor trip on loss of CCW to RCS pumps when a steam leak in an SG blowdown line caused a CCW valve to close
Surry 1	01/17/77	192	Loss of SW due to flooding of four SW isolation valves when an SW drain valve was left open
Surry 2	04/29/81	203	Loss of one SW pump due to water splashing from nearby SW pump during maintenance
Surry 2	09/12/83	205	Loss of one AFW pump due to water leaking from the roof
Trojan	04/19/81	217	Loss of containment isolation valve for RCS drain tank due to flooding of the containment sump
Turkey Point 4	09/06/82	222	Pressurizer spray valve leaked on its I/P converter that caused spray valve to open

during maintenance (event 203). (Category 3 also contains events where components were affected by the spray from fire protection systems.)

Excessive moisture. Steam can also cause components, especially electrical components, to fail because of excessive humidity (events 70, 86, 122, 124, and 188). Piping leaks or ruptures can produce high temperatures and humidity in surrounding areas. For example, personnel at Fort Calhoun discovered that all of the AFW pumps could be disabled by a single break of the steam supply line to the turbine-driven AFW pump (event 70). The break, if it occurred inside the pump room, would disrupt the supply of steam to the turbine-driven pump and create an adverse environment, disabling the (electric) motor-driven pump. The NRC created Generic Issue 68, "Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture," to study this problem.[1]

Miscellaneous spatial events. No specific problem area exists for the remaining 16 events in this category. Two of these events, however, fall under the scope of two USIs. USI A-36, "Control of Heavy Loads Near Spent Fuel," evaluated the control of lifting heavy loads. At Rancho Seco (event 157), a sling on the reactor building polar crane broke and dropped a 3000-lb load into the fuel transfer canal. USI B-54, "Ice Condenser Containments," evaluated ice condenser designs.[1] It originated after the NRC expressed concern over the possibility of nonsymmetric ice losses caused by sublimation. At Cook 2 (event 45), ice loss and migration occurred as a result of temperature-gradient-induced air currents (caused by heat conducted through the crane wall). This could degrade the effectiveness of the ice condensers.

REFERENCES

1. R. Emrit et al., *A Prioritization of Generic Safety Issues*, NUREG-0933, U.S. Nuclear Regulatory Commission, December 1983.

2. U.S. Nuclear Regulatory Commission, *Unresolved Safety Issues Summary (Aqua Book)*, NUREG-0606, Vol. 6, No. 2, May 18, 1984.

3. U.S. Department of Energy, *DOE/RECON User's Manual*, DOE-TIC-4586, Rev. 1, May 1981.

4. *Sequence Coding and Search System Coder's Manual for Licensee Event Reports*, ORNL/NSIC-189, Rev. 0, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., March 1984.

5. U.S. Nuclear Regulatory Commission, *Foreign Event File (FEF) Data Base Description*, March 1984.

6. *CERCRS Data Base Development Project Final Report and User's Handbook*, ORNL/NSIC-221, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., January 1984.

7. The Institute of Nuclear Power Operations, *Nuclear Plant Reliability Data System*, Atlanta, Georgia, 1983.

8. *The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report*, ORNL/TM-8271, NUREG/CR-2641, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., July 1982.

9. A. W. Serkiz et al., *Evaluation of Water Hammer Occurrence in Nuclear Power Plants, Technical Findings Relevant to Unresolved Safety Issue A-1*, NUREG-0927, Rev. 1, U.S. Nuclear Regulatory Commission, March 1984.

10. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, January–June 1975*, NUREG-75/090, October 1975.

11. U.S. Nuclear Regulatory Commission, *Report to Congress on Abnormal Occurrences, January–March 1976*, NUREG-0090-3, July 1976.

12. U.S. Nuclear Regulatory Commission, *Case Study Report for the Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982*, AEOD/C403, July 30, 1984.

Appendix A

EVENT SOURCES

This appendix contains detailed assessments of the operating experience data sources. The sources reviewed are listed below:

1. Licensee Event Reports;
2. Foreign Event Reports;
3. Construction Event Reports;
4. Nuclear Plant Reliability Data System;
5. In-Plant Reliability Data System;
6. system interaction methodology assessment reports;
7. system interaction analysis application reports and related material;
8. reports describing significant events;
9. IE bulletins, circulars, and information notices;
10. AEOD reports;
11. INEL special topics reports;
12. documents from the Safety Implications of Control Systems program (USI A-47); and
13. reports to Congress of abnormal occurrences.

A.1. Licensee Event Reports

A.1.1 Source

Each nuclear power plant licensed by the NRC must report certain events. These events, designated as reportable occurrences, are instances that meet the reporting requirements delineated in the *Code of Federal Regulations*, Title 10, Part 50 (10 CFR 50) (Ref. 1), in the facility's Technical Specifications, and in the facility's license provisions. The method of reporting these events, as established by the NRC, is in the form of licensee event reports. The LER reporting requirements are described in Regulatory Guide 1.16 (Ref. 2). Recent changes to the requirements are contained in NUREG-1022 (Ref. 3).

A.1.2 Contents

The LER input form has a free field for an abstract of the event plus several fields for specific codes. The abstract is a narrative description of the event and includes pertinent information such as the circumstances that led up to the event, the initiators of the event and their cause, and any occurrences (including system, component, and operator responses or failures) resulting from the initiators. The LER may also include component vendors, repair action necessary, the type of personnel involved, related IE bulletins, radiological data on releases or exposures, etc. The amount of information included in an LER may vary.

A.1.3 Availability

The NOAC at ORNL maintains two data bases for the NRC that contain LER data: (1) the LER file on the DOE/RECON* network[4] and (2) the SCSS data base.[5] The LER file, established in 1967, contains the LER abstract plus other pertinent information from the LER form (e.g., event date and unit power level). Keywords describing the event are assigned to each LER. The primary method for searching and selecting LERs is through keywords. The LER file is available through ORNL's RECON network.

Data in the SCSS file are in the form of coded sequences. The sequences contain information about the event initiators and their causes, all subsequent component and system failures, personnel errors, unit effects, and radiological releases. The data base also contains the LER abstracts. There are several methods of selecting data in the SCSS data base; however, searching the sequences for particular code combinations is the most useful and effective. The SCSS data base contains no data prior to 1981; therefore, its use is limited to post-1981 data. This data base is available through ORNL's IBM-3033 computer.

NOAC maintains hard copy files for LERs issued since 1978. The ORNL library maintains microfiche files of all docket information, including LERs.

A.1.4 Usefulness to project

LERs are the most comprehensive operating experience data base available. The LER and SCSS files provide an effective means for sorting and selecting events. In certain instances, specific event information, in addition to that provided in the data files, was needed for final screening of some events. Such information was obtained from the hard copy or microfiche files at NOAC.

