
Annotated Bibliography of Reliability and Risk Data Sources

Prepared by Oren V. Hester, Sharon R. Brown, Cynthia D. Gentillon

EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory
Commission

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Manuscript Completed: September 1987
Date Published: March 1988

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NRC FIN A6237

ABSTRACT

This document is an annotated bibliography of nuclear, nonnuclear, and foreign data sources that are useful in nuclear power plant reliability and risk analysis applications. A brief description of the contents, areas of usefulness, access information, and the name and address of a contact is provided for data sources of all types. In addition, for nuclear data sources, tabular comparisons are made. These comparisons include the scope of the data sources; their operational, special-purpose operational, pedigree, aggregated, and derived data; the operational and design data each data source originates from; and access information.

Probabilistic risk assessments (PRAs) are profiled separately. For each PRA, background information that describes the PRA and the plant itself is provided. Also the input data sources used to support each PRA are identified. Of special interest is the identification of unique plant-specific data sources that have evolved from the PRA analyses. To the extent possible, how the data sources were used in the analysis is also discussed.

SUMMARY

The United States Nuclear Regulatory Commission's Office of Analysis and Evaluation of Operational Data (USNRC-AEOD) established two primary objectives for FY 1986 for its Integrated Risk Assessment Data Acquisition Program (IRADAP). The objectives were to a) identify and prioritize the operational data needs of the USNRC's reliability and risk programs, and b) to document and profile existing data sources which can be used as direct input to or provide support to USNRC programs and activities.

For IRADAP's second objective, EG&G Idaho (a prime operating contractor at the Department of Energy's Idaho National Engineering Laboratory) developed an annotated bibliography of data sources for use in reliability and risk analysis. The data sources include both computerized data bases and hard copy reports.

Most input for this document originated from three sources. Two were previous reports that identified a number of data sources: Nuclear Regulatory Commission, Integrated Risk Assessment Data Acquisition Program Status Report to the Office of the Executive Director, October 4, 1985; and C. Kido et al., A Bibliography of Data Bases for Nuclear Power Plant Risk Assessment, EGG-EA-6100, November 1982. The third source was responses to a questionnaire sent to USNRC and contractor personnel.

The data sources included in this document are grouped by origin: nuclear, nonnuclear, or foreign. Because the intended use of this document is in nuclear industry applications, the emphasis is placed on the nuclear data sources of U.S. origin. The foreign nuclear data sources are distinguished and given less attention because they are generally less readily accessible and often do not provide data on the population base of interest (U.S. commercial reactors).

A profile is provided for each foreign and nonnuclear data source. The profile includes a brief description of the source's contents, areas of usefulness, access information, and reference information for a contact individual. Similar profiles of 38 nuclear data sources are given. In addition, tabular comparisons are provided for the nuclear sources. The comparisons include the scope of the data sources, the operational, special purpose operational, pedigree, aggregated, and derived data contained in the data source; the operational and design data used to generate the source; the published data sources used to generate the source; and access information for the source. The tables provide for quick referencing of the data sources and making an assessment of how applicable a source is to meeting a data need.

A wide variety of data sources are included. The sources vary considerably in size, potential application, and visibility through the nuclear industry. For example, computerized data sources include the sequence coding and search system, engineered safety features actuation, human error reliability analysis, and maintenance performance indicators. Reports address topics in areas such as diesel generator performance, harsh environment common cause dependencies, and initiating event frequencies.

Probabilistic risk assessments (PRAs) have been addressed separately from other nuclear data sources. PRAs have been highlighted for a number of reasons. Because of the level of effort associated with performing PRAs, they often provide a wealth of information on a particular nuclear power plant. PRAs include useful data originating as input in addition to their useful outputs (e.g., core-melt probabilities, dominant accident sequences). For this document, emphasis is placed on the operational data sources tabulated for each PRA, in particular the identification of plant-specific data sources used in the analysis. These data sources represent a valuable commodity and have potential use in a variety of other applications, including other PRAs.

For each of over 30 PRAs, the following are provided:

1. A discussion of the data sources used in the analysis and where they appear in the reference document
2. A table of background information that describes the PRA and the plant itself (e.g., plant type, architect/engineer, PRA completion date, type of analysis)
3. A table identifying the operational data sources and how they were used in the analysis.

A review of the input data sources used in PRAs as PRA analysis has evolved over the past 10 years has shown a considerable refinement in the input data sources. Better estimates are being produced for quantities such as the number of failures, demands, exposure times, and unavailability. The use of plant-specific data for these estimates is increasing.

For this document to maintain its usefulness, it is imperative that it be updated periodically to reflect the availability of new data sources. A future data sources section is included in this report. In this section, data sources that were not published or were still in a developmental stage at the time this document was written are enumerated. A brief description is provided for each of these sources.

ACKNOWLEDGMENTS

The authors appreciate the efforts of Walter S. Mings in collecting the input for this document, all U.S. Nuclear Regulatory Commission and contractor personnel who supplied input on data sources, and Dr. Ali Mosleh of Pickard, Lowe, & Garrick Incorporated for his extensive input to the probabilistic risk assessment section of this report. In addition, the table comparison approach and foreign and nonnuclear data source information contributed by Joseph R. Fragola of Science Applications International Corporation for an earlier bibliography that was a springboard for this one are gratefully acknowledged.

CONTENTS

ABSTRACT	iii
SUMMARY	v
ACKNOWLEDGMENTS	viii
ACRONYMS	xii
1. INTRODUCTION	1
2. NUCLEAR DATA SOURCE DESCRIPTIONS AND ACCESS	3
2.1 Accident Sequence Precursor	3
2.2 Aging Root Cause	5
2.3 Component Root Cause	7
2.4 Centralized Reliability Data Organization	8
2.5 Dependent Failure/Harsh Environment Events	11
2.6 Reliability of Emergency AC Power Systems	13
2.7 Diesel Generator Availability	14
2.8 Diesel Generator Reliability	16
2.9 Diesel Generator Test Intervals	17
2.10 Electric Motor Aging	18
2.11 Engineered Safety Feature Actuation / Scram Data Base	20
2.12 Electric Power Research Institute Reports	21
2.13 Diesel Generator Performance	23
2.14 Evaluation and Update of Baseline Data	24
2.15 Human Error in Risk Assessment	26
2.16 INEL Component Data Summaries of Licensee Event Reports	27
2.17 NRC Reactor Safety Data Bank	32
2.18 Initiating Events	35
2.19 In-Plant Reliability Data System	37
2.20 IEEE Standard 500-1984	41
2.21 Licensed Operating Reactors Status (Grey Book)	43
2.22 Licensee Event Report Compilation (NUREG/CR-2000)	44
2.23 Maintenance Data Base	46
2.24 Nuclear Plant Reliability Data System	48
2.25 Nuclear Power Experience	51
2.26 Nuclear Safety Information Center Licensee Event Report Data Base	52
2.27 Operating Plant Evaluation Code (OPEC)-2	54
2.28 Pipe Break Frequency	57
2.29 Reactor Safety Study (WASH-1400)	58
2.30 Reactor Coolant Pump Seal Failures	60
2.31 Safety System Unavailability	61
2.32 Reportable Events File	63
2.33 Sequence Coding and Search System	64
2.34 Snubber Performance	66

2.35	System Interaction Events	67
2.36	Statistical Analysis of IPRDS Data	69
2.37	Technical Specifications	71
2.38	Burns and Roe Valve Study	72
3.	NONNUCLEAR DATA SOURCE DESCRIPTIONS AND ACCESS	75
3.1	Failure and Inventory Reporting System	75
3.2	Generating Availability Data System	76
3.3	Government-Industry Data Exchange Program	79
3.4	Military Handbook 217E	81
3.5	Reliability Analysis Center Handbooks	83
4.	FOREIGN DATA SOURCE DESCRIPTIONS AND ACCESS	85
4.1	Swedish Thermal Power Reliability Data System	85
4.2	European Reliability Data System	86
4.3	Gesellschaft fur Reaktorsicherheit	88
4.4	Offshore Reliability Data Handbook	90
4.5	Systeme Recueil Donnee' Fialilite'	91
4.6	System Reliability Service	92
5.	PROBABILISTIC RISK ASSESSMENT DESCRIPTIONS	95
5.1	Arkansas Nuclear Unit 1	95
5.2	Big Rock Point	97
5.3	Browns Ferry Unit 1	98
5.4	Browns Ferry Unit 1 Second Evaluation	100
5.5	Calvert Cliffs Unit 1	101
5.6	Calvert Cliffs Unit 1 Second Evaluation	102
5.7	Calvert Cliffs Unit 2	103
5.8	Connecticut Yankee	104
5.9	Crystal River Unit 3	106
5.10	Calvert Cliffs Unit 3 Second Evaluation	107
5.11	Grand Gulf Unit 1	108
5.12	H. B. Robinson Unit 2	109
5.13	Indian Point Units 2 and 3	109
5.14	Limerick Units 1 and 2	111
5.15	Millstone Unit 1	112
5.16	Millstone Unit 1 Second Evaluation	113
5.17	Millstone Unit 3	115
5.18	Oconee Unit 1	116
5.19	Oconee Unit 3	116
5.20	Oconee Unit 3 Second Evaluation	117
5.21	Peach Bottom Unit 2	118
5.22	Seabrook Units 1 and 2	119
5.23	Sequoyah Unit 1	121
5.24	Shoreham	121
5.25	Surry 1	122
5.26	Yankee Rowe	123

5.27	Zion Station	124
6.	DATA SOURCE COMPARISONS	127
6.1	Nuclear Data Source Comparisons	127
6.2	Probabilistic Risk Assessment Comparisons	152
7.	FUTURE DATA SOURCES	169
8.	REFERENCES	171

TABLES

1.	Data source overview	130
2.	Pedigree data	133
3.	Operating data	135
4.	Special purpose operational data	137
5.	Aggregated data	139
6.	Derived data	141
7.	Data origin (operational sources)	144
8.	Data origin (design sources)	146
9.	Data origin (primary published sources)	148
10.	Data access	150
11.	Summary of PRA studies	153
12.	Containment design explanation	156
13.	PRA analysis levels	157
14.	Use of reliability data in PRAs	158
15.	PRA generic data sources	165
16.	Future bibliography data sources	170

ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
AEOD	USNRC Office of Analysis and Evaluation of Operational Data
ANI	American Nuclear Insurers
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
ASP	Accident sequence precursor
ATV	Swedish Thermal Power Reliability Data System
B&R	Burns and Roe
B&W	Babcock & Wilcox
BEARDS	Baseline Events Reliability Analysis Data System
BNL	Brookhaven National Laboratory
BWR	Boiling water reactor
CDC	Control Data Corporation
CDR	Construction deficiency report
CE	Combustion Engineering
CER	Construction event report
CFR	Code of Federal Regulations
CRCTA	Composite Reactor Components Test Activity
CRDM	Control rod drive mechanism
CREDO	Centralized Reliability Data Organization
DAE	Division of Accident Evaluation
DARCOM	U.S. Army Development and Readiness Command
DBMS	Data Base Management System
DG	Diesel generator
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
ECCS	Emergency core cooling system
EDF	Electricite de France
EEI	Edison Electric Institute
EPRI	Electric Power Research Institute
ERDS	European Reliability Data System
ESF	Engineered safety feature
FFTF	Fast Flux Test Facility
FIRS	Failure and Inventory Reporting System
FIST	Fully Integrated System Test
FLECHT	Full-Length Emergency Core Heat Transfer
FSAR	Final safety analysis report
GADS	Generating Availability Data System
GDC	Generic design criteria
GE	General Electric
GIDEP	Government-Industry Data Exchange Program
GL	Generic letter

GPL	General purpose loop
GRS	Gesellschaft fur Reaktorsicherheit
HEPA	High efficiency particulate air (filters)
HTGR	High-temperature gas-cooled reactor
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IE	USNRC Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronic Engineers
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IPRDS	In-Plant Reliability Data System
IREP	Interim Reliability Evaluation Program
IRS	Incident Reporting Systems
ISDMS	INEL Scientific Data Management System
JAERI	Japan Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LCO	Limiting condition of operation
LER	Licensee event report
LMFBR	Liquid metal fast breeder reactor
LOCA	Loss-of-coolant accident
LOFT	Loss-of-fluid test
LOSP	Loss-of-offsite power
LPCI	Low pressure coolant injection
LTSF	LOFT Technical Support Facility
MDB	Maintenance data base
MIL 217E	Military Standardization Handbook 217E
MOV	Motor-operated valve
MRBT	Multi-rod burst test
MSAR	Mine Safety Appliance Research
MTBF	Mean time between failure
MTTR	Mean time to repair
NA	Not applicable
NEA	Nuclear Energy Agency
NERC	National Electric Reliability Council
NIH	National Institute of Health
NOAC	Nuclear Operation Analysis Center
NPAR	Nuclear Plant Aging Research
NPE	Nuclear Power Experience
NPEars	NPE Automated Retrieval System
NPP	Nuclear power plant
NPRDS	Nuclear Plant Reliability Data System
NREP	National Reliability Evaluation Program
NRR	USNRC Office of Nuclear Reactor Regulation
NSIC	Nuclear Safety Information Center
NSSS	Nuclear Steam Supply System