A.2 Foreign Event File

A.2.1 Source

The Foreign Operating Experience Program[6] receives event reports from reactors around the world. The reports are received under bilateral agreements between the United States and foreign countries to exchange reactor operating experience information. NOAC reviews and abstracts these reports and stores the abstracts in the Foreign Event File.

A.2.2 Contents

The foreign operating experience program reviews both periodic and topical reports. Significant or potentially significant events are categorized, abstracted, and keyworded. The event descriptions generally

*The LER file is one of many data sets on the DOE/RECON data base [see *DOE/RECON User's Manual*, DOE/TIC-4586 (Ref. 4)].

provide the event date, operating mode, cause, systems and components affected, operator and system responses, and corrective actions.

A.2.3 Availability

The file allows NRC organizations to benefit from overseas operating experience and can be used to identify potential problems with U.S. reactors. However, the event descriptions are considered proprietary and, as such, cannot be released or discussed publicly. The information is available, on a restricted basis, on an ORNL computer data base.

A.2.4 Usefulness to project

The FEF was used to identify systems interaction events that occurred outside the United States. Although the events could not be discussed publicly, they alerted the staff to potential intersystem dependencies. Such information was then used in screening U.S. operating experience, particularly LERs.

A.3 Construction Event Reports

A.3.1 Source

This source includes two forms of reports, 10 CFR Part 21 notices and 10 CFR Part 50.55(e) reports.[1] Part 21 notices address component deficiencies that create, or could create, a substantial safety hazard in any facility regulated by the Atomic Energy Act. Part 50.55(e) reports involve deficiencies in design and construction that could adversely affect the safety of operations of a nuclear power plant.

These reports are available from two sources at NOAC. Prior to 1979, these reports were included in the LER file and are still available on this file. Recently, a new program was initiated for handling the construction event reports. A data base, the CER file,[7] was developed to manage the coded reports.

A.3.2 Contents

For the events contained in the LER file, the information available is similar to that described in Sect. A.1. For the events in the CER file, the basic information coded for each event includes: system, component, failure cause, manufacturer, vendor, architect-engineer, and facilities involved in the deficiency. Also coded is a description of the deficiency. References to related information and additional facts about the deficiency are provided in a text field.

A.3.3 Availability

Data are available from both of the sources discussed above. The construction deficiency events in the LER file were accessed along with LER abstracts. The CER file is maintained on ORNL computers. Hard copies of the reports are also available in NOAC files.

A.3.4 Usefulness to project

Part 21 and Part 50.55(e) notices also contain useful information for system interaction purposes. Part 21 notices tend to report primarily common-mode failures because they are component oriented. Certain reports were also pertinent. Part 50.55(e) notices were more relevant to this project because they include design deficiencies discovered during construction activities. This includes previously unrecognized inter-system dependencies.

A.4 Nuclear Plant Reliability Data System

A.4.1 Source

NPRDS,[8] developed in 1973 by the Edison Electric Institute, is presently under the direction of the Institute of Nuclear Power Operations. The NPRDS file provides generic reliability and failure data for safety-related components and systems and selected balance-of-plant components and systems. Such statistics are used in deriving reliability data of interest.

A.4.2 Contents

The NPRDS file contains the following data: general descriptive information for each reactor facility, engineering data on certain selected systems and their components, inservice data for each reactor (submitted quarterly), and descriptive data for all failures occurring in the selected systems. These selected systems include the reactor coolant systems and pressure vessels; emergency core cooling systems; decay-heat removal systems; reactor containment systems for pressure suppression, isolation, cooling, spray and hydrogen control; reactor protection systems; control rod systems; instrument systems initiating safety functions; the main steam system; and feedwater and condensate systems.

The component failure reports, which contain the most useful information, include the following data: plant identification, system or component that failed, dates and times for duration of event, a short description of failure and its cause and corrective action, component failure mode, effect of failure on system and plant, and the associated LER issued.

A.4.3 Availability

Annual reports provide reliability parameter estimates; the quarterly reports provide failure event information. Reports that identify the specific plant reporting the failure are available only to the specific plant. Certain non-plant-specific data sorts are available.

A.4.4 Usefulness to project

Because only limited generic system information is available and because the plant involved is not identified, NPRDS was not used for this project.

A.5 In-plant Reliability Data System

A.5.1 Source

Operated by ORNL, the IPRDS[9] contains data for specific equipment types collected at six reactor sites. The data were collected from 1976 through 1980.

A.5.2 Content

The IPRDS collects maintenance and repair data on four equipment types: pumps, valves, battery chargers/inverters, and diesel generators. Data analysis on this information then produces failure rates and mean-time-to-repair data.

A.5.3 Availability

Initial reports from IPRDS described the methodology and provided initial pump and motor-operated valve data. System level information is not available from IPRDS. Also, information identifying the plant where the data were collected is not released.

A.5.4 Usefulness to project

IPRDS was not particularly useful for this project because it is component oriented and system oriented data were required for this project. As a reliability data base, IPRDS focuses on providing reliability parameter estimates, rather than failure event information. Also, because of the agreements under which the data are collected, identification of the specific plants at which failures occurred is not permitted.

A.6. System Interaction Methodology Assessment/Reports

A.6.1 Source

Several studies, conducted in the last 10 years, have assessed and proposed methods for performing system interaction analyses. Most of these studies were funded by the NRC to address USI A-17. The project staff reviewed a number of reports written for these studies. Table A.1 lists these reports.

A.6.2 Contents

These reports primarily describe techniques applicable to thorough analyses of systems. The brief remarks in the table provide an overview of each report's approach to system interaction analysis.

A.6.3 Availability

These reports are available through ORNL's technical library.

A.6.4 Usefulness to project

Despite the focus of these reports on system analysis techniques, they provided excellent background material. Also, the system interaction definition and criteria used by each study were useful to this project during the development of screening criteria for event data. Several of the reports used one or more significant operating events for examples. These were also reviewed.

A.7. System Interaction Analysis Application Reports and Related Material

A.7.1 Source

Several nuclear power plants have undergone analyses for identifying possible system interactions. (A few of the studies were funded by the NRC, under the USI A-17 programs, as follow-on to the methodology assessment studies.) The project staff reviewed the results of these studies, plus several letters and related documents pertaining to system interaction analyses. Tables A.2 through A.4 list the documents reviewed.

A.7.2 Contents

The analysis reports and related documents contain a number of system interaction events identified by the studies. Of particular interest is the review of LERs done by the ACRS (NUREG-0572). One of the classes of events examined by the ACRS was "systems interactions."