NU	Northeast Utilities
CREDA	Offshore Reliability Data Handbook
ORNL	Oak Ridge National Laboratory
OTIS	Once-through integral system
PAL	Prototype Applications Loop
PBF	Power Burst Facility (INEL)
PC	Personal computer
PLG	Pickard, Lowe and Garrick
PRA	Probabilistic risk assessment
PRIS	Power Reactor Information System
PSA	Probabilistic safety analysis
PSS	Probabilistic safety study
PWR	Pressurized water reactor
RAC	Reliability Analysis Center
RADC	Rome Air Development Center
RCP	Reactor coolant pump
RES	USNRC Office of Nuclear Regulatory Research
RMIEP	Risk Management Integration and Evaluation Program
ROSA	Rig of Safety Apparatus
RPS	Reactor Protection System
RSSMAP	Reactor Safety Study Methodologies Applications Program
RWE	Rheinische Westalisches Elektrizitätswerke
SAIC	Science Applications International Corporation
SALP	Systematic assessment of licensee performance
SAS	Statistical Analysis System
SCSS	Sequence Coding and Search System
SGTR	Steam generator tube rupture
SHBF	Single Heated Bundle Facility (GE)
SMSC	The S. M. Stoller Corporation
SNL	Sandia National Laboratories
SRDF	Systems Recueil Donnee' Fialilite'
SRE	Sodium Reactor Experiment
SRV	Safety relief valves
SSTF	Steam Sector Test Facility
SWS	Service water system
SYREL	System Reliability Service Data Banks
TAMMS	Army Maintenance Management System
TDI	Transamerica Delaval, Inc.
THERP	Technique for human error rate prediction
THORS	Thermal Hydraulic Out-of-Reactor Safety Facility
THTF	Thermal Hydraulic Test Facility (ORNL)
TI	Test interval

TLTA	Twin Loop Test Apparatus
TS	Technical specification
TTL	Transient test loop
TWC	Through-the-wall crack
UKAEA	United Kingdom Atomic Energy Authority
USNRC	United States Nuclear Regulatory Commission
W	Westinghouse
3M	Navy Maintenance, Material, Management System

ANNOTATED BIBLIOGRAPHY OF RELIABILITY AND RISK DATA SOURCES

1. INTRODUCTION

This report provides the United States Nuclear Regulatory Commission (USNRC) with a comprehensive list of sources of operational data that can be used in reliability and risk assessments of nuclear power plants (NPPs). The Integrated Risk Assessment Data Acquisition Program (IRADAP) exists to make the USNRC's use of operational data as efficient and effective as possible; increasing the availability of current operational data to all USNRC data users is a part of that goal.

To fulfill the FY 1986 scope of work for IRADAP defined by the USNRC Office for Analysis and Evaluation of Operational Data (AEOD), personnel from EG&G Idaho, Inc., interviewed USNRC and contractor personnel currently working on USNRC programs that use operational data. The primary purpose of these visits and exchanges was to assess data needs;¹ however, information on the data sets--both formal and informal--that are currently being used to meet those needs was also collected. In addition, questionnaires were distributed and, in some cases, followed by phone contacts to clarify information. This information was supplemented by source information gathered in the FY 1985 IRADAP effort² and by a data base bibliography³ compiled in 1982 for USNRC use as part of the USNRC Office of Research's National Reliability Evaluation Program.

Summaries describing each of the resulting data sources are contained in the next four sections. The greatest amount of detail is found in the section on U.S. nuclear sources; foreign sources and nonnuclear sources are included for completeness where information was available. In each case, a brief summary of the content of the data source and access information is provided. The sources include conventional computerized data bases, data bases that are supplemented by reports, and reports based on information data records that may or may not be computerized. Information about the

sources also includes the time frames covered by their data and whether they are updated.

The fourth data source section deals with probabilistic risk assessments (PRAs) of U.S. commercial NPPs. A specially designed questionnaire was distributed for this study to collect information about the operational data sources that have been used in PRAs. This information has two benefits. First, a knowledge of how existing generic sources have been used in past risk assessments provides further insights on their applicability. The second and most important benefit is that many of the PRAs contain plant-specific operational data that could be used in risk assessments of similar plants. Thus, the PRAs and their supporting documentation in many cases are operational data sources in their own right. The PRA source section of this report provides background information and an evaluation of all the NPP PRAs from this point of view.

The individual data source description sections are followed by a summary section that contains tabular comparisons of the profiled U.S. nuclear data sources. Finally, a section lists future sources that are currently being developed. These sources should be described in more detail in future editions of this bibliography. The updating of this bibliography every other year is recommended to keep it current.

2. NUCLEAR DATA SOURCE DESCRIPTIONS AND ACCESS

A description of the fundamental features of the nuclear data sources and access information is provided in this section. Also included is itemized background information for each source. The itemization includes the type of source (e.g., report versus computerized data base), updating frequency, and the time frame that the data, or the input data to the source, span. This itemization provides a quick means of assessing, at an unfocused level, a data source's potential use (i.e., whether further inquiries are warranted).

The data source descriptions address topics such as the data origin, how the source was developed, the specific types of data the source includes, and potential applications of the data.

The access information for each source includes a contact person, how to obtain a copy of the source, computer access methods, and access limitations.

Tabular comparisons of the nuclear data sources are presented in a later section of this report.

2.1 Accident Sequence Precursor

Sponsor:	USNRC-AEOD
Type of source:	Reports/computerized data base
Industry:	U.S. commercial nuclear
Number and type of records:	350 events representing occurrences of accident precursors
Frequency of record update:	Annual
Data Source Boundary:	Events reported under licensee event report (LER) requirements and meeting one of the following requirements:

- o the failure of at least one function required to mitigate a loss of main feedwater, loss-of-offsite power (LOSP), small-break loss-of-coolant accident (LOCA), or steam line break
- o the degradation of more than one function required to mitigate one of the above initiating events
- o an actual initiating event that required safety function response.

Time frame: January 1969 through December 1981 and January 1985 through December 1985

2.1.1 Description

The Accident Sequence Precursor (ASP) program was established at Oak Ridge National Laboratory (ORNL) in 1979. The program has provided a data base of nuclear power plant potentially severe accident experience. The program has produced three reports (NUREG/CR-2497,⁴ NUREG/CR-3591,⁵ NUREG/CR-4674⁶) that identify, categorize, and quantify precursors to potential severe core damage accidents over different time periods. Collectively, the reports cover precursors occurring from 1969 through 1981 and during 1985.

In the reports, LER and other plant data are used to calculate the ordinary unavailability of plant safety functions. The expected average frequency of initiating events (loss of feedwater, LOSP, LOCA, and steam line breaks) is determined, when possible from the precursors. Next, the calculated unavailabilities and initiating event frequencies are used with the event details to evaluate the potential impact of the safety function unavailability and/or initiating event occurrences that happened during each precursor event. This potential impact is expressed in the form of an estimate of the conditional probability of potential severe core damage for each precursor, given the initiating event frequency and function failure probability estimates that would apply during the precursor event. These inputs are used in conjunction with event trees.

The reports also include the precursors ranked by significance, a listing of dominant sequences, and descriptions and data on each precursor.

The precursor description and supporting data are maintained on a personal computer (PC) for storage and easy retrieval purposes. The data include the record number, date, the failure sequence, cause components involved, and a host of other data. In addition, interactive user-friendly software exists for assessing the potential for core damage for a specific precursor and scenario of events.

2.1.2 Data Source Access

Contact: Joe Minarick
Address: Science Applications International Corp.
P.O. Box 2501
Oak Ridge, TN 37831
Phone: 615-482-9031
Report ordering address: National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161
Report cost: Unknown
Report accessibility: No restrictions
Computer access: All requests for this data and the analysis code should be made through Fred Manning at NRC/AEOD, 301-492-7418 (FTS 492-7418)

2.2 Aging Root Cause

Sponsor: USNRC-RES
Type of source: Report/computerized data base
Industry: U.S. commercial nuclear
Number and type of records: 852 component failure records from the Nuclear Plant Reliability Data System (NPRDS)
Frequency of record update: Irregular

Data source boundary: NPRDS reports on service water system (SWS) for BWR and Babcock & Wilcox (B&W) plants and the 1E electrical power distribution system for B&W plants

Time frame: NRPDS records since its inception (1974)

2.2.1 Description

The report⁷ presents the results of a detailed root cause analysis of component failure performed for the USNRC Nuclear Plant Aging Research (NPAR) Program. In the analysis, the categorization and coding technique developed in the EG&G Idaho Root Cause of Component Failure Program was used. The report identifies aging-related failures at the root cause level from NPRDS data. The analysis was performed on a system basis (SWS and 1E electrical). The data provide insights into the effects of aging versus nonaging failures on system performance. The components dominating the failure contributions are identified. Also, root cause and aging-related failure fractions for components are given.

The root cause data was processed using dBASE III. The data base is structured so that easy retrieval of information specific to system, component, failure mode, root cause, and age are possible. Because NPRDS is proprietary, the data was reported generically with respect to plants. To obtain a PC disk of the data, permission from the Institute of Nuclear Power Operations (INPO) (the sponsor of NPRDS) would be required in addition to USNRC-RES approval.

2.2.2 Data Source Access

Contact: David Satterwhite
Address: EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415
Phone: 208-526-6067 (FTS 583-6067)
Report ordering address: Same as above
Report cost: None

Report accessibility: No restrictions

Computer access: A copy of the data base in the form of a floppy disk is obtainable with approval from J. T. Vora (USNRC-RES), 301-443-7673 (FTS 443-7673) and INPO

2.3 Component Root Cause

Sponsor: USNRC-RES

Type of source: Report/computerized data base

Industry: U.S. commercial nuclear

Number and type of records: Approximately 375 events describing the root causes of component failures

Frequency of record update: None

Data source boundary: Root cause data as captured by LER reporting for the following components: motor-operated valves, pneumatic-operated valves, check valves, relief valves, turbine-driven pumps, motor-driven pumps, and circuit breakers

Time frame: 1980 through 1985

2.3.1 Description

The report⁸ presents the results of the development and application of a three-tier categorization scheme for root causes of component failures. The data provide a valuable input for reliability assurance and probabilistic risk assessments.

Root cause information on component failures was coded for the seven components cited above. For selected failure modes for each component, 25 reports with third-tier root cause data were coded. The data were obtained from LERs. When necessary, NPRDS data from a trial coding session was used to supplement the LER data. Common cause failure information was included in the coding.

Root cause fractions that indicate the relative contribution of a particular root cause for a component failure in a given failure mode were calculated.

The data are maintained on a PC through use of dBASE III for easy retrieval and editing.

2.3.2 Data Source Access

Contact: David Satterwhite
Address: EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415
Phone: 208-526-6067 (FTS 583-6072)
Report ordering address: Same as above
Report cost: None
Report accessibility: No restrictions
Computer access: A copy of the data base in the form of a floppy disk is obtainable with approval from P. K. Niyogi (USNRC-RES), 301-443-7612 (FTS 443-7612).

2.4 Centralized Reliability Data Organization

Sponsor: U.S. Department of Energy (DOE)
Type of source: Periodic reports/computerized data base
Industry: U.S. nuclear [liquid metal fast breeder reactor (LMFBR), high-temperature, gas-cooled reactor (HTGR)]
Number and type of records: 1500 failure events; engineering data for 20,000 components found in advanced reactors (e.g., LMFBRs)
Frequency of record update: Daily
Data source boundary: The following advanced reactor sites and test facilities: Composite Reactor Components Test Activity (CRCTA), Experimental Breeder

Reactor-II (EBR-II), Fast Flux Test Facility (FFTF), General Purpose Loop (GPL), Mine Safety Appliance Research (MSAR), Prototype Applications Loop (PAL), Sodium Reactor Experiment (SRE), Thermal Hydraulic Out-Reactor Safety Facility (THORS), and Transient Test Loop (TTL)

Time frame:

Engineering data have been collected since 1978, and historical data have been compiled since the early 1960s

2.4.1 Description

The Centralized Reliability Data Organization (CREDO) data base is maintained at ORNL to provide a central computer-based source of accurate, timely data and information for use in reliability and availability analysis of advanced reactors (LMFBRs and HTGRs).⁹ CREDO is a component-based system, i.e., engineering, operating, and event data focus on a specific item identified as a component within a given system. CREDO uses a comprehensive list of 45 generic components that are representative of all components found at any reactor site. Each component may be associated with only one system and subsystem at a time. Because system and subsystem boundaries and nomenclature vary from site to site, CREDO has also developed a generic list of systems and subsystems.

The data base management system (DBMS) used for CREDO catalogs and stores data in three types of files that correspond to their respective data types:

1. Engineering data--design and operating characteristics and information on the component's application in its particular unit/system. Usually, engineering data are reported once per component and updated when necessary
2. Operating data--chronological sequential reports that provide a profile of the accumulated operating history of the reporting unit. The length of the reporting period is based on the most

logical operational increment (e.g., quarterly, length of test run, reactor experiment run). Each operating data report form includes the hours of unit operation in specified reactor modes and provides information about facility availability

3. Event data--data concerning any CREDO reportable event that occurs to components being tracked by the CREDO system. These data include a description of the event, method of detection, failure mode, failure cause, corrective action, etc.

The combination of the information contained in the engineering, operating, and event data files will allow for the calculation of various reliability parameters such as average failure rate, mean-time-to-failure, mean-time-to-repair, etc. In addition, various types of qualitative outputs can be generated for use by a data analyst.