Table A.1. System interaction methodology reports

Report No.	Report/Remarks
1	A. J. Buslik, I. A. Papazoglou, and R. A. Bari, *System Interactions and Common Mode Failure: Review of Methods*, BNL-NUREG-23815, Brookhaven National Laboratory, Upton, New York, January 1978. Remarks: This review addresses both qualitative and quantitative analysis methods. It concluded that system interaction analyses require modification or extension of existing methods.
2	G. J. Boyd et al., *Final Report-Phase 1 Systems Interaction Methodology Applications Program*, NUREG/CR-1321, SAND80-0384, Sandia National Laboratories, April 1980. Remarks: This report contains both methodology description and application. It uses computer-aided evaluation of safety function fault trees to identify potential system interactions. It also provides a generic analysis of the Standard Review Plan to identify weaknesses in its orientation to system interaction evaluation. The screening criteria used in the report were useful to this project.
3	A. J. Buslik, I. A. Papazoglou, and R. A. Bari, *Review and Evaluation of System Interactions Methods*, NUREG/CR-1901, BNL-NUREG-51333, Brookhaven National Laboratory, Upton, New York, January 1981. Remarks: This report, by the same authors as report 1, evaluates four approaches to system interaction analysis: failure modes and effects analysis, plant walk throughs, fault tree analysis, and event tree/fault tree analysis. It recommends a methodology using an event tree/fault tree approach supported by Failure Modes and Effects Analysis, walk throughs, and operating experience reviews. A screening criteria from the report was also useful to this project.
4	P. Cybulskis et al., *Review of Systems Interaction Methodologies*, NUREG/CR-1896, BMI-2073, Battelle Columbus Laboratories, Columbus, Ohio, January 1981. Remarks: This report reviews and compares existing analytical methods that have possible applications to system interaction analyses. It recommends a methodology comprised of two parts: (1) a qualitative part to identify and screen systems interactions candidates and (2) a quantitative part to evaluate the importance of identified system interactions. The suggested screening criteria were useful to this project.

Table A.1 (continued)

Report No.	Report/Remarks
5	J. J. Lim, T. R. Rice, R. K. McCord, and J. E. Kelly, *Systems Interaction: State-of-the-Art Review and Methods Evaluation*, NUREG/CR-1859, UCRL-53016, Lawrence Livermore Laboratory, Livermore, California, January 1981. Remarks: This review addresses both current methods for evaluating system interactions and some past analyses of system interactions. It concluded that a combination of reviews of reactor operating experience, graph-based analyses, and on site inspections can anticipate most types of system interactions. The suggested screening critieria given in the report were useful to this project, particularly the criteria for screening LERs.
6	R. Gallucci and A. Plummer, *Development and Application of a Methodology for Systems Interaction Analysis* (Abstract and Summary Paper), PNL-SA-9471, Pacific Northwest Laboratories, Richland, Washington, April 1981. Remarks: This report provides a brief presentation of a digraph-fault tree methodology for system interaction analyses. For demonstration purposes, the report applied this methodology to two reactor incidents: the Browns Ferry 3 partial failure-to-scram of June 28, 1980, and the Crystal River 3 small LOCA of February 26, 1980. The report provides background material for this project.
7	H. P. Alesso, *Some Fundamental Aspects of Fault-Tree and Digraph-Matrix Relationships for a Systems-Interactions Procedure*, UCID-19131, Lawrence Livermore National Laboratory, Livermore, California, February 28, 1982. Remarks: This report reviews some fundamental mathematical background of both fault-oriented and success-oriented risk analyses, discussing the advantages and disadvantages of each. In addition, it outlines several fault-oriented/dependency analysis approaches and several success-oriented/digraph-matrix approaches. The mathematical background information was useful to this project.
8	H. P. Alesso, I. J. Sacks, and C. F. Smith, *Initial Guidance on Digraph-Matrix Analysis for Systems Interaction Studies*, NUREG/CR-2915, UCID-19457, Lawrence Livermore Laboratory, Livermore California, March 1983. Remarks: This report contains a four-step procedure that provides guidance for digraph-matrix analysis of system interactions. The procedure may be performed independently, or it may be incorporated into a Probabilistic Risk Assessment effort. This report provided background information for this project.

Table A.2. System interaction application reports

Plant	Reports/papers currently available
Diablo Canyon	*Seismically Induced Systems Interaction Program — Completion of Containment Activities* Pacific Gas and Electric Company/Bechtel Power Corporation Undated "Diablo Canyon Seismically-Induced System Interaction Program" ANS/ENS Topical Meeting on PRA September 1981
Indian Point 3	*Review of the PASNY Systems Interaction Study* Lawrence Livermore National Laboratory (UCID-19130) March 1, 1982 Letter from New York Power Authority to NRC Division of Licensing "Authority Review/Evaluation of Ebasco Findings" November 30, 1983 "Systems Interaction Program for the Indian Point 3 Nuclear Power Plant" ANS Winter Meeting November 1982 *Preliminary Investigation of Interconnected Systems Interactions for the Safety Injection System of Indian Point 3* Lawrence Livermore National Laboratory (UCID-19473) March 4, 1983
Watts Bar	*Final Report — Phase I Systems Interaction Methodology Applications Program* Sandia National Laboratories (NUREG/CR-1321) April 1980 *Preliminary Systems-Interaction Results From the Diagraph Matrix Analysis of the Watts Bar Nuclear Power Plant Safety-Injection Systems* Lawrence Livermore Laboratory (UCID-19707) June 1983

Table A.2 (continued)

Plant	Reports/papers currently available
Watts Bar (cont.)	*Systems Interaction Results from the Diagraph Matrix Analysis of the Watts Bar Nuclear Power Plant High Pressure Safety Injection Systems — Volume I* Lawrence Livermore Laboratory (UCID-19707) June 1983
	Systems Interaction Results from the Digraph Matrix Analysis of the Watts Bar Nuclear Power Plant High Pressure Safety Injection Systems — Volume I Lawrence Livermore National Laboratory (UCRL-preprint) July 1983
Zion	*Commonwealth Edison Company Zion Station Systems Interaction Study* Fluor Pioneer, Inc. June 16, 1978
	"Review of Zion Station for Potential Systems Interaction Events" ANS Winter Meeting November 1978
Grand Gulf	*Safety Evaluation Report Related to the Operation of Grand Gulf Nuclear Generating Station* U.S. Nuclear Regulatory Commission (NUREG-0831) September 1981
San Onofre	*Safety Evaluation Report Related to the Operation of San Onofre Nuclear Generating Station, Units 2 and 3* U.S. Nuclear Regulatory Commission (NUREG-0712, Supplement 2) May 1981