The data base is maintained on an IBM mainframe computer with a combination of JOSHUA and ORCHIS DBMS. Although the DBMS is too large for transport on floppy disk, subsets of the data base are obtainable in the form of PC floppy disks. In addition, data base searches will be performed by ORNL on request.

There are costs associated with access and use of the data base. However, the following organizations are exempt from this fee: ORNL, DOE, DOE contractors, and users under DOE's Applied Technology classification. Anyone outside of these groups must obtain DOE approval for access to the data base.

2.4.2 Data Source Access

Contact:	John J. Manning
Address:	Martin Marietta Oak Ridge National Laboratory P.O. Box X Oak Ridge, TN 37830
Phone:	615-574-5288 (FTS 624-5288)

Report ordering address: Same as above
Report cost: Unknown
Report accessibility: No restrictions
Computer access: See contact above for information on how to obtain DOE approval for ORNL data base searches or PC floppy disks with subsets of the data base

2.5 Dependent Failure/Harsh Environment Events

Sponsor: USNRC-RES
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: 700 events representing occurrences of common cause failures and failures caused by harsh environments
Frequency of record update: None
Data source boundary: LERs on failures of 26 component and subcomponent types
Time frame: Intermittent: January 1971 through December 1985

2.5.1 Description

This is a letter report¹⁰ from JBF Associates Inc., to Sandia National Laboratories (SNL) summarizing JBF's efforts to analyze dependent (common cause) failures and failures caused by harsh environments. The information used for the analysis was comprised of over 1000 failure reports (mostly abstracts of LERs that were assembled for other studies).

The 26 groups of components selected for study are: accumulators, batteries, cables, control rod drives, dampers, diesel generators, drains, air filters, fuel bundles, heat exchangers, heaters, instrumentation and controls (I&C), motors, offsite power supplies, access penetrations,

pipings, condensate polishers, pumps, electrical equipment, scram discharge volumes, shock suppressors, spargers and nozzles, strainers, transformers, valves, and other. Air filters include moisture separators, steam traps, high efficiency particulate air (HEPA) filters, charcoal filters, roughing filters, and prefilters. The I&C category includes sensors, switches, detectors, monitors, transmitters, cables, amplifiers, bistables, calculators, comparators, and summators. Electrical equipment includes relays, breakers, starters, and timers. The "other" category applies to those events in which the components involved are not explicitly indicated (e.g., the following event has been reported for a BWR: "ECCS components have doubtful temperature ratings for LOCA").

These data were studied and 700 unique events of interest were identified and subsequently sorted into 10 data sets for analysis. The data sets were constructed with different areas of focus for dependent failure (e.g., PWR-BWR comparisons, component type, harsh environment). The description for each event includes LER number, NSIC number, date, plant, event description, the failure cause, common cause classification, component type, and, in several cases, the number of failed/unfailed components.

2.5.2 Data Source Access

Contact:	Michael Bohn
Address:	Sandia National Laboratories Division 6412 P.O. Box 5800 Albuquerque, NM 87185-5800
Phone:	505-966-5232 (FTS 844-5232)
Report ordering address:	Same as above
Report cost:	Unknown
Report accessibility:	This report is the property of SNL and must be obtained through SNL

2.6 Reliability of Emergency AC Power Systems

Sponsor: USNRC-NRR

Type of source: Report

Industry: U.S. commercial nuclear

Number and type of records: 900 occurrences of diesel generator failure at U.S. nuclear power plants

Frequency of record update: One update in 1985

Data source boundary: LERs, NUREG-0737¹¹ responses, and responses to a questionnaire designed specifically for the study

Time frame: January 1976 through December 1980

2.6.1 Description

Station blackout, or loss of all ac power, has been identified in many PRAs as a major contributor to risk. This is because of the disablement of all normal and most emergency cooling systems that occur during loss of ac power. In addition, most engineered safety feature systems that would contain radioactive material given a reactor accident are disabled as a result of station blackout. The seriousness of these consequences provided the major motivation for this study.¹² Specific plants were selected to estimate onsite ac power system reliability based on the most realistic data available, but the reliability estimates calculated for the specific plants were calculated with the intent of being used as representative figures for any plant with the design and operational features identified in this report.

The sources of data for this report were:

1. Abstracts of LERs
2. Emergency core cooling system (ECCS) outage data submitted to the USNRC by licensees in response to a questionnaire associated with NUREG-0737

3. Diesel generator data submitted to the USNRC in response to a questionnaire prepared as part of the study.

The bulk of the information in the report is included in a 317-page appendix that contains systems descriptions, station blackout fault trees, diesel generator historical data, and diesel generator common cause failure analysis results for 18 different nuclear power plants. Tables and graphs are well organized and present data correlated to each plant studied. The study also contains conclusions and recommendations for improving reliability.

2.6.2 Data Source Access

Contact: Ronald E. Battle
Address: Oak Ridge National Laboratory
Building 3500, MS-8
Oak Ridge, TN 37831
Phone: 615-574-5531 (FTS 624-5531)
Report ordering address: GPO Sales Program
Division of Technical Information and Document
Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Report cost: \$8.00
Report accessibility: No restrictions

2.7 Diesel Generator Availability

Sponsor: USNRC-NRR
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: 600 occurrences of diesel generator (DG)
failure
Frequency of record update: None scheduled at this time

Data source boundary: U.S. nuclear plant DG failures for the specified time frame of the study as reported in LERs, and responses to Generic Letter 84-15

Time frame: January 1981 through December 1983

2.7.1 Description

The purpose of this report¹³ is to update the analysis of the operating experience of emergency DGs in nuclear power plants contained in NUREG/CR-2989 (see Section 2.6 above). The contents of the reports differ and are therefore described separately. However, the sources are combined for the tabular comparisons of nuclear data sources (Section 6.1).

The LER data base served as the primary source of DG failure data, while a data base for DG successes was formed from nuclear plant licensees' responses to a USNRC questionnaire (Generic Letter 84-15). Estimates of DG failure on demand were calculated from the LER data, DG test data, and response data from the questionnaire. The questionnaire also provided data on DG performance during complete and partial LOSP and in response to safety injection actuation signals. Trends in DG performance are profiled. The effects of testing schedules on diesel reliability are assessed. Individual failures are identified in an appendix.

2.7.2 Data Source Access

Contact: Ronald E. Battle

Address: Oak Ridge National Laboratory
Building 3500, MS-8
Oak Ridge, TN 37831

Phone: 615-574-5531 (FTS 624-5531)

Report ordering address: GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

2.8 Diesel Generator Reliability

Sponsor: USNRC-NRR
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: 5000 events representing occurrences of attempts to start emergency DGs (successes and failures)
Frequency of record update: None
Data source boundary: U.S. nuclear industry response to USNRC and Brookhaven National Laboratory (BNL) questionnaires
Time frame: January 1980 through December 1985

2.8.1 Description

The report¹⁴ provides an analysis of data received from utilities in response to the USNRC Generic Letter 84-15. Inputs obtained through responses to a BNL questionnaire designed specifically for the study were also included in the analysis. Recommendations made for DG reliability by other various groups were also evaluated. The other groups include industry organizations (such as INPO and ASME), DG manufacturers or vendors, foreign DG users, the Advisory Committee on Reactor Safeguards (ACRS), and some miscellaneous groups.

Report recommendations include ways to improve DG reliability and ways to improve DG maintenance programs. Other information provided includes: the DG reliability at every site for the last 20 starts and last 100 starts, a listing of the number of DGs per unit, summaries of responses to GL 84-15 including population data, and summaries of individual utility responses to the BNL questionnaire.

2.8.2 Data Source Access

Contact: James Higgins
Address: Brookhaven National Laboratory

Building 130
Upton, NY 11973

Phone: 516-282-2432 (FTS 666-2432)

Report ordering address: NRC Public Document Room
1717 H Street, N.W.
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

2.9 Diesel Generator Test Intervals

Sponsor: USNRC-NRR

Type of source: Report/computerized data base

Industry: U.S. commercial nuclear

Number and type of records: Number of records is unknown. Records are for DG unavailability for 35 nuclear power plants, with test and accident unavailability distinguished

Frequency of record update: None

Data source boundary: Scope and detail of plant maintenance records

Time frame: Unknown

2.9.1 Description

This report¹⁵ presents an analysis of DG unavailability, caused both by failure occurring while the DG is on standby and test-caused failures. The report presents a methodology for determining testing intervals (TIs) so that diesel unavailability is at an acceptably low level. Sensitivity analyses of test unavailability and accident unavailability to varying TIs are presented.

PC-based models are presented for evaluating diesel unavailability. Parameters for the models are discussed in the report, but individual DG unavailability events are not listed. Generic TIs for a range of parameters and population of plants are displayed.

The software for the PC-based algorithm evaluating the effect of varying TI on unavailability is transferable by PC disk. The software required to support the analysis is Lotus 1-2-3. One hypothetical set of parameters for the model is included on the disk.

2.9.2 Data Source Access

Contact: John Boccio
Address: Brookhaven National Laboratory
Building 130
Upton, NY 11973
Phone: 516-282-7690 (FTS 666-7690)
Report ordering address: Same as above
Report cost: Unknown
Report accessibility: No restrictions
Computer access: See contact above for analysis software. The data base is not available

2.10 Electric Motor Aging

Sponsor: USNRC-RES
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: Over 500 events representing occurrences of electric motor failure in nuclear power plants
Frequency of record update: None
Data source boundary: Failures of electric motors captured by LER, NPRDS, IPRDS, and Nuclear Power Experience (NPE) reporting
Time frame: January 1974 through December 1983

2.10.1 Description

The report¹⁶ provides an aging assessment of electric motors and was conducted under the auspices of the USNRC NPAR. Pertinent failure-related information was derived from a variety of sources for the study. The sources used were LERs, IPRDS, NPRDS, and NPE. Failure modes, mechanisms, and causes for motor problems were reviewed from operating experiences described in these sources.

In addition, motor design and materials of construction were reviewed to identify age-sensitive components. The study included consideration of the seismic susceptibility of age-degraded motor components, that is whether the failure mode, mechanism, and cause are directly attributable or potentially susceptible to externally induced vibrational effects.

The aforementioned reviews and assessments were assimilated to characterize the effect of dielectric, rotational, and mechanical hazards on motor performance and operational readiness. The functional indicators were identified that can be monitored to assess motor component deterioration caused by aging or other accidental stressors. The study also includes a preliminary discussion of current standards and guides, maintenance programs, and research activities pertaining to nuclear power plant safety-related electric motors. Included are motor manufacturer recommendations, responses from repair facilities to a questionnaire, in-service inspection data, expert knowledge, USNRC-IE audit reports, and standards and guides published by the Institute of Electrical and Electronics Engineers (IEEE).

2.10.2 Data Source Access

Contact:	Manomohan Subudhi
Address:	Brookhaven National Laboratory Building 130 Upton, New York, 11973
Phone:	516-282-2429 (FTS 666-2429)

Report ordering address: GPO Sales Program
Division of Technical Information and Document
Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

2.11 Engineered Safety Feature Actuation/Scram Data Base

Sponsor: USNRC-AEOD

Type of source: Computerized data base

Industry: U.S. commercial nuclear

Number and type of records: Approximately 1650 events per year describing
engineered safety feature (ESF) actuations and
650 scram events/year

Frequency of record update: Annually

Data source boundary: Events reported in accordance with
10 CFR 50.73(a)(2)(iv)

Time frame: January 1985 to current

2.11.1 Description

The data base is the product of a study conducted for AEOD by the Idaho National Engineering Laboratory (INEL). Unplanned ESF actuations and scrams are reportable as Licensee Event Reports under 10 CFR 50.73 (a)(2)(iv) and have been identified and categorized. These events are also noted in the NRC's Reportable Events File (see Section 2.32); as a result, the events are entered into the ESF/Scram data base within 5 days of their occurrence.

The characterization of ESF occurrences and scrams provides an understanding of the frequency and cause of these events for the industry as a whole and for various subsets [plants, utilities, nuclear steam supply system (NSSS) vendors, reactor types, etc.]. A study of the encoded data also helps one assess the significance and implications of the current ESF actuation and scram occurrence rates, whether specific actions by the NRC or

industry appear warranted, and the usefulness of such rates as suitable measures for indicating licensee performance.

The data base was developed in dBASE III PLUS and is maintained on a PC. Documentation on the structure of the data base exists but is not contained in any formal publication. A disk of the data base is available only through AEOD (see below).

2.11.2 Data Source Access

Contact: Wayne Christenson
Address: EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415
Phone: 208-526-9289 (FTS 583-9289)
Computer access: Contact M. R. Harper
USNRC-AEOD (FTS 492-4497)

2.12 Electric Power Research Institute Reports

Sponsor: Electric Power Research Institute (EPRI)
Type of source: Reports
Industry: U.S. electrical power generation; reports with "NP" in their number describe nuclear plants
Number and type of records: Varies from one report to another; see examples in text below
Frequency of record update: Irregular
Data source boundary: Varies; see examples in text below
Time frame: Varies

2.12.1 Description

EPRI funds research on various topics dealing with the generation of electric power. Reports for most of these projects are available from the

Research Reports Center free of charge to EPRI member utilities and affiliates, contributing nonmembers, U.S. utility associations, U.S. government agencies, and foreign organizations with which EPRI maintains exchange agreements. EPRI maintains a catalog of all its publications (EPRI Guide¹⁷) for the four-year period prior to the catalog's date of publication. The number of these reports rendered it infeasible to attempt to identify and profile each individually. The EPRI Guide is readily available and provides a brief synopsis for each report and can be used as a preliminary screening tool for assessment of a report's potential use. Examples of EPRI reports are discussed below. These reports are in the areas of transients, DGs, and component reliability.

ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients, EPRI NP-2230,¹⁸ and Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis, EPRI NP-2301,¹⁹ both deal with transient events. EPRI NP-2230 lists various transient categories which have resulted in fast reactor shutdowns (scrams). Fifty-two nuclear power plants have contributed data on a total of 2,996 events, representing 315 plant years of experience. The data are from the plants' commercial dates through approximately April 1980. Occurrence rates are featured with means and standard deviations, broken down by plant type, power level, and the cause of the transients. EPRI NP-2301 examines data on 45 LOSP incidents. The data were collected from 47 nuclear power plant sites and used to derive frequency estimates of LOSP and time to regain offsite power (recovery time).

Diesel Generator Reliability at Nuclear Power Plants: Data and Preliminary Analysis, EPRI NP-2433,²⁰ addresses emergency DGs. The report uses DG performance data from nuclear power plants to estimate probabilities of failure to start and failure to continue to run. The data on diesel failures is taken from plant records, utility surveys, or LERs; the event descriptions are provided in two appendixes.

Component Failure and Repair Data for Gasification--Combined Cycle Power Generation Units, EPRI AP-2205,²¹ addresses components in coal-fired

plants. The failure rate estimates for various generic components are garnered from Edison Electric Institute (EEI) estimates, failure mode analysis, and expert consensus. Failure rates and restoration rates are featured as mean time between failures (MTBF) and mean time to repair (MTTR). This data, while not directly applicable to nuclear components, could be applicable to reliability and risk assessment where components are similar.

2.12.2 Data Source Access

Contact (for reports related to nuclear power): D. H. Worledge

Address: Electric Power Research Institute
3412 Hillview Avenue
Palo Alto, CA 94304

Phone: 415-965-4081

Report ordering address: Research Reports Center
P.O. Box 50490
Palo Alto, CA 94303

2.13 Diesel Generator Performance

Sponsor: USNRC-IE

Type of source: Report

Industry: U.S. commercial nuclear

Number and type of records: 500 occurrences of DG failure at nuclear power plants

Frequency of record update: None

Data source boundary: DG failures reported in LERs, 10 CFR 50.55E, Part 21, NPRDS, and EPRI document files

Time frame: January 1980 through December 1983

2.13.1 Description

The report²² evaluates recent DG and DG vendor performance. All DG vendors were reviewed with the exception of Transamerica Delaval, Inc.

(TDI), because of the emphasis already being given to TDI diesels in other studies. For the period 1980 through 1983 inclusive, BNL reviewed and evaluated DG failure data, DG vendor inspection reports, the TDI lessons learned as they related to the other vendors, and previous pertinent studies. The data sources used for DG failure analysis were LERs, 10 CFR 50.55E, Part 21, NPRDS, and EPRI document files. The DG failures were classified relative to the DG component that failed (e.g., main bearings, starting system). The failures were also categorized and analyzed by mode, manufacturer, and cause. Manufacturers with significant failures are identified in the report.

2.13.2 Data Source Access

Contact: James Higgins
Address: Brookhaven National Laboratory
Building 130
Upton, NY 11973
Phone: 516-282-2432 (FTS 666-2432)
Report ordering address: NRC Public Document Room
1717 H Street, N.W.
Washington, D.C. 20555
Report cost: Unknown
Report accessibility: No restrictions

2.14 Evaluation and Update of Baseline Data

Sponsor: USNRC-RES
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: Not applicable
Frequency of record update: None
Data source boundary: A variety of nuclear power industry operational data sources used to supplement the original baseline data generated by

experts from the USNRC, electric utilities, NSSS vendors, nuclear consulting firms, and national laboratories

Time frame: Varies; see text below

2.14.1 Description

In April 1982, a data workshop was held to evaluate, discuss, and critique data in order to establish a consensus generic data set for the USNRC-RES National Reliability Evaluation Program (NREP).²³ The data set contained component failure rates and probability estimates for LOCA events, transients, LOSP events, and human errors that could be applied consistently across the nuclear power industry as screening values for initial identification of dominant accident sequences in PRAs. This data set was used in the development of guidance documents for the performance of PRAs.^{24,25}

In May 1984, an update effort was undertaken by EG&G Idaho, Inc., at INEL to reevaluate and update the original data set as presented in Reference 25 so that it might be more applicable to fault tree/event model quantification. The subject report²⁶ describes the results of this effort. The sources/topics discussed include the following: LERs,²⁷ INEL LER Data Summaries,²⁸⁻³⁴ plant-specific data,³⁵⁻³⁹ WASH-1400,⁴⁰ IEEE Std. 500-1984,⁴¹ transient initiating event frequency reports,⁴²⁻⁴⁴ Nuclear Power Experience,⁴⁵ NPRDS failure rate estimates, the Handbook of Human Reliability,⁴⁶ information from the In-Plant Reliability Data System (IPRDS),⁴⁷⁻⁴⁸ DG common mode failure and recovery,^{49,50} LOSP,¹⁹ failure-to-scrum probability,⁵¹ and fire and floods.⁵² Most of these references and the time frames they cover are discussed elsewhere in this bibliography. The EG&G Idaho report contains information summarized from these sources; it contains little new data.

The component failure rate data provide high-level information on common nuclear plant component types, but do not go to the level of detail necessary for most PRA applications. Data on transient occurrence appear to

be applicable to initiating event frequency calculations because they originate from plant-specific data and attempt to address both reactor types (PWR and BWR) and plant anomalies. Special studies may find the specific diesel, feedwater, LOSP, and failure-to-scrum data helpful, especially when plant unavailability considerations are being made.

2.14.2 Data Source Access

Contact: Cynthia D. Gentillon
Address: EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415
Phone: 208-526-9891 (FTS 583-9891)
Report ordering address: Same as above
Report cost: Unknown
Report accessibility: Report is informal; contact
Richard C. Robinson of USNRC-RES
(FTS 443-7620) for access permission

2.15 Human Error in Risk Assessment

Sponsor: USNRC-RES
Type of source: Computerized data base/report
Industry: U.S. commercial nuclear
Number and type of records: The data base contains 1976 records on human errors analyzed for 19 different PRAs
Frequency of record update: None
Data source boundary: Human reliability data from PRAs of nuclear power plants
Time frame: PRAs completed as of 1984

2.15.1 Description

The human error in risk assessment (HERA) data base contains 1,976 records on human errors that were analyzed in the course of completing 19

PRA's (7 for BWRs and 12 for PWRs). The data for HERA were extracted from the human reliability analysis (HRA) segment of each PRA. The purpose of establishing this data base was to assess the degree to which human reliability data from PRA's of nuclear power plants are useful in addressing human risk reduction issues of concern to the USNRC.

The development, contents, and structure of the HERA data base are documented in NUREG/CR-4103.⁵³ The contents of each HERA record is provided in the report.

The HERA data base is maintained on a PC. A copy of the data base can be obtained in the form of a PC disk.

2.15.2 Data Source Access

Contact: Claire M. Spettell
Address: Brookhaven National Laboratory
Building 130
Upton, New York 11973
Phone: 516-282-2700 (FTS 666-2700)
Report ordering address: National Technical Information Service
Springfield, Virginia 22161
Report cost: Unknown
Computer access: See contact above

2.16 INEL Component Data Summaries of Licensee Event Reports

Sponsor: USNRC-RES
Type of source: Report/computerized data base
Industry: U.S. commercial nuclear
Number and type of records: One-line event descriptions, as follows:
Batteries and battery chargers 212
Control rods and drive mechanisms 199
Diesel generators 425

Instrumentation and controls	5092
Inverters	161
Primary containment penetrations	143
Protective relays, circuit breakers	1028
Pumps	1026
Valves	2923

Frequency of record update: Irregular; none expected in the future
(the program ended in January of 1985)

Data source boundary: LERs reported according to plant Technical
Specification requirements and USNRC
Regulatory Guide 1.16

Time frame:	Battery/battery chargers	1/76 to 12/81
	Control rods/drive mech.	1/72 to 04/78
	Diesel generators	1/76 to 12/78
	Instrumentation/controls	1/76 to 12/81
	Inverters	1/76 to 12/82
	Primary containment pen.	1/72 to 12/78
	Pro. relays/circuit breakers	1/76 to 12/83
	Pumps	1/72 to 09/80
	Valves	1/76 to 12/80

2.16.1 Description

EG&G Idaho, Inc., at the INEL reviewed LERs, both qualitatively and quantitatively, to extract reliability information in support of the USNRC's effort to gather and analyze component fault data for U.S. commercial nuclear power plants. LERs describing failures or command faults for selected components (listed above) have been analyzed in this program. Failures are instances of components not functioning as required that require repair; command faults, on the other hand, are instances of components not functioning as required because of inputs or lack of inputs from personnel, other components, or the external environment.

The data reported in the LERs were qualitatively evaluated and pertinent information (e.g., fault mode, fault cause, and event date) contained in each relevant LER was coded into a one-line event description. Each one-line description was then stored in a computer-based data file for future use. Data in this computerized file can be searched, sorted, collated, retrieved, updated, and displayed by almost any item of

information contained in the original LER. This feature makes the one-line LER data base useful for obtaining various LER summary statistics for use in analyses of component and fault events. A summary description of the data bases for most of the components is available in the Licensee Event Report Data Base User's Guide.⁵⁴

The data bases for each selected component type have been described in a series of reports. With a few exceptions, the body of each report has two major parts. First, the methodology used in encoding the LERs is described. Included are the assumptions, definitions, and limitations used in carrying out the analysis. Next, a summary of the data according to various encoded characteristics is provided. The summary of results also contains fault and failure rates for significant component-fault mode combinations. These are in most cases restricted to safety systems, where LER reporting is likely to be more complete. Denominator information for the rates comes from plant final safety analysis reports (FSARs) together with very coarse estimates of numbers of demands. Specific plant fault data were averaged to obtain rates for the four NSSS vendors, for PWRs, for BWRs, and for the aggregate of both reactor types. Chi-square bounds for the rates are computed.

Appendices in the reports provide explanations of LER reporting variations, the LER coding scheme, and the methods used to estimate the fault rates. Additional appendixes contain sorts of the one-line descriptions by such attributes as NSSS vendor, human factor causes, system, and type of event (command faults, common cause susceptibilities, recurring events). Detailed plant-specific failure rate estimates are also contained in appendixes.

In the paragraphs below, comments on special features of the one-line data bases for each component type and exceptions to the general outline presented above are noted.

The battery and battery charger report⁵⁵ is the only one that does not contain fault rate estimates. Also, it is not a NUREG. The scope of

that particular study stopped short of obtaining population counts and fault rates because the USNRC's Reactor Risk Branch had more detailed population and duty cycle information for these calculations. Fault modes for battery and charger faults were categorized as degraded or inoperable.

For control rod drives and drive mechanisms,²⁸ gross constant demand and hourly fault rates were estimated for selected fault modes and combinations of fault modes. Due to improved LER reporting changes (effective January 1, 1976), two time periods were used in the estimates: (a) January 1, 1972 to April 30, 1978 and (b) January 1, 1976 to April 30, 1978.

For DGs,²⁹ LER-based fault rates were estimated for selected fault modes and classified as standby, demand, or operating fault rates. Command faults were included in the estimation process because LERs generally report safety system degradation; the fault rates should reflect this by including all events that would prevent the DG from supplying emergency power on demand.

For the instrumentation and control component report,³³ gross constant fault rates were estimated for major components and instrument channels that provide a direct reactor trip.

For inverters,³⁰ gross constant failure rates were estimated for essential ac distribution and BWR low-pressure coolant injection system static inverters having the inoperable fault mode.

For primary containment penetrations,³¹ gross constant standby and operating failure rates were estimated for selected failure modes and combinations of failure modes. Demand failure rates for access penetrations were not estimated because the component demands necessary for a demand rate estimation could not be determined. Due to improved LER reporting changes (effective January 1, 1976), two time periods were used in the estimates: (a) January 1, 1972 to December 31, 1978 and (b) January 1, 1976 to December 31, 1978.

For protective relays and circuit breakers,⁵⁶ the sorting of data by numerous influencing parameters and their display in bar graphs allows the analyst to see which factors affect relays and circuit breakers, and to what degree. Gross constant fault rates were estimated for those relays and circuit breakers found in the Class 1E medium and low voltage power systems. Note that this report is not a formal NUREG and is not available through the GPO Sales Program; it was issued in draft form and funds were not available to make it formal and publish it.

For pumps,³² LER-based demand, standby, and operating fault rates were estimated according to various fault modes. The validity and applicability of this data may depend on recent changes in technical specifications or preventive maintenance and testing schedules that alter the demand spectrum. Inclusion of separate fault rates by operating mode (i.e., running, alternating, or standby) provides useful distinctions for PRA model applications.

For valves,³⁴ the limited availability of system component population information limited the scope of the demand and standby fault rate estimates to valves in selected engineered safety features systems plus primary relief and safety valves. The following fault/failure modes were included in this report: failed to open, failed to close, internal leakage, external leakage, reverse leakage, failed to operate as required, plugged, premature open, maintenance, test not performed, and improper valve configuration.