Table A.3. ACRS system interaction material

Item/subject/date
Letter from Chairman, ACRS, to Director of Regulation, NRC Subject: Systems Analysis of Engineered Safety Systems November 8, 1974
Letter from Chairman, ACRS, to Executive Director for Operations, NRC Subject: Review of Systems Interaction June 17, 1977
Letter from Executive Director, ACRS, to Acting Director, Office of Nuclear Reactor Regulation, NRC Subject: Correspondence Regarding Systems Interaction Analysis June 28, 1977
Letter from Assistant Director for Operating Reactors, NRC, to Chairman, ACRS Subject: Zion System Interaction Analysis October 21, 1977
Advisory Committee on Reactor Safeguards Report (NUREG-0572) *Review of Licensee Event Reports (1976–1978)* September 1979
Letter from Chairman, ACRS, to Executive Director for Operations, NRC Subject: Systems Interactions Study for Indian Point Nuclear Generating Unit No. 3 October 12, 1979
Minutes of the ACRS Plant Arrangements Subcommittee Meeting Subject: Draft Report on the Systems Interaction Methodology Application Program (Sandia Study) February 20, 1980
Memorandum for ACRS Members from R. Savio, Senior Staff Engineer Subject: Possible System Interaction Study Topics March 3, 1982

Table A.4. Licensing correspondence addressing systems interaction

Plant	Correspondence
San Onofre 2 and 3	Letter from Southern California Edison Company to Director, Office of Nuclear Reactor Regulation, NRC Subject: Response to NRC Systems Interaction Branch Question 510.1 March 9, 1981
Midland I and II	Letter from Consumers Power Company to Director, Office of Nuclear Reactor Regulations, NRC Subject: Systems Interaction Program for Midland Units I and II January 28, 1983
LaSalle 1 and 2	Letter from Division of Systems Integration to Division of Licensing Subject: Supplemental Safety Evaluation Report Input for Chapter 7 Regarding Control Systems Failure December 7, 1983

A.7.3 Availability

These documents were provided by the NRC Generic Issues Branch.

A.7.4 Usefulness to project

These documents provide some event data but were of more use during the development of screening criteria for data selection. Events meeting the criteria for this project (Sects. 1.3 and 3.2) were included in the final results.

A.8 Reports Describing Significant Events

A.8.1 Source

The project staff reviewed several reports from programs using or evaluating operating experience data. These programs focused on a number of areas of interest: potential severe core damage accidents, pressure vessel thermal shock, unplanned boron dilution, and station blackout accidents. Tables A.5 and A.6 list the reports that were reviewed.

Table A.5. NSIC reports of interest

Report No.	Title
NUREG/CR-2497	*Precursors to Potential Severe Core Damage Accidents: 1969–1979 A Status Report*, June 1982
NUREG/CR-0566	*Common-Mode/Common-Cause Failure: A Review and a Bibliography*, May 1979
NUREG/CR-0848	*Operating Experience with Valves in Light-Water-Reactor Nuclear Power Plants for the Period 1965–1978*, July 1979
ORNL/NSIC-176	*Descriptions of Selected Accidents that Have Occurred at Nuclear Reactor Facilities*, April 1980
NUREG/CR-2789	*Pressure Vessel Thermal Shock at U.S. Pressurized-Water Reactors: Events and Precursors, 1963–1981* April 1983
NUREG/CR-2797	*Evaluation of Events Involving Service Water Systems in Nuclear Power Plants*, November 1982
NUREG/CR-2798	*Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants*, July 1982
NUREG/CR-2799	*Evaluation of Events Involving Decay Heat Removal Systems in Nuclear Power Plants*, July 1982
NUREG/CR-3122	*Potentially Damaging Failure Modes of High- and Medium-Voltage Electrical Equipment*, August 1983

Table A.6. Miscellaneous reports

Report No.	Title
NUREG/CR-1722	*Interim Report on Systematic Errors in Nuclear Power Plants* Lawrence Livermore National Laboratory October 1980
NUREG-0305	*Technical Report on D.C. Power Supplies in Nuclear Power Plants* Office of Nuclear Reactor Regulation, NRC July 1977
NUREG-0886	*A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants* Office of Nuclear Regulatory Research, NRC April 1981
NUREG/CR-3226	*Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)* Sandia National Laboratory May 1983

A.8.2 Contents

These reports primarily contain data for significant operating events. The studies focus on areas of concern and generally involve safety and safety-related equipment.

A.8.3 Availability

These reports are available through the NSIC files and the ORNL technical library.

A.8.4 Usefulness to project

These reports contain useful event data. The ASP reports were of particular use because that program selected events where multiple safety functions were degraded. For the events selected by the ASP Program, both the LER abstracts (from the RECON LER file) and the ASP reports for the final evaluation were used.

A.9. IE Bulletin, Circulars, and Information Notices

A.9.1 Source

IE bulletins, circulars, and information notices are issued by the NRC to licensees and construction permit holders informing them of events that may have generic implications.

Each issuance is based on events reported by licensees, NRC inspectors, agreement states, or others where a preliminary evaluation indicates that the event may affect other licensees.

A.9.2 Content

IE bulletins provide information about one or more similar events and require that licensees take specific actions. The licensee reports actions taken or to be taken and provides information the NRC may need to assess the need for further action. Prompt response by licensees is required and failure to respond will normally result in NRC enforcement action.

IE circulars are used when the implication of one or more similar events indicate that both licensee notification and specific licensee action is recommended. Circulars do not require that licensees submit a reply to the NRC describing their actions. Licensees review the information and implement the recommendations if they are applicable. The use of circulars was discontinued after 1981.

IE information notices provide information but do not require specific actions; they are rapid transmittals of information which may not yet have been completely analyzed by the NRC, but of which licensees should be aware. Licensees receiving an information notice are expected

to review the information for applicability to their current and future licensed operations. If the information does apply, licensees are expected to take action necessary to avoid repetition of the problem.

A.9.3 Availability

All issued IE bulletins, circulars, and information notices are available in hard copy or microfiche at NOAC.

A.9.4 Usefulness to project

The IE bulletins, circulars, and information notices were a source of operating event data (some events were postulated). The events described in these documents were screened for possible system interactions. For those events reported in these documents that were also reported as LERs, the abstracts (retrieved from the RECON LER file) were also used to provide more in-depth details about the events.

A.10. AEOD Reports

A.10.1 Sources

AEOD conducted numerous case studies and engineering evaluations covering operating situations of interest to the NRC. The project reviewed all of the AEOD reports available during the event selection task.

A.10.2 Contents

AEOD reports contain a detailed description of specific operational events. In addition, an explanation of the actions taken by the reactor operator and the NRC (when appropriate) is included. The reports address the effort of the analysis to determine the "root" cause.

A.10.3 Availability

The reports, issued since the 1980 establishment of AEOD, are available in NOAC files.

A.10.4 Usefulness to project

Many of the reports relate to actual or potential systems interaction events. Most reports include listings of related LERs found during the analysis. These reports were extremely useful to this project, and a number of events from them were selected as adverse system interactions. Appendix D lists the AEOD reports that are used as a reference for events selected as ASI events.

A.11. Idaho National Engineering Laboratory Special Topic Reports

A.11.1 Source

These reports form part of the Selected Operating Reactor Issues Program being conducted for the NRC Office of Nuclear Reactor Regulation by INEL Reliability and Statistics Branch.