The component faults and failure rates summarized in all these reports should be interpreted as tentative gross indicators of true fault trends and failure rates. Because subjective judgments had to be made regarding pertinence of recorded events, and because some component faults may not be recorded in the LERs, individual analysts should confirm the applicability of the component faults and failure rates for their specific uses. The failure rates are useful in gross risk and reliability evaluations.

Each data base, composed of one-line data for a selected component or component group, was developed on a CDC Cyber 176/NOS Operating System using CDC's "Query Update" software package. The data bases are maintained on a magnetic tapes in archival storage. Copies of the tapes are obtainable through EG&G Idaho with RES approval. The development of applications programs could be required to produce output other than that provided in the data summary report. Standard INEL Cyber processing charges are applicable.

2.16.2 Data Source Access

Contact: Ms. Cynthia D. Gentillon

Address: EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

Phone: 208-526-9891 (FTS 583-9891)

Report ordering address: GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555,
except for the battery/battery charger and relay/circuit breaker reports; for these, see above

Report cost: Unknown

Report accessibility: No restrictions

Computer access: Contact Richard C. Robinson, USNRC-RES
(301-443-7620, FTS 443-7620) for access approval

2.17 NRC Reactor Safety Data Bank

Sponsor: USNRC-RES

Type of source: Computerized data base

Industry: U.S. and foreign nuclear (primarily test facilities)

Number and type of records: 1000 magnetic tapes with data from tests, primarily showing how measured parameters changed with time during experiments

Frequency of record update: As data and funding are available. A bulletin describing data base updates is issued approximately five times per year

Data source boundary: Nuclear reactor test facilities and two U.S. commercial nuclear power plants that will contribute to the overall water reactor safety research objective of safe, acceptable nuclear power production. The projects or facilities that supply data bank information are listed below

<u>Project/Facility</u>	<u>Description or Location</u>
Browns Ferry Nuclear Plant	United States
FEBA	West Germany
FIST	Fully Integrated System Test (INEL and GE)
FLECHT	Full-Length Emergency Core Heater Transfer (Westinghouse)
G-2	Westinghouse
GERDA	West Germany
GOETA	Sweden
HALDEN	Norway
JAERI	Japan Atomic Energy Research Institute
KUOSHENG	Taiwan
LOFT	Loss-of-Fluid Test (INEL)
LTSF	LOFT Technical Support Facility (INEL)
MARVIKEN	Sweden
MRBT	Multi-Rod Burst Test (ORNL)
NEPTUNE	Sweden
NRU	Canada
OTIS	Once-Through Integral System (B&W)
PBF	Power Burst Facility (INEL)
Peach Bottom Nuclear Plant	United States
ROSA	Rig of Safety Apparatus (JAERI)
SEMISCALE	INEL
SHBF	Single Heated Bundle Facility (GE)
SSTF	Steam Sector Test Facility (GE)
THTF	Thermal Hydraulic Test Facility (ORNL)
TLTA	Twin Loop Test Apparatus (GE)
U of OTTAWA	Canada

Time frame: 1976 to present

2.17.1 Description

The USNRC Division of Reactor Systems Safety (DRSS) maintains a Reactor Safety Data Bank Program administered by EG&G Idaho, Inc. It provides a

central computer-based storage mechanism and access software for data from 26 foreign and domestic water reactor safety research programs. The User's Manual⁵⁷ provides an introduction to the data bank and its capabilities, and guides the general user through some of the data storage, retrieval, and graphics functions that are available. The data will be used by code development, code assessment, and experimentation groups in meeting the overall objective of increased nuclear safety.

In general, only mechanical systems are considered although thermal-hydraulic and nuclear fuel handling data are also entered. The operating environment is within the component design basis and includes transients during startup of commercial plants. Each facility is responsible for its own data qualification procedures. The USNRC and the data bank administration are notified about any data errors, and the user community is informed about resulting corrections. The data covers information from 1976 to present, comprising nearly 1000 individual component and system tests, and is stored on 1000 digital magnetic tapes. The data are described in the form of abstracts of the individual tests. Failure rate values are not provided, but can be derived from event counts, populations, and operating profiles. Maintenance and repair data, as well as recovery actions, are not considered. However, the raw data can be retrieved from the data storage at the individual test facilities.

The Reactor Safety Data Bank is maintained at the INEL on its CDC Cyber 176 computer (NOS Operating System) as part of the INEL Scientific Data Management System (ISDMS). The data base can be accessed with permission of the USNRC DRSS. Less than 10% of the the data is confidential, depending on the disclosure agreement with the participating data source. There is a nominal user charge to cover the cost of processing a search request and the corresponding software time. Real-time access is available through terminal interfacing. Access to the data bank in ISDMS enables a user to store, select, discriminate, manipulate, and plot data as a function of time or against other data. New data and data management

software are continually being added to the data bank. However, information in a form directly applicable to reliability and risk analysis is not provided.

2.17.2 Data Source Access

Contact: E. Thomas Laats

Address: EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

Phone: 208-526-9507 (FTS 583-9507)

Report ordering address: GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

Computer access: See contact above

2.18 Initiating Events

Sponsor: USNRC-RES

Type of source: Report

Industry: U.S. commercial nuclear

Number and type of records: Over 5400 events representing transient initiating events causing scram as reported in an earlier EPRI report and in the USNRC's Operating Unit Status Reports

Frequency of record update: None

Data source boundary: Events causing scrams

Time frame: All operating plant data through December 1983

2.18.1 Description

The report⁴³ documents the development of transient initiating event frequencies for events causing scrams in nuclear power plants. This study is based on nuclear power plant operational data. Existing data published by EPRI (EPRI NP-2230)¹⁸ and information from the Nuclear Regulatory Commission's Operating Unit Status Reports⁵⁸ are combined to produce a comprehensive statistical data base and derive initiating event frequencies for use in PRAs of nuclear power plants. This new data base describes operating experience at 24 more plants than the NP-2230 study (76 plants in total), and covers at least three additional years of data for all the plants (thus covering events through December 1983). In all, the new data base has over 5400 events, compared with approximately 3000 events in the NP-2230 study.

The validity of the EPRI operational data, the adequacy of the transient event categories used by EPRI, and the EPRI data analysis assumptions and methodology are investigated. New transient initiating event frequencies with upper bounds are presented, based on data from all operating plants through 1983. More detailed transient event categories and a more rigorous data analysis methodology are developed and suggested to allow further refinement of the transient initiating event frequencies derived from the new, larger data base.

The report includes tables that contain event-related results from the combined data base. For each of the nearly 80 transient categories, a table is provided showing plant-specific event counts as a function of plant age in years. Other tables show, for PWRs and for BWRs, the total number of transients in the categories, the average transient frequencies, their upper bounds, the average outage time for each type of transient, and the ranking of the categories in descending order by number of occurrences. Additional analyses explore the impact of the outages on plant operations and the effect of power levels and scheduled scrams on transient event frequencies. A table provides plant-specific reactor critical hours from 1961 to 1983 by

calendar year to support analysis of time trends in the overall transient rates. A microfiche appendix lists one-line records of all the events tabulated by EG&G Idaho. These records, sorted both by plant and event date and by transient category, provide the reactor status before and after each event, the reactor power level, the outage length estimate, whether the scram was scheduled, and a brief event description.

A magnetic tape of the data base exists. The tape contains data from EPRI and EG&G Idaho. However, the data obtained from EPRI was from responses to a questionnaire sent to the plants and may be proprietary. The "PLUNGE" software obtained from EPRI along with the data was updated only as required to support the tables generated in the report. Thus, no support software or computerized data base version of the data is available.

2.18.2 Data Source Access

Contact:	David Mackowiak
Address:	EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, ID 83415
Phone:	208-526-4382 (FTS 583-4382)
Report ordering address:	GPO Sales Program Division of Technical Information and Document Control U.S. Nuclear Regulatory Commission Washington, D.C. 20555
Report accessibility:	No restrictions
Computer access:	None

2.19 In-Plant Reliability Data System

Sponsor:	USNRC-RES
Type of source:	Reports/computerized data base
Industry:	U.S. commercial nuclear

Number and type of records: Records describe component populations and failures, as follows:

	<u>Population</u>	<u>Failures</u>	<u>Unit Years</u>
Pumps	1468	3998	27
Valves	24825	5712	24
Electrical	213	698	33

Repair action records also exist in some cases

Frequency of record update: None

Data source boundary: Maintenance work orders describing corrective maintenance at units as follows:

Pumps and valves	two PWR, four BWR
Electrical	five PWR, four BWR

Time frame:	Pumps	1975 through 1982
	Valves	1975 through 1983
	Electrical	1978 through 1984

2.19.1 Description

The main objective of the In-Plant Reliability Data System (IPRDS) is to develop a comprehensive and component-specific data base for PRA and other component reliability-related statistical analysis.⁵⁹ Data base personnel visited selected plants and copied all the plant maintenance work requests. They also gathered plant equipment lists and plant drawings and in some cases interviewed plant personnel for information on component populations and duty cycles. They subsequently screened the maintenance records to separate out the cases of corrective maintenance applying to particular components; these were reviewed to determine such things as failure modes, severity, and, if possible, failure cause. The data from these reports were encoded into a computerized data base. By having a team of data base experts process all the data, problems with differences in the interpretation of reporting requirements were minimized. Using in-plant records provides assurance that even disabling incipient hardware problems are included. Thus, the resulting data base has consistency and depth.

Three reports have been issued containing IPRDS failure data. Information on pumps,⁴⁷ valves,⁴⁸ and major components in NPP electrical

distribution systems⁶⁰ has been encoded and analyzed. All three reports provide introductions to the IPRDS, explain failure data calculations, discuss the type of failure data in the data base, and summarize the findings. They all contain comprehensive breakdowns of failure rates by failure modes with the results compared with WASH-1400 and the corresponding LER summaries. In addition, information on failure causes and repair time distributions are presented. Statistical tables and plant-specific data are found in the appendixes. Additional details particular to each report and not contained in the tabular summary above are discussed below.

The pump report includes estimates of annual demands and duty cycle for each pump. Failure rate probabilities and maintenance frequencies are also calculated. Repair times are included for one PWR unit. Aggregated failure rate calculations are shown for 14 selected functional pump types (primarily safety related).

In the valve report, a sample of statistics on valves from one PWR and one BWR plant are developed to illustrate the degree of refinement possible when using IPRDS. In addition to failure rates and repair time distributions, failure, population, demand, and service-hour data for a full list of valve types are provided and can be used to calculate failure rates for particular data needs. An appendix provides a short study of safety relief valves. The major observation in this report is that the preliminary results obtained from the data base indicate WASH-1400 statistics may be nonconservative for reliability estimates for some valve types in certain failure modes. Because of the size of the data base, however, conclusive results are not possible.

The electrical component report presents data for DGs, batteries, battery chargers, and inverters. Only components in the essential ac power systems have been considered. Batteries, chargers, and inverters below the 120-V level have been excluded. Inclusion of certain dedicated electrical components (inverters, batteries, etc.) that are found in some safety-related systems is beyond the scope of this report.

The IPRDS data base is very helpful for corrective maintenance frequencies and repair times. However, further information on component duty cycles and use would improve the demand-based failure rate estimates contained in the reports. In addition, because the data base was developed from only four nuclear power stations, caution should be used for other than generic applications.

The computerized IPRDS data base is maintained at ORNL on an IBM 3033. The records are structured for interface with the Statistical Analysis System (SAS). However, many of the original data records are proprietary; in some cases agreements with the power plant licensees to obtain the maintenance records preclude release of the identity of the nuclear units. To obtain sorted listings of these records, one needs permission of the IEEE Office of Standards, which provides assistance in data handling and storage. Joel Buchanan (615-574-0377) should be contacted regarding access to the data base.

2.19.2 Data Source Access

Contact:	Raymond J. Borkowski
Address:	Martin Marietta Oak Ridge National Laboratory Building 9104-1 P.O. Box Y Oak Ridge, TN 37831
Phone:	615-574-0376 (FTS 624-0376)
Report ordering address:	GPO Sales Program Division of Technical Information and Document Control U.S. Nuclear Regulatory Commission Washington, D.C. 20555
Report cost:	Unknown
Report accessibility:	No restrictions
Computer access:	For inquiries about data listings, contact Joel Buchanan at ORNL (615-574-0377)

2.20 IEEE Standard 500-1984

Sponsor:	The Institute of Electrical and Electronic Engineers, Inc. (IEEE)
Type of source:	Report (book)
Industry:	U.S. commercial nuclear power
Number and type of records:	Over 1000 data sheets, each containing failure rate estimates and upper and lower bounds for various failure modes for a specified type of component. Where available, repair times are also included
Frequency of record update:	The standard is reviewed every five years
Data source boundary:	Electrical, instrumentation, and mechanical components in nuclear power plants
Time frame:	Varies, see text below

2.20.1 Description

The IEEE Standard 500 document was originally issued in 1977 as a manual of reliability data on electrical, electronic, and sensing components.⁶¹ Since that time, data have been gathered on mechanical equipment reliability, making the 1984 version of IEEE Std 500 a more comprehensive work by including data on over 1,000 components.⁴¹ The primary source for the reliability data presented in IEEE Std 500-1977 was the judgment of nearly 200 experts on component reliability in the nuclear power field. The Delphi method was used to compile and refine the expert opinion data. This information is included in IEEE Std 500-1984 for the nonmechanical components, but it is supplemented by data from 25 additional sources. These include nuclear industry data from published reports such as LER, NPRDS, and IPRDS data summaries and various manufacturers, utilities, and plants as well as component data from U.S. military sources. Therefore, the data developed represent either recorded data or the best collective judgment of a group of specialists. In each case, the source of the data is clearly identified. IEEE Std 500-1984 contains little unique data.

The IEEE Std 500 document is based on a hierarchical structure of component types set down in the manual's table of contents. The preface for each subsection (defined by a component type) provides a tree diagram that clearly shows the way the component classes have been subdivided to determine "data cells." The failure modes for each component class are also hierarchically organized according to failure severity: catastrophic, degraded, or incipient. Rates per hour and demand rates (per cycle) are both included, as well as upper and lower bounds.

The Std 500-1984 reliability data for electrical and electronic components are useful for PRA applications because this is one of the few sources that exists for many of the components it covers. In some cases, repair time data are provided for application toward component unavailability calculations. However, users of these data should be aware of the original sources used and the fact that some of them may overlap (a single failure may be reflected both in LERs and IPRDS, for example). There is also overlap between design-based groupings and application-based groupings of the data. For example, the pump data are primarily structured at a high level by the pump's internal method of operation (centrifugal, positive displacement) rather than by type of driver (turbine, diesel), but data sheets with pump and driver both are included and some data sheets have this information at a system-specific level. The amount of actual operational data supporting the rate estimates should be evaluated from a knowledge of the sources used by IEEE.

2.20.2 Data Source Access

Contact:	Lewie E. Booth
Address:	Fluor Engineers and Constructors, Inc. 2801 Kelvin Avenue Irvine, CA 92714
Phone:	714-966-5856
Report ordering address:	IEEE Service Center 445 Hoes Lane Piscataway, NJ 08854

Report cost: IEEE Member--\$121.50
Nonmember--\$135.00

Report accessibility: No restrictions

2.21 Licensed Operating Reactors Status (Grey Book)

Sponsor: USNRC's Office of Resource Management
Type of source: Periodic report
Industry: U.S. commercial nuclear
Number and type of records: Operational history of U.S. commercial reactors
Frequency of record update: Monthly
Data source boundary: U.S. commercial nuclear reactors
Time frame: 1974 to present

2.21.1 Description

The Licensed Operating Reactors Status Summary Report (Grey Book)⁶² is published monthly by the U.S. Nuclear Regulatory Commission's Office of Resource Management. The report covers all U.S. licensed power reactors and deals mainly with the availability of plants licensed for power generation.

The Grey Book is divided into three sections. The first section contains overall monthly statistics on power production for all the reactors covered. The second section is composed of facility data for each plant. The data include general information about the reactor, an inspection report, reports from the licensee during the month, and operating status reports from the plant. The final section includes general information about spent fuel storage capability, reactor-years of operation, and nonpower (experimental) reactors in the United States.

The Grey Book contains a good deal of information directly applicable to event analysis. Plots show the average daily power output (in percent)

for each unit for each month. A brief explanation of each major power reduction, including each shutdown, is provided. Since the data are not statistically reduced, an analyst cannot directly use the information for reliability and risk analysis. However, the Grey Book gives a good overall picture of how efficient a nuclear plant actually is in terms of power production and can be used to check other sources of event data.

Annual summaries of Grey Book information were issued as NUREGs covering the years from 1973 to 1980.⁶³

Grey Book data are maintained on magnetic tapes; copies of these tapes are obtainable. However, the data are not maintained in a computerized data base; no manipulation capabilities exist.

2.21.2 Data Source Access

Contact:	R. A. Hartfield
Address:	USNRC M/S MNBB-7602 Washington, D.C. 20555
Phone:	301-492-7834 (FTS 492-7834)
Report ordering address:	For a year's subscription: GPO Sales Program Division of Technical Information and Document Control U.S. Nuclear Regulatory Commission Washington, D.C. 20555 For single copies: National Technical Information Service Springfield, VA 22161
Report cost:	Unknown
Report accessibility:	No restrictions

2.22 Licensee Event Report Compilation (NUREG/CR-2000)

Sponsor:	USNRC-AE0D
Type of source:	Report

Industry: U.S. commercial nuclear

Number and type of records: There are approximately 350 LER abstracts in each monthly report

Frequency of record update: Monthly

Data source boundary: Event data from all U.S. commercial nuclear power plants licensed for producing power, in accordance with 10 CFR 50.73. Pre-1984, USNRC Regulatory Guide 1.16 and Licensee's Technical Specifications (NUREG-0161)

Time frame: 1982 to present

2.22.1 Description

This monthly report⁶⁴ contains LER operational information that was processed into the NSIC LER data file during a given month. The LERs, from which this information is derived, are submitted to the USNRC by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring before 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports.⁶⁵ For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in 10 CFR 50.73--Licensee Event Report System which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022,⁶⁶ Licensee Event Report System--Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.^a

The LER summaries in this report consist of the event titles and abstracts provided by the licensees. They are arranged alphabetically by

a. Supplement No. 2 to NUREG-1022 (Licensee Event Report System--Evaluation of First Year Results, and Recommendations for Improvements, September, 1985) provides further guidance.

facility name and chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System (See Section 2.33).

2.22.2 Data Source Access

Contact: Gary T. Mays
Address: Nuclear Safety Information Center
Oak Ridge National Laboratory
P.O. Box Y
Oak Ridge, TN 37831
Phone: 615-574-0391 (FTS 624-0391)
Report ordering address: Superintendent of Documents
U.S. Government Printing Office
P.O. Box 37082
Washington, D.C. 20013-7082
Report cost: \$37.00/annual subscription
Report accessibility: No restrictions

2.23 Maintenance Data Base

Sponsor: USNRC-NRR
Type of source: Periodic report/computerized data base
Industry: U.S. commercial nuclear
Number and type of records: 11,000 records containing plant background information, maintenance performance information, LERs, and '766' violations
Frequency of record update: Variable
Data source boundary: All U.S. commercial NPP; reports as listed below
Time frame: January 1980 through December 1985

2.23.1 Description

The Maintenance Data Base (MDB) contains data on equipment maintenance performance at nuclear plants. The MDB is maintained by the USNRC Division of Human Factors Technology. Six types of information are stored for each plant in the MDB:

1. Background information about the plant
2. Plant performance on a yearly basis
3. Detailed information on plant outages
4. The plant's systematic assessment of licensee performance (SALP) ratings
5. Summaries of LERs for the plants
6. Summaries of '766' Violations⁶⁷ investigated by the USNRC

The MDB includes over 100,000 pieces of information stored in dBASE III data base files. The data can be accessed through two methods: dBASE III commands can be used to retrieve and analyze the data, or a MDB menu system can be used. The menu system consists of a group of programs, each of which accomplishes one or more predefined data retrieval tasks. A comprehensive discussion of the contents and structure of the MDB is provided in the MDB Programmer's Guide.⁶⁸

Periodic reports containing an evaluation of the information stored in the MDB are anticipated. One report, NUREG/CR-4611,⁶⁹ includes a trends and patterns analysis of maintenance performance during 1980-1985.

The MDB data base is maintained on a PC. A 10-megabyte hard disk and dBASE III are required to use the data base. The data management system can be used to generate special output reports. Access to the data base and floppy disk copies of the data base can be obtained from Peter McLaughlin.

2.23.2 Data Source Access

Contact: Peter McLaughlin

Address: USNRC
Washington, D.C. 20555

Phone: 301-492-4904 (FTS 492-4904)

Report ordering address: GPO Sales Program
Division of Technical Information and Document
Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

Computer access: See contact above

2.24 Nuclear Plant Reliability Data System

Sponsor: Institute of Nuclear Power Operations (INPO).

Type of source: Reports/computerized data base

Industry: U.S. commercial nuclear

Number and type of records: Engineering data from all U.S. NPP (currently 91 plants) with approximately 30 major systems per plant and 100 mechanical and electro-mechanical components per system; failure data currently describing 44,000 events

Frequency of record update: Daily

Data source boundary: Voluntary reporting of engineering information and failures for selected systems and components as defined in a reportable scope manual

Time frame: 1974 to present

2.24.1 Description

The NPRDS is an industry-wide system for monitoring the performance of selected systems and components at U.S. commercial nuclear power plants.⁷⁰⁻⁷⁵ INPO has been responsible for this data base since January of 1982; previously, Southwest Research Institute was its sponsor.

Information in NPRDS is derived from a standardized format input report prepared by U.S. nuclear plant licensees. The plants are asked to submit failure reports on catastrophic events and degraded failures within the defined reportable scope; reporting of incipient events is optional. Catastrophic and degraded failure events are events that involve some loss of component function for which repair is required. Command faults are not reportable unless they make an entire system unavailable.

In addition, the plants are asked to file component engineering reports on all components within the selected systems and reportable scope. These reports contain detailed design data, operating characteristics, and performance data on the selected systems and components (over 3000 components, from approximately 30 systems, per unit). The selected systems are primarily safety systems.

NPRDS data are available to users, either through annual summary reports^{76,77} (to resume publication in the coming year) or through direct on-line data base access from a computer terminal. Special reports and listings are available through requests for extraction or data analysis. A requester is required to define the analyses to be performed.

With the engineering data, component and system failure statistics can be derived from the data base. The reliability and failure data are useful to utility and plant staff, designers, architect-engineers, NSSS suppliers, construction firms, regulatory agencies, and PRA practitioners. Although the reporting period covers approximately 12 years, the most detailed reporting has been implemented by most of the licensees since 1984 when the new LER rule came into effect and fewer component failures were reported to the LER system. Also, INPO has actively encouraged data base participation, particularly since 1984.

This data base represents an extensive, somewhat unique compilation of operating data from all U.S. commercial nuclear plants. With real-time data base access, a spectrum of output products can be generated. Component and

system failure rates can be derived to generate specific output reports. One should note that no human-error-related data are included in the data bank except for actions that cause broken hardware. By nature, the degree of reliance on statistical data generated from NPRDS data is dependent on the consistency and completeness of reporting by individual contributors. A detailed procedures manual and other user oriented material are available to NPRD system users.

The NPRDS is maintained at INPO on a Prime 750 computer. The data base is currently being converted to an IBM 3083. NPRDS data can be obtained in the form of magnetic tapes; also, the data can be downloaded onto floppy disks that are dBASE III or LOTUS 1-2-3 compatible. Real-time access is available with documented procedures for record viewing, sorting, and retrieval.

The NPRDS data are proprietary. Access is permitted only to INPO members, participants, and the USNRC. Access is free to INPO members and participants. USNRC access costs \$145 per connect hour.

2.24.2 Data Source Access

Contact:	Ron Simard
Address:	Institute for Nuclear Power Operations 1100 Circle 75 Parkway, Suite 1500 Atlanta, GA 30339-3064
Phone:	404-953-5337
Report ordering address:	Same as above For annual summaries of data base for 1978-1980: GPO Sales Program Division of Technical Information and Document Control U.S. Nuclear Regulatory Commission Washington, D.C. 20555
Report cost:	Unknown
Cost exemptions:	Free to INPO members/participants

Report accessibility: Plant-specific NPRD data are proprietary. The availability of information depends both on the requester and the specific information desired. Summary reports with component and system data are readily available

Computer access: Limited, as with report access

2.25 Nuclear Power Experience

Sponsor: The S.M. Stoller Corporation (SMSC)

Type of source: Report/computerized data base

Industry: All U.S. and some foreign commercial nuclear power

Number and type of records: More than 50,000 events indexed by plant, system, component categories and keywords

Frequency of record update: Monthly

Data source boundary: NPE data are compiled from sources available in the public domain that list events with safety, generic, or outage-causing significance

Time frame: 1957 to present

2.25.1 Description

The purpose of the Nuclear Power Experience (NPE) data base is to record and index information for engineering analysis applications on operating event occurrences at large U.S. PWRs and BWRs and selected foreign facilities from startup to the end of commercial operation.⁴⁵ The information is supplied in two ways: a) a multi-volume loose-leaf subscription service that provides hard copies of narrative event reports, plant operating histories, plant descriptions, and plant availability tables, and b) an automated retrieval system (NPEars) that provides a fully automated, on-line search capability for the entire NPE article base. NPE excludes minor events relating to missed tests, instrument drifts (unless a generic fix is indicated), technical specifications, and other similar

criteria. The NPE is indexed by plant, system, component categories, and by keywords.

The NPE is a good source of event descriptions and is useful as a cross-reference for individual event data derived from other sources. Because NPE and NPEars are geared to written narrative reports, statistical data must be compiled by system users. This can be fairly difficult, because population information cannot be easily derived from this source. The keyword indexing system is not entirely adequate, because, in a very complicated event, not every component of interest is always indexed.

The NPEars is proprietary to SMCS, but is available to NPE subscribers as part of the service (the two cannot be acquired separately). The data base is maintained on a VAX11/780 with a VMS Version 4.3 operating system. Currently terminal access (with modem) is through the Telenet communication system. The data base retrieval system provides a user with a standard menu of search options. The user may also build search criteria based on key word selections.