A.11.2 Content

Numerous reports are included in this series. Topics include degraded grid protection for Class 1E power systems, adequacy of station electric distribution system voltages, technical specifications for redundant decay heat removal capability, audit of the environmental qualification of safety-related electrical equipment, testing of reactor trip system and engineered safety features, and electrical penetrations of reactor containment. Each topic was evaluated for several plants.

A.11.3 Availability

The reports are available in hard copy from the NOAC files.

A.11.4 Usefulness to project

No system interaction events were specifically identified in these reports. These reports were not intended to relate information on system interaction problem areas and as such were not of direct use to this project.

A.12. Safety Implications of Control Systems (USI A-47)

A.12.1 Source

The objective of USI A-47 is to assess the safety implications of control systems by examining two areas: (1) the effects of control system malfunctions on plant dynamic behavior and (2) the interactions of these malfunctioning controls with other plant systems.

A.12.2 Content

The safety implications of nuclear power plant control system failures and action, both planned and unplanned, are being examined. Current efforts include systems analyses of both PWR and BWR control systems. These analyses address reactor transients resulting from control system malfunctions. The work focuses on steam generator/reactor vessel overfill transients, reactor overcooling transients, loss of control system

power supplies, and other non-safety-grade equipment failures with safety implications.

A.12.3 Availability

Because the USI A-47 efforts are currently under way, no final reports are available at this time.

A.12.4 Usefulness to project

The work being performed on this task is of interest to this project; however, no final reports are available at this time.

A.13 Reports to Congress of Abnormal Occurrences

A.13.1 Source

The NRC reports to the Congress each quarter any abnormal occurrences involving facilities or activities regulated by the NRC. An abnormal occurrence is defined as an unscheduled incident or event that the NRC determines is significant from the standpoint of public health or safety.

A.13.2 Content

For each event reported as an abnormal occurrence, the information contained in the report to Congress includes date and place, nature and probable consequences, causes, and licensee and regulatory actions taken to prevent recurrence.

A.13.3 Availability

Copies of all the AO reports are available at NOAC. This series of reports has existed since 1975 and is currently issued quarterly.

A.13.4 Usefulness to project

The AO reports include the most significant events that occurred during the quarter. Descriptions of the events selected (by the staff) as possible system interactions were reviewed. Because almost all of the events reported as abnormal occurrences were also reported in LERs, both the AO reports and the LER abstracts (retrieved from the RECON LER file) were reviewed.

References for Appendix A

1. *Code of Federal Regulations*, Title 10, Part 21 "Reporting of Defects and Noncompliance" and Part 50 "Domestic Licensing of Production and Utilization Facilities."

2. Nuclear Regulatory Commission, Regulatory Guide 1.16, *Reporting of Operating Information, Appendix A: Technical Specifications*, Rev. 4, August 1975.

3. U.S. Nuclear Regulatory Commission, *Licensee Event Report System*, NUREG-1022, September 1983.

4. U.S. Department of Energy, *DOE/RECON User's Manual*, DOE-TIC-4586, Rev. 1, May 1981.

5. *Sequence Coding and Search System Coder's Manual for Licensee Event Reports*, ORNL/NSIC-189, Rev. 0, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., March 1984.

6. U.S. Nuclear Regulatory Commission, *Foreign Event File (FEF) Data Base Description*, March 1984.

7. *CERCRS Data Base Development Project Final Report and User's Handbook*, ORNL/NSIC-221, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., January 1984.

8. Institute of Nuclear Power Operations, *Nuclear Plant Reliability Data System*, Atlanta, Ga., 1983.

9. *The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report*, ORNL/TM-8271, NUREG/CR-2641, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., July 1982.

Appendix B

EVENT ATTRIBUTE DEFINITIONS

B.1 Introduction

Appendix B defines the event attributes that make up the event listings. The event listing provides a brief description of the system interaction event and includes pertinent information about the event such as systems and components involved, the undesirable result, the unanticipated dependency, and corrective actions. The attributes shown in the example event listing, Exhibit B.1, are described in the following sections. (Appendix C contains event listings, printed in the same format as Exhibit B.1, for the ASI events selected by this project.)

B.2 Event Attributes

B.2.1 Plant

This attribute contains the name of the plant at which the event occurred. Table B.1 lists the nuclear power plants by name and gives the docket number, reactor type, NSSS vendor, and architectural engineering firm (only for those plants having ASI events included in this report).

B.2.2 Plant type

This attribute lists the name of the NSSS vendor (Babcock & Wilcox, Combustion Engineering, General Atomic, General Electric, or Westinghouse) and the reactor type (boiling water, pressurized water, or high temperature gas-cooled) for the plant of interest (see Table B.1).

B.2.3 Event date

This attribute identifies (1) the date on which the event occurred or (2) the date on which a postulated event was discovered. If the date was not known, then the date of the reference document was used.

B.2.4 Experience

This attribute indicates whether an event actually occurred or was identified as having the potential to occur. Events were assigned as ACTUAL or POTENTIAL.

Plant: St. Lucie 1 Plant type: CE PWR

Event date: 3/31/1978 Experience: Potential

Operating status: Construction

Initiating system and component

Medium voltage AC (35 kV to 600 V)
Electrical/I&C function items

Systems/components between which the dependency occurred

Medium voltage AC (35 kV to 600 V)
Electrical conductors

Emergency power generation
Electrical conductors

Safety systems/components affected

Emergency power generation
Subsystem occurrence

Type of coupling: Functional

Result type: 2 Discovery: AE/vendor notification

Initiating event: Seismic event causes failure of non-Class 1E transformer disconnect contacts

Propagation: Contacts could dislodge causing short circuit on bus, defeating emergency power

Dependency: Normal and emergency power share bus with non-Class 1E contacts

Undesirable result: Nonsafety system can cause loss of safety bus during seismic event

Remarks: Same design used at Unit 2

Corrective action: Design change/modification

References: L0098 Event No. 187

Exhibit B.1. Example event listing.