2.25.2 Data Source Access

Contact:	James F. Franks
Address:	The S. M. Stoller Corporation 1919 14th Street, Suite 500 Boulder, CO 80302
Phone:	303-449-7220
Report ordering address:	Same as above
Report cost:	\$2,172 annual subscription cost for NPE and NPEars
Cost Exemptions:	None
Computer access:	See contact above

2.26 Nuclear Safety Information Center Licensee Event Report Data Base

Sponsor:	USNRC-AEOD
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Type of source:	Computerized data base
Industry:	U.S. commercial nuclear
Number and type of records:	50,000 to 60,000 records containing LER abstracts and a keyword listing
Frequency of record update:	Monthly
Data source boundary:	LERs; see the LER compilation section
Time frame:	1960 to present

2.26.1 Description

LERs are submitted to the USNRC by U.S. utilities having licensed nuclear plants. Pre-1984 LERs contain information about events reportable under the technical specifications of the various plants.⁶⁵ LERs submitted after 1984 follow the reporting requirements of 10 CFR 50.73⁷⁸ and the Licensee Event Report System.²⁷ LERs consist of a narrative description and coded information (e.g., component, manufacturer); they generally cover failures of safety related systems or components. Since 1984, their emphasis is on unusual events.

The purpose of the National Safety Information Center (NSIC) LER Data Base is to provide a keyworded data file for retrieval of various information (e.g. components, system, type of failure). LER data from 1960 to the present are maintained in a 50,000- to 60,000-event data file by NSIC at ORNL under contract to the USNRC-AEOD. The data base provides users with an abstract of the LER and keywords so various events can be found in computer searches. The data can be accessed on a cost-recovery basis. Monthly abstract listings with a keyword index are issued by subscription (see Section 2.22; currently, information for these monthly compilations comes from the LER Sequence Coding and Search System (see Section 2.33)).

The NSIC LER data base is the primary reference for older (pre-1980) LERs. However, the data file contains no statistically reduced data, so

users must derive failure rates. Gross failure rate estimates for various components have been calculated from LER data; see Section 2.16.

The NSIC LER data base has been maintained on an IBM-3033 with an MVS Operating System. Subsets of the data base are obtainable on tape or on PC disks. Computer charges are \$300 an hour central processing unit (CPU) time and \$4 an hour connect time. These charges are based on ORNL maintaining a cost-recovery status. There are no exemptions from these charges. To assist users who do not wish to use on-line retrieval, NSIC will provide special searches and report production on an ad hoc basis.

2.26.2 Data Source Access

Contact: W. P. Poore
Address: Nuclear Safety Information Center
Oak Ridge National Laboratory
P.O. Box Y
Oak Ridge, TN 37831
Phone: 615-574-0325 (FTS 624-0325)
Computer access: See contact above

2.27 Operating Plant Evaluation Code (OPEC)-2

Sponsor: The S. M. Stoller Corporation
Type of source: Periodic reports/computerized data base
Industry: U.S. commercial nuclear
Number and type of records: 67,000 events (operating experiences) of commercial U.S. nuclear power plants
Frequency of record update: Quarterly
Data source boundary: Each event at a greater than 400 MW(e) U.S. commercial nuclear power plant that fits into one or more of the following categories is recorded:

1. All events that result in an outage (full or partial)

2. All events which contribute to the length of an outage (on the critical path)
3. All significant maintenance or repair work on major components. Any work that requires full or partial disassembly of the component is considered to be significant. Major components consist of the turbine, generator, control rod drives, reactor coolant pumps, steam generators, etc.
4. The 1978, 1979, and part of year 1980 data also includes noncurtailing events that resulted in either a safety-system actuation, the failure or performance degradation (beyond technical specification limits) of a safety component, or significant work on a safety-system component

Time frame: 1968 to present

2.27.1 Description

The Operating Plant Evaluation Code, Rev. 2 (OPEC-2) computer program and data base⁷⁹ provide a comprehensive and organized compilation of operating experience of modern commercial U.S. nuclear power plants. The data base includes complete operating data for all nuclear plants in the United States that are larger than 400 MW(e). Data for each plant extend from commercial operation to within approximately six months of the present date. Data consist of overall plant performance indices, the cause and magnitude of each plant shutdown or power reduction, the cause and duration of curtailing work on major components, and the nature of each plant transient experienced. For calendar years 1978, 1979, and part of 1980, the data base includes the cause of each problem with safety-related equipment that resulted in an LER. The OPEC-2 computer program provides an efficient means of storing, sorting, and analyzing this data. Analyses can be performed to determine failure rates and repair times for selected equipment and to determine the effects that selected problems or groups of problems have had on the availability and capacity factors of selected groups of plants.

The primary source of data for OPEC-2 is the plant operating reports that are periodically submitted to the USNRC. Generally these reports contain extensive and informative verbal chronologies of plant operating history. The NRC Grey Books (see Section 2.21) are used to supplement data from the plant operating reports. Data from these two sources are supplemented by information from technical papers, equipment vendors, and direct communications with plant operators. LER descriptions and lists of safety-related maintenance in plant operating reports have also been major sources of information for the more recent data.

The OPEC-2 software provides a means for sorting the data base and generating preformatted output reports. The four basic output reports are: unit performance statistics, problem component statistics, events listing all information, and events listing selected information. The reports include results of analysis on the specified data set (i.e., failure rates, repair times, trends, outages).

OPEC-2 is available through a nonexclusive licensing arrangement with SMSC. The system is proprietary to SMSC. The OPEC-2 data base and software system can be operated on the licensee's own machine using FORTRAN 5 on a VAX, CDC, IBM mainframe, etc. Alternatively, a data tape can be downloaded to a PC disk or accessed at SMSC using a time-sharing and telephone link arrangement.

2.27.2 Data Source Access

Contact:	Eric Olson
Address:	The S. M. Stoller Corporation 1919 14th Street, Suite 500 Boulder, CO 80302
Phone:	303-449-7220
Report ordering address:	Same as above
Report cost:	Variable
User fee:	Variable

Cost exemptions: None
Computer access: See contact above

2.28 Pipe Break Frequency

Sponsor: USNRC-RES
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: Nineteen occurrences of pipe failures (breaks), supplemented by expert-opinion estimates of frequencies of safety significant pipe breaks in commercial U.S. nuclear power plants
Frequency of record update: None
Data source boundary: Source covers all breaks in U.S. commercial nuclear power plant piping (81 plants) having a leak rate of at least 1 gpm for pipes at least 2 inches in diameter and all leak rates of 50 gpm or more regardless of pipe size
Time frame: The operational history of power plants through December 1984

2.28.1 Description

This study⁸⁰ empirically develops frequencies and bounds for safety-significant pipe failures in commercial NPPs. Its purpose is to update the pipe break frequencies reported in the Reactor Safety Study (WASH-1400),⁴⁰ which are used in many risk analyses. The study involved reviewing various data sources for actual piping failure events of significant magnitude. The data sources reviewed were LERs, NPE, and several other sources documented in the report. When extant in the documentation reviewed, information was extracted concerning conditional factors such as the system in which the failure occurred, the operational mode of the plant, and the size of the pipe involved to permit estimation of conditional pipe break frequencies useful to risk analysts.

Because there have been few significant pipe failures, the sparse real data was supplemented with expert-opinion data. The report presents the results of combining the real and subjective data through Bayesian statistical methods. That is, posterior probabilities of given failure rates were determined and are presented in the report.

The rates of pipe failures are also analyzed to determine whether or not the rates are dependent on the system under consideration, the operational mode of the plant, the size of the pipe, or other factors. In statistical terms, an analysis of variance assessment was made on the rates of pipe failure.

2.28.2 Data Source Access

Contact:	Ronald E. Wright
Address:	EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, ID 83415
Phone:	208-526-9467 (FTS 583-9467)
Report ordering address:	Same as above
Report cost:	Unknown
Report accessibility:	No restrictions

2.29 Reactor Safety Study (WASH-1400)

Sponsor:	USNRC
Type of source:	Report
Industry:	U.S. commercial nuclear
Number and type of records:	Not applicable
Frequency of record update:	None
Data source boundary:	Numerous sources such as Systems Reliability Service (United Kingdom), Failure Rate Data Handbooks (U.S. military and NASA), LMEC, NASA reports, General Electric Company reports, and Defense Documentation Center reports. Failure

data were gathered from eight PWRs and nine BWRs for 1972

Time frame: Varies; all prior to 1975

2.29.1 Description

The Reactor Safety Study (WASH-1400)⁴⁰ was published by the USNRC in 1975 to set down a methodology for assessing nuclear plant reliability and risk. Of particular interest to the data analyst are Appendix III, "Failure Data," and Appendix IV, "Common Mode Failures."

Appendix III contains failure rate estimates for various generic types of mechanical and electrical equipment. Included are listings of failure rates with range estimates for specified component failure modes, demand probabilities, and times to maintain or repair. It also contains some discussion on such special topics as human errors, aircraft crash probabilities, loss of electric power, and pipe breaks. Appendix III contains a great deal of general information of use to analysts on the methodology of data assessment for PRA.

Appendix IV contains a thorough discussion of quantification techniques and engineering studies of common mode failures. Large LOCA, small LOCA, and transient sequences are considered.

WASH-1400 is a fundamental document for PRA methodology. The data appendixes contain a great deal of useful information on methods of data assessment. A large number of sources for data are considered, and very general failure rate estimates for nuclear power plant PRAs are presented. Studies using these estimates will produce only gross approximations. Since the advent of data collection schemes across and within plants, the WASH-1400 data are solely useful as a constituent to a data aggregation process or as widely bounded figures that provide a basis for comparison.

2.29.2 Data Source Access

Report ordering address: National Technical Information Service

Springfield, VA 22161

Report cost: \$68 for all volumes

Report accessibility: No restrictions

2.30 Reactor Coolant Pump Seal Failures

Sponsor: USNRC-NRR

Type of source: Report

Industry: U.S. commercial nuclear

Number and type of records: 200 events describing reactor coolant pump (RCP) failures

Data source boundary: RCP failures captured by LER and NPRDS reporting at Arkansas Nuclear Unit 1, Calvert Cliffs Unit 1, and Indian Point Unit 3 nuclear plants

Time frame: January 1969 through December 1984

2.30.1 Description

The report⁸¹ briefly describes a group of RCP seal failures that occurred at Arkansas Nuclear Unit 1, Calvert Cliffs Unit 1, and Indian Point Unit 3. Both mechanical and maintenance-induced RCP failures are discussed. For each event, the following information is provided: the date, the pump identification code, the nature of the failure, the maximum leakage per minute, and the total leakage in gallons. Data sources used as input were LERs, NPRDS, and final safety analysis reports (FSARs).

The report includes pedigree information on each plant. This includes plant name and unit number, type, vendor, number of pumps, pump designer, pump model number, and number of seal stages.

The report presents the findings made in analysis of the RCP failures. Estimates of the annual frequency for the spectrum of leak rates induced by RCP seal failures and their impact on plant safety (contribution to core-melt frequency) are made. The safety impact of smaller RCP seal leaks

was assessed qualitatively, whereas for leaks above the normal makeup capacity, formal PRA methodologies were applied. Also included are the life distribution of RCP seals and the conditional leak rate distributions, given a RCP seal failure; the contribution of various root causes; and estimates for the dependency factors and the failure intensity for the different combinations of pump designers and plant vendors.

2.30.2 Data Source Access

Contact: M. Ali Azarm

Address: Brookhaven National Laboratory
Department of Nuclear Energy
Upton, NY 11973

Phone: 516-282-4922 (FTS 666-4922)

Report ordering address: Same as above

Report cost: None

Report accessibility: No restrictions

Computer access: The raw records used to produce this report are maintained at Brookhaven National Laboratory. However, they are not readily available for external dissemination. Inquiries concerning these records should be made to M. Ali Azarm

2.31 Safety System Unavailability

Sponsor: USNRC-AEOD

Type of source: Report/computerized data base

Industry: U.S. commercial nuclear

Number and type of records: 1040 events that were actual or potential system safety unavailabilities

Frequency of record update: Annual

Data source boundary: Events recorded in accordance with LER reportability code 10 CFR 50.73(a)(2)(v)

Time frame: January 1984 through December 1985

2.31.1 Description

The data base is in the form of coded records describing actual or potential nuclear plant safety system unavailabilities. The records are derived by extracting information from LERs submitted in accordance with LER reportability code 10 CFR 50.73(a)(2)(v) (actual or potential loss of safety system function).

The data base was developed at the INEL for AEOD in support of its trends and patterns analysis of operational data program. The data base and supporting software are maintained on a PC using dBASE III PLUS. The content and structure of the data base are documented in a report.⁸² The report describes methods for developing the data base, data entry procedures, coding methods, and basic reports which can be produced from the data base. In addition, information is presented on the number of system unavailability events described in the data base and a summary of the total number of LERs that were screened for entry into the data base.

Access to the safety system unavailability data base requires a PC with dBASE III PLUS. A floppy disk of the data base is obtainable. Obtaining a copy of the data base or portions thereof requires AEOD approval.

2.31.2 Data Source Access

Contact:	Charles M. Landgraver
Address:	EG&G Idaho, Inc. P.O. Box 1625 Idaho Falls, Idaho 83415
Phone:	208-526-9295 (FTS 583-9295)
Report ordering address:	Same as above
Report cost:	Unknown
Report accessibility:	No restrictions
Computer access:	Contact Patrick O'Reilly, USNRC AEOD (FTS 492-8858)

2.32 Reportable Events File

Sponsor: USNRC

Type of source: Computerized data base

Industry: U.S. commercial nuclear

Number and type of records: Over 5700 records of events specified in 10 CFR paragraphs: 50.72 (significant events), 50.55 (construction permits), 73.71 (lost nuclear material), 20.402 (theft or loss of licensed material), and 20.403 (radiological incident notification)

Frequency of record update: Daily

Data source boundary: Events reportable in accordance with CFR paragraphs cited above

Time frame: July 1982 to present

2.32.1 Description

This source is a data base of event reports maintained and used by the NRC. The data base is maintained on NRC computers and also at the National Institutes of Health (NIH). Often this source is referred to as the "50.72 Data Base." The original reports that are coded into this computerized file are submitted by nuclear power plants in accordance with the reporting requirements set forth in the CFR paragraphs cited above.