Table B.1. Nuclear power plant facilities sorted by facility name

Facility name	Docket No.	Reactor type	NSSS vendor	AE[a]
Arkansas Nuclear 1	313	PWR	B&W	Bech
Arkansas Nuclear 2	368	PWR	CE	Bech
Arnold	331	BWR	GE	Bech
Beaver Valley 1	334	PWR	WES	S&W
Beaver Valley 2	412	PWR	WES	S&W
Bellefonte 1	438	PWR	B&W	TVA
Bellefonte 2	439	PWR	B&W	TVA
Big Rock Point	155	BWR	GE	Bech
Browns Ferry 1	259	BWR	GE	TVA
Browns Ferry 2	260	BWR	GE	TVA
Browns Ferry 3	296	BWR	GE	TVA
Brunswick 1	325	BWR	GE	UE&C
Brunswick 2	324	BWR	GE	UE&C
Calvert Cliffs 1	317	PWR	CE	Bech
Calvert Cliffs 2	318	PWR	CE	Bech
Clinton 1	461	BWR	GE	S&L
Connecticut Yankee	213	PWR	WES	S&W
Cook 2	316	PWR	WES	AEPSC
Cooper	298	BWR	GE	Burns/Roe
Crystal River 3	302	PWR	B&W	Gil
Davis-Besse 1	346	PWR	B&W	Bech
Diablo Canyon 1	275	PWR	WES	PG&E
Diablo Canyon 2	323	PWR	WES	PG&E
Dresden 2	237	BWR	GE	S&L
Dresden 3	249	BWR	GE	S&L
Farley 1	348	PWR	WES	Bech & SCSI
Farley 2	364	PWR	WES	Bech & SCSI
Fermi 2	341	BWR	GE	S&L
Ft. Calhoun 1	285	PWR	CE	G&H
Ft. St. Vrain	267	HTGR	GA	S&L
Ginna	244	PWR	WES	Gil
Grand Gulf 1	416	BWR	GE	Bech
Hatch 1	321	BWR	GE	SSI & Bech
Hatch 2	366	BWR	GE	SSI
Indian Point 2	247	PWR	WES	UE&C
Indian Point 3	286	PWR	WES	UE&C
Kewaunee	305	PWR	WES	FPS
Maine Yankee	309	PWR	CE	S&W
McGuire 1	369	PWR	WES	Duke
Midland 1	329	PWR	B&W	Bech
Midland 2	330	PWR	B&W	Bech
Millstone 1	245	BWR	GE	Ebasco
Millstone 2	336	PWR	CE	Bech
Monticello	263	BWR	GE	Bech
Nine Mile Point 1	220	BWR	GE	NM
North Anna 1	338	PWR	WES	S&W
North Anna 2	339	PWR	WES	S&W
North Anna 3	404	PWR	B&W	S&W

Table B.1 (continued)

Facility name	Docket No.	Reactor type	NSSS vendor	AE[a]
Oconee 3	287	PWR	B&W	Duke/Bech
Oyster Creek	219	BWR	GE	Burns/Roe
Palisades	255	PWR	CE	Bech
Peach Bottom 2	277	BWR	GE	Bech
Pilgrim 1	293	BWR	GE	Bech
Point Beach 1	266	PWR	WES	Bech
Point Beach 2	301	PWR	WES	Bech
Prairie Island 1	282	PWR	WES	FPS
Quad Cities 1	254	BWR	GE	S&L
Quad Cities 2	265	BWR	GE	S&L
Rancho Seco	312	PWR	B&W	Bech
Robinson 2	261	PWR	WES	Ebasco
Salem 1	272	PWR	WES	PSE&G
San Onofre 1	206	PWR	WES	Bech
San Onofre 2	361	PWR	CE	Bech
San Onofre 3	362	PWR	CE	Bech
Sequoyah 1	327	PWR	WES	TVA
Sequoyah 2	328	PWR	WES	TVA
St. Lucie 1	335	PWR	CE	Ebasco
St. Lucie 2	389	PWR	CE	Ebasco
Surry 1	280	PWR	WES	S&W
Surry 2	281	PWR	WES	S&W
Susquehanna 1	387	BWR	GE	Bech
TMI-1	289	PWR	B&W	Gil
TMI-2	320	PWR	B&W	Burns/Roe
Trojan	344	PWR	WES	Bech
Turkey Point 3	250	PWR	WES	Bech
Turkey Point 4	251	PWR	WES	Bech
Watts Bar 1	390	PWR	WES	TVA
Watts Bar 2	391	PWR	WES	TVA
WNP 1	460	PWR	B&W	UE&C
WNP 4	513	PWR	B&W	UE&C
Zion 1	295	PWR	WES	S&L
Zion 2	304	PWR	WES	S&L

[a] AEPSC American Electric Power Service Corporation
BECH Bechtel Corporation
S&W Stone and Webster
TVA Tennessee Valley Authority
UE&C United Engineers and Constructors, Inc.
S&L Sargent and Lundy
GIL Gilbert Associates Inc.
PG&E Pacific Gas & Electric
SCSI Southern Company Services, Inc.
G&H Gibbs and Hill
SSI Southern Services Inc.
DUKE Duke Power Co.
PSE&G Public Services Electric & Gas (New Jersey)
FPS Fluor Power Services
NM Niagara Mohawk Power Corp.

B.2.5 Operating status

This attribute identifies the mode at which the plant was operating when the event occurred. For a postulated event, the mode inferred or identified in the reference document is used. Table B.2 lists the operating modes used.

Table B.2. Operating modes

Description
Construction
Cold shutdown
Hot shutdown
Hot standby
Load change during routine power operation
Preoperational/startup/power ascension tests
Refueling
Routine shutdown
Routine startup
Steady state operation
Unknown/not applicable
Other

B.2.6 Initiating system and component

A system interaction is characterized by an initiating event that can be the failure, action, or inaction of a system, train, component, or structure (see definition in Sect. 1.3). This attribute identifies the system in which the initiating event occurred. Table B.3 contains the system designations used for this project. The system designations were taken directly from the SCSS program. In addition, six system designations were added (indicated by an asterisk in Table B.3). These systems were used when (1) the actual systems affected were not known or (2) multiple systems were affected. (The *SCSS Coder's Manual* provides descriptions of each of these systems.)

This attribute also identifies the component that initiated a system interaction. The initiating component is part of the initiating system. For system interaction events that began with an operator error, personnel were used as the initiating component. Table B.4 lists the component designations and typical components included in each component designation. These component designations were taken directly from the SCSS program.

Table B.3. System designations

Description[a]
A. PRIMARY REACTOR SYSTEMS
Reactor core
Control rod drive (PWR)
Control rod drive (BWR)
Reactor vessel
Primary coolant (PWR)
Pressurizer (PWR)
Steam generator (PWR)
Recirculating water (BWR)
B. ESSENTIAL REACTOR AUXILIARY SYSTEMS
Auxiliary feedwater (PWR)
Isolation condenser (BWR)
Reactor core isolation cooling (BWR)
Residual heat removal (PWR)
Residual heat removal (BWR)
Low-pressure coolant injection (BWR)
CVCS/high-pressure safety injection (PWR)
Intermediate pressure injection (PWR)
High-pressure coolant injection (BWR)
Steam generator pressure relief (PWR)
Reactor overpressure protection (BWR)
Core flooding accumulator (PWR)
Upper head injection (PWR)
High-pressure core spray (BWR)
Low-pressure core spray (BWR)
Multiple safety systems*[b]
All ECCS systems*
Multiple ECCS systems*
All ESF systems*
Multiple ECCS systems*
All systems requiring emergency power*
C. ESSENTIAL SERVICE SYSTEMS
Component cooling water
Essential raw cooling/service
Essential compressed air
Borated/refueling water storage (PWR)
Condensate storage
Emergency generator fuel
Emergency generator cooling
D. ESSENTIAL AUXILIARY SYSTEMS
Fuel pool cooling and cleanup
Containment isolation
Containment spray
Containment pressure suppression makeup (BWR)
Containment combustive gas control
Containment ice condenser (PWR)

Table B.3 (continued)

Description[a]
E. ELECTRICAL SYSTEMS
High voltage ac (greater than 35 kV)
Medium voltage ac (35 kV to 600 V)
Low voltage ac (less than 600 V)
Vital instrument, control, and computer ac dc power
Electrical heat tracing
Emergency power generation
Conduit and cable tray
F. FEEDWATER, STEAM, AND POWER CONVERSION SYSTEMS
Main steam
Turbine generator
Main condenser
Condensate and feedwater
Circulating water (open cycle)
Seal water
G. HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS
Reactor building HVAC (PWR)
Reactor building HVAC (BWR)
Primary containment vacuum relief
Sec containment recirc and exhaust
Dry-well/torus HVAC and purge (BWR)
Reactor auxiliary building HVAC
Control building HVAC
Fuel building HVAC
H. INSTRUMENTATION AND CONTROLS SYSTEMS
Control room panels
Fire detection
Emergency generator instrumentation and controls
Turbine generator instrumentation and control
Plant monitoring
Leak monitoring
Radiation monitoring
Reactor power control (PWR)
Feedwater control
Reactor protection
Engineered safety features actuation
Nonnuclear instrumentation
I. SERVICE AUXILIARY SYSTEMS
Auxiliary steam
Sampling
Control and service air
Demineralized water
Material and equipment handling
Fire protection
Compressed gas
Potable and sanitary water
Insulating oil

Table B.3 (continued)

Description[a]
I. SERVICE AUXILIARY SYSTEMS (continued)
Fuel storage
Steam generator startup
Lube oil
Boron recovery
Control rod drive cooling water
Raw cooling water
Raw service water
Chemical additive injection
J. WASTE MANAGEMENT SYSTEMS
Liquid radwaste
Solid radwaste
Gaseous radwaste (PWR)
Gaseous radwaste (BWR)
Nonradioactive waste (liquid, solid, and gaseous)
Steam generator blowdown (PWR)
Cooling tower blowdown
Plant drainage
Equipment drainage (including vents)
Roof drainage
Suppression pool cleanup (BWR)
Reactor water cleanup (BWR)
Initial unit conditions/unit effects
Effect on environment/personnel
Other
Multiple known
Unknown
K. STRUCTURAL SYSTEMS
Control building
Emergency generator building
Environment (external to any structure)
Fuel building
Miscellaneous/unknown structures
Primary reactor containment (PWR)
Reactor auxiliary building
Reactor dry well (BWR)
Reactor torus/suppression pool (BWR)
Secondary reactor containment (BWR)
Secondary reactor containment (PWR)
Turbine building
L. PERSONNEL ACTIVITIES
Construction activity
Operation activity

[a]NOTE: "(BWR)" and "(PWR)" denote systems applicable only to that reactor type.

[b]Entries followed by asterisks are designations that indicate when the actual systems affected were not known or when multiple systems were affected.

Table B.4. Component designations

Description
1. Accumulators/Reservoirs
Includes: Accumulator Gas bottles and manifold Reservoir Tank
2. Air Dryers
Includes: Air dryer, absorption/adsorption Dryer
3. Annunciators
Includes: All audio/visual annunciators and alarms
4. Batteries/Chargers
Includes: All batteries and battery chargers
5. Blowers/Compressors
Includes: Compressor Eductor Ejector Fan/blower Turbocharger Ventilator
6. Chemical Function Items
Includes: Chemical addition injector Demineralizer
7. Cleaning Equipment
Includes: All cleaning equipment
8. Communications Equipment
Includes: Intercom Phones
9. Control Rods
Includes: All control rods both full and partial length
10. Control Rod Drives
Includes: All control rod drives including hydraulic units
11. Electrical Conductors
Includes: Bus Cable/wire Transmission line

Table B.4 (continued)

Description
12. Electrical/I&C Function Items
Includes: Card, circuit Cathode ray tube Coil Conduit Contactor/contacts Interlock Monitor Monitor, atmospheric condition Monitor, mechanical condition Oscillator Potential device Power supply, electric Power supply, uninterruptible Rectifier Solenoid Surge protection package Synchroscope Telemeter Tray, cable Typewriter/printer/plotter
13. Engines, Internal Combustion
Includes: All engines including diesel generator engine
14. Equipment Interface Items
Includes: Board/panel Box, junction Box, other type Connector Console Control station Control unit, remote Rack/cabinet Station, sample Terminal block
15. Filters, Non-I&C
Includes: Filter (process) Screen Separator Strainer
16. Fuel Elements
Includes: All reactor core fuel elements

Table B.4 (continued)

Description
17. Generators
Includes: Converter Generator[a] Generator, motor Inverter
18. Handling Equipment
Includes: Crane Fuel handling equipment Handling equipment, miscellaneous
19. Electric Heaters
Includes: Heater, electric Heat tracing
20. Heat Exchangers
Includes: Air handling/conditioning unit (heating and ventilation) Boiler Coil, cooling Coil, heating Condenser Condenser, ice Cooler Cooling tower Fan cooler unit Heater, other type Heat exchanger Steam generator
21. I&C General
Includes: Capacitor Diode Resistor
22. I&C/Circuit Breakers
Includes: Circuit breaker, ac Circuit breaker, dc Fuse
23. I&C/Computational Modules
Includes: Amplifier Averager Computer Differentiator Integrator Modifier Summer Totalizer/integrator

Table B.4 (continued)

	Description
24.	I&C/Controllers
	Includes: All controllers including speed, frequency, power, level, temperature, voltage, pressure, position, etc.
25.	I&C/Filters and Isolators
	Includes: Filter (I&C) Isolator/buffer Lightning arrestor Transducer
26.	I&C/Indicators
	Includes: All indicators including speed, frequency, power, voltage, current, temperature, level, pressure, flow, position, etc.
27.	I&C/Recorders
	Includes: All recorders including data loggers
28.	I&C/Relays
	Includes: All relays
29.	I&C/Sensors
	Includes: All primary sensors/detectors/monitors including fire/smoke, voltage, power, radiation, flux/neutron, temperature, pressure, flow level, position, etc.
30.	I&C/Switches
	Includes: All switches including bistables
31.	I&C/Transmitters
	Includes: All transmitters
32.	Lighting Equipment
	Includes: All lighting equipment excluding indicator lamps or bulbs
33.	Mechanical Function Items
	Includes: Basket, ice condenser Bearing/bushing Belt Brake Clutch Coil, drain Collector Coupling Diaphragm Duct

Table B.4 (continued)

	Description
33.	Mechanical Function Items (continued)
	Includes: Fastener Gear Governor Hose Hydrant Insulation Sample Seal Shaft/stem Valve seat
34.	Motors
	Includes: Exciter Motor Motor starter
35.	Penetrations
	Includes: All pentrations including personnel penetrations
36.	Personnel
	Includes: All utility or contractor personnel
37.	Pipes and Fittings
	Includes: Nozzle Pipe Plug Rupture disk Sensing line Sleeve Tubing Well, special process monitor
38.	Pumps
	Includes: All pumps including jet pumps
39.	Recombiners
	Includes: All recombiners
40.	Shock Suppressors and Supports
	Includes: Anchor Hanger Snubber Support

Table B.4 (continued)

	Description
41.	Structural Function Items
	Includes: Access platform/stair/ladder Concrete structure/shield Door/cover/hatch Drain Discharge flume Elevator Flame arrestor/fire barrier Miscellaneous structural features Pit Pool Prestressed concrete/tendon and anchorage Structural framing and foundation Sump Wall/bulkhead
42.	Transformers
	Includes: All transformers
43.	Turbines
	Includes: All turbines
44.	Valves
	Includes: All valves including vacuum breakers and dampers
45.	Valve Operators
	Includes: All valve operators
46.	Vessels
	Includes: Pressurizer Vessel, reactor Vessel
47.	Miscellaneous
	Includes: All miscellaneous or unknown components
48.	Total System Occurrence
	Use when total system is inoperable
49.	Subsystem Occurrences
	Use when one or more trains/channels are inoperable but the total system is not

[a]This includes the generator on the diesel generator.

B.2.7 Systems/components between which the dependency occurred

A system interaction event is characterized by the inconspicuous or unanticipated dependency between two or more systems, trains, components, or structures. This attribute identifies two system/component pairs between which the dependency occurred. The system and component designations are those listed in Tables B.3 and B.4, respectively.

B.2.8 Safety system/components affected

To be classified as an adverse system interaction event, the event must degrade or have potential to degrade one or more safety systems. This attribute identifies the safety system(s) and component(s) affected. Up to three system/component pairs may be listed. (Tables B.3 and B.4 list the system and component designations used.)

B.2.9 Type of coupling

This field is used to group system interaction events into three categories based on the reason for the dependency. All events were designated as FUNCTIONAL, SPATIAL, or HUMAN.

B.2.10 Plant area

For spatial system interactions only, this attribute identifies the physical location where the system interaction event took place. Item K in Table B.3 lists the plant area designations. The plant area designations were taken from the SCSS program.

B.2.11 Result type

This attribute describes the degraded level of safety that occurs as a result of the dependent failures. Table B.5 defines the six undesirable result types.

B.2.12 Discovery

This attribute describes the method of discovery for the event. Table B.6 lists the methods of discovery used.

B.2.13 Initiating event

This short text describes the initiating event and the resulting actions, inactions, or failures leading to the unanticipated dependency. (This field and the next four fields each have a maximum length of 80 characters.)

Table B.5. Result types

Type	Description[a]
0	No degradation of a safety system.
1	Degradation of redundant portions of a safety system, including consideration of all auxiliary support functions. Redundant portions are those considered to be independent in the design and analysis of the plant. This also includes redundant portions of two safety systems that can accomplish the same safety function.
2	Degradation of a safety system by a nonsafety system.
3	Initiation of an "accident" (e.g., LOCA, MSLB) *and* (a) the degradation of *at least one* redundant portion of any one of the safety systems required to mitigate that event; or (b) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action.
4	Initiation of a "transient" (including reactor trip), *and* (a) the degradation of *at least one* redundant portion of any one of the safety systems required to mitigate the event; or (b) degradation of critical operator information sufficient to cause him to perform unanalyzed, unassumed, or incorrect action.
5	Initiation of an event that (a) requires actions of the plant operators in areas outside the control room area *and* (b) disruption of the access to these areas.

[a]Note: In some cases, combinations of undesirable results occurred. For example, failure of a non-safety-related system that caused a transient and degraded a safety system would be a result type 2 and type 4 event (recorded as 2, 4).

Table B.6. Methods of discovery

Description
AE/vendor notification
Audio/visual alarm
Design calculation/verification
Installation
Maintenance/modification
NRC notification
Operational abnormality
Other
Review of procedure/test result
Routine test/inspection
Special test/inspection
Unknown

B.2.14 Propagation

This short text describes how the failures, actions, or inactions propagated.

B.2.15 Dependency

This short text describes the unanticipated dependency that exists.

B.2.16 Undesirable result

This text describes the undesirable result (i.e., safety system degradation) that resulted from the event. Any undesirable failures, actions, or inactions that resulted may also be described, if unusual or significant in nature.

B.2.17 Remarks

This field describes any additional event information needed for clarity.

B.2.18 Corrective action

This attribute describes the action taken by the utility to correct the *dependencies* and prevent their reoccurrence. Corrective actions used in this study include: design change/modification, administrative/procedural change, repair/replacement, and other.

B.2.19 Category

This field contains the category number of each event as described in Chap. 4.

B.2.20 References

This attribute lists the references for the system interaction event. Appendix D contains the reference information for the selected events. Up to five references per event were allowed. References start with a letter that indicates document type. Table B.7 shows the reference codes used by the project.

B.2.21 Event number

The event number is an identification number for each event in the project's computer data base.

Table B.7. Reference codes

Reference codes	Type
A00001 to A99999	AEOD reports
C00001 to C99999	CER
E00001 to E99999	SEP reports
I00001 to I99999	IE bulletins/notices
L0000I to L99999	LERs
M0000I to M99999	Miscellaneous documents
S00001 to S99999	SI reports
X0000I to X99999	ACRS documents

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13. ABSTRACT (200 words or less)

This report describes the first phase of an NRC-sponsored project that identified and evaluated system interaction (SI) events that have occurred at commercial nuclear power plants in the United States. The project included: an assessment of nuclear power plant operating experience data sources; the development of search methods and event selection criteria for identifying SI events; review of possible SI events; and final evaluation and categorization of events. The report outlines each of these steps and presents the results of the project. The results include 235 events identified as adverse system interactions and 23 categories into which those events were assigned. The categories represent groups of similar events and include areas such as: adverse interactions between normal or offsite power and emergency power systems; degradation of safety systems by vapor or gas intrusion; degradation of safety-related equipment by fire protection systems; and flooding of safety-related equipment through plant drain systems. After evaluating each category (and the events contained in them), the project made two major recommendations: the safety significance of each category with emphasis on the potential for continued problems in these areas should be examined; and current system interaction analyses methods should be studied to determine their effectiveness for identifying system interaction events. (Phase II of this project, "Evaluation of System Interaction Methods," will assess the effectiveness of current methods using the events identified in this report).

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