The bulk of the reports are related to 10 CFR 50.72, which describes requirements for immediate notification of significant events at nuclear power plants. Significant events include the declaration of any of the emergency classes specified in a licensee's approved emergency plan, or any of a set of nonemergency events specified in 10 CFR 50.72 paragraph (b) (e.g., plant shutdown required by technical specifications, serious degrading of principal safety barriers).

The data base's content is as-reported, without elaboration and analysis; these reports sometimes prove to be an inaccurate description of

the event after all facts are known. Requirements for 50.72 reporting and LERs differ, so a one-to-one correspondence doesn't exist between the two. Where LERs are written, however, they are, by far, the more accurate of the two data sources.

The data base is structured so that users can search on various fields of the reports, but any "statistical" summaries would be invalid because the event reports are not updated to correct the initial statements. (Some update and tracking may be available in 1987 to cover the time period until an LER is available). Gross numbers of reports (per plant, region, emergency classification, for example) are frequently obtained to answer inquiries and support more quantitative analysis.

The reportable events report file is maintained on two separate computers, a Data General MV-6000 at the USNRC and an IBM 3081 at the NIH. Sorts can be performed on the data and then one can download the information to a disk for PC use. Access and cost information can be obtained from James Carter (see below).

2.32.2 Data Source Access

Contact:	James Carter
Address:	USNRC Office of Administration and Resource Management M/S 3302 Washington, D.C. 20555
Phone:	301-492-9860 (FTS 492-9860)
Computer access:	See contact above

2.33 Sequence Coding and Search System

Sponsor:	USNRC-AEOD
Type of source:	Computerized data base

Industry: U.S. commercial nuclear

Number and type of records: More than 21,000 LER abstracts and approximately 100,000 component/system event occurrence records

Frequency of record update: Weekly

Data source boundary: Event data from all U.S. commercial NPP, as required by 10 CFR 50.73 (Licensee Event Report System). Pre-1984, event reporting per USNRC Regulatory Guide 1.16 and Licensee's Technical Specifications requirements

Time frame: 1980 to present

2.33.1 Description

The Sequence Coding and Search System (SCSS) data base contains data from all LERs submitted by nuclear power plants after January 1, 1980. The data base is maintained at ORNL by the NSIC.

SCSS's primary objective is to reduce the descriptive text of LERs to a coded sequence of occurrences that is both computer readable and searchable. This system provides a structured format for detailed coding of faults for piece-parts, components, trains, systems, unit effects, and personnel errors. Information for hardware faults includes cause, system, interfacing system (if applicable), component vendors, performance (total or partial loss of function), detection method, and effect. Records describing personnel actions include the type of activity being done and whether the action was an omission or a commission. These individual records are linked in a network corresponding to the interrelationships among the individual occurrences that together form the reportable event. The data base also contains references to similar events for each applicable LER.

A four-volume report documents and describes SCSS in detail.⁸³ Volume 1 is a user's guide for searching the SCSS data base. Chapter 2 of this guide is a tutorial on retrieving, displaying, and analyzing LERs and provides hands-on experience in executing basic commands. Volume 2 contains all valid and acceptable codes used for searching and encoding the LER

data. Volumes 3 and 4 provide the information and methodology necessary to capture descriptive data from the LER and to codify that data into a structured format. These volumes serve as reference material for the more experienced technical processor and contain information essential to a more advanced user who needs to be familiar with the intricate coding techniques in order to retrieve specific details in a sequence.

The SCSS data base is maintained at ORNL on an IBM-3033 (MVS Operating System). The data base is set up through the JOSHUA DBMS, with software specifically written to manage the data base. On-line searches and sorts of SCSS are possible. Subsets of the data base can be downloaded to PC disks. Data base queries generally can be performed by ORNL personnel or one can obtain direct access to SCSS. Charges for use of SCSS are on a cost recovery basis; currently, CPU time is \$300 per hour and connect time is \$4 per hour. Access to the data base requires AEOD approval.

2.33.2 Data Source Access

Contact: Gary T. Mays
Address: Nuclear Safety Information Center
Oak Ridge National Laboratory
P.O. Box Y
Oak Ridge, TN 37831
Phone: 615-574-0391 (FTS 624-0391)
Computer access: Contact Eugenia Boyle of NRC-AEOD
(FTS 492-4498) for access permission

2.34 Snubber Performance

Sponsor: USNRC-IE
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: 400 occurrences of snubber failure at U.S. nuclear power plants
Frequency of record update: None

Data source boundary: Reporting scope of LERs, NPRDS, CFR 50.55(e) reports, and CFR Part 21 reports

Time frame: July 1980 through December 1984

2.34.1 Description

The report⁸⁴ contains a review of snubber operating experience at nuclear power plants from 1980 to 1984. Both hydraulic and mechanical snubber types are reviewed.

The report includes an evaluation of snubber performance; operational, installation-related, and manufacturing problems are identified. General failure data with failure modes and respective frequencies is provided.

Also included are a review of pertinent vendor activities and recommendations for effective reliability and overall nuclear plant safety.

2.34.2 Data Source Access

Contact: Monomohan Subudhi

Address: Brookhaven National Laboratory
Building 130
Upton, NY 11973

Phone: 516-282-2429 (FTS 666-2429)

Report ordering address: Same as above

Report cost: Unknown

Report accessibility: No restrictions

2.35 System Interaction Events

Sponsor: USNRC-NRR

Type of source: Report/computerized data base

Industry: U.S. commercial nuclear

Number and type of records: 235 events divided into 23 categories of occurrences of adverse system interaction

Frequency of record update: None

Data source boundary: Events selected are adverse system interaction occurrences (as defined by this study; see below)

Time frame: LER data from 1969 through January 1, 1984; also, additional data sources supplementing the LER data as they became available and of use

2.35.1 Description

The report⁸⁵ describes the first phase of a USNRC-sponsored project that identified and evaluated system interaction events that have occurred at commercial nuclear power plants in the United States. 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.

For each system interaction event, the following are provided: plant, date, plant operating status, interacting systems and components, and the systems and components affected. Additional descriptive information is also included.

The report is in two volumes. Volume 1 contains appendixes that review the data sources used in identifying events and outlines the information collected for each event. The primary data sources used to describe individual events were LERs, SCSS, and Construction Event Reports (CERs). In addition, because significant events often involve intersystem dependencies, a number of information sources on significant events were used to supplement these primary data sources. These additional data sources are documented in the report.

Volume 2 provides a description of each adverse system interaction event and lists the references for the events. The information in Volume 2 describing each adverse system interaction occurrence is maintained on an IBM computer. The set of events is also available on a PC disk in the form of an ASCII file.

2.35.2 Data Source Access

Contact: George A. Murphy

Address: Oak Ridge National Laboratory
Building 9201-3, MS-5
P.O. Box Y
Oak Ridge, TN 37831

Phone: 615-576-7053 (FTS 626-7053)

Report ordering address: GPO Sales Program
Division of Technical Information and Document
Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Report cost: Unknown

Report accessibility: No restrictions

Computer access: Contact Dale Thatcher (USNRC-NRR,
(FTS 492-8358)

2.36 Statistical Analysis of IPRDS Data

Sponsor: USNRC-RES

Type of source: Report

Industry: U.S. commercial nuclear

Number and type of records: All IPRDS data base records for the pumps and valves selected for analysis

Frequency of record update: None

Data source boundary: The set of valves and pumps selected for analysis from the IPRDS data base

Time frame: 1975 through 1983

2.36.1 Description

Los Alamos National Laboratory performed separate statistical analyses using the Failure Rate Analysis Code (FRAC)⁸⁶ on IPRDS failure data for pumps and valves. The major purpose of the studies was to determine which environmental, system, and operating factors adequately explain the variability in the failure data. The results of the pump study are documented in NUREG/CR-3650.⁸⁷ The valve study findings are presented in NUREG/CR-4217.⁸⁸

In the analysis of pumps, IPRDS failure data for 60 selected pumps at four nuclear power plants were statistically analyzed using FRAC. The data cover 23 functionally different pumps for each of two PWRs, and 21 and 17 functionally different pumps, respectively, for two BWRs. Catastrophic, degraded, and incipient failure severity categories were considered for both demand-related and time-dependent failures.

For catastrophic demand-related pump failures, the variability is explained by the following factors listed in their order of importance: system application, pump driver, operating mode, reactor type, pump type, and unidentified plant-specific influences. Quantitative failure rate adjustments are provided for the effects of these factors.

In the case of catastrophic time-dependent pump failures, the failure rate variability is explained by three factors: reactor type, pump driver, and unidentified plant-specific influences.

Point and confidence interval failure rate estimates are provided for each selected pump by considering the influential factors. Both types of estimates represent an improvement over the estimates computed exclusively from the data on each pump.

The coded IPRDS data used in the analysis is provided in an appendix.

A similar treatment applies to the valve data.

2.36.2 Data Source Access

Contact: Elizabeth Kelly
Address: Los Alamos National Laboratory
Group S-1, MSF 600
Los Alamos, NM 87545
Phone: 505-667-3308 (FTS 843-3308)
Report ordering address: GPO Sales Program
Division of Technical Information and Document
Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Report cost: Unknown
Report accessibility: No restrictions

2.37 Technical Specifications

Sponsor: USNRC-AEOD
Type of source: Report/computerized data base
Industry: U.S. commercial nuclear
Number and type of records: 3181 events representing occurrences of either
technical-specification violations or
technical-specification required shutdowns
Frequency of record update: Annually
Data source boundary: Events reportable per 10 CFR 50.73 (a)(2)(i)
Time frame: January 1984 through December 1986

2.37.1 Description

This source is a listing of incidents of technical-specification violations and of plant shutdowns required by technical specifications at each U.S. commercial NPP. Incident descriptions are extracted from LERs. The data base is maintained on a PC and was developed through use of dBASE

III PLUS. Software for report production is maintained at INEL. For each event, the data base contains fields describing: a) the impact on plant operation (outage hours), b) the system involved, c) the type of technical specification involved, d) the cause of the original problems, and e) the safety significance.

A complete description of the data base and its contents is provided in an EG&G informal report.⁸⁹ Copies of the data base are obtainable on PC disks. AEOD approval is required to obtain a copy of the data base.

2.37.2 Data Source Access

Contact: Carl Lovell
Address: EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415
Phone: 208-526-9459 (FTS 583-9459)
Report ordering address: Same as above
Report cost: Unknown
Report accessibility: No restrictions
Computer access: Contact G. L. Plumlee (USNRC-AEOD)
(FTS 492-4492)

2.38 Burns and Roe Valve Study

Sponsor: US DOE, Division of Nuclear Power Development
Type of source: Report
Industry: U.S. commercial nuclear
Number and type of records: 195 valve failures causing shutdowns or scrams reported in LERs in the period from December 1982 to March 1979, supplemented by records of all reported valve failures for 10 selected stations from February 1966 to January 1979
Frequency of record update: None
Data source boundary: Valve-related events reported in LERs, as

noted above

Time frame:

February 1966 through January 1979

2.38.1 Description

The study⁹⁰ performed by Burns and Roe (B&R) shows that valve failures constitute the component category most responsible for the shutdown of PWR and BWR plants. This investigation, contracted with SNL for DOE, identified the principal types and causes of valve failures that lead to plant trips for the period from December 1972 to December 1978. The primary source of data for the report was searches of the data base. Additional sources included the monthly Gray Books and Nuclear Power Experience publications, as well as discussions with utilities, valve manufacturers, and suppliers.

As the result of a cursory review of the NSIC/LER abstracts, the report investigates in greater detail the statistically most common types of valve failures. These valves include power-operated and spring-loaded relief, main steam isolation, feedwater regulator, pressurizer spray, and solenoid-operated pilot valves. Also considered are generic problems such as valve stem leakage, valve actuation malfunction, and special valves that do not require packing. Each subsection of the report discusses the system application of the valves. Typical system flow diagrams present outline and sectional drawings of the valve and typical installation arrangements. Regulatory and code requirements as well as design responsibilities are discussed. The report provides an analysis and statistics for each valve type summarizes the utilities' and manufacturers' experience. It also discusses possible solutions to the failure problems, the status of any ongoing corrective action and maintenance programs. The report concludes with a summary of recommendations that could lead to long-term solutions.

The B&R valve failure report is a comprehensive study of valve failure modes and the factors that contribute to their malfunction in nuclear power plants. Each valve is investigated based on operating experience as well as

discussions with utilities, manufacturers, and suppliers. This study is a good reference for the construction of fault/event trees of systems that are affected by valve performance. The valve failure modes are identified, the associated mechanisms are described in detail, and preventive measures are offered. However, there is a limited amount of quantification. The failure calculations do not present statistics on the number of demands, operating hours, or component populations. The number of failures are tallied, but rates are not contained in the report. The calculations of the study are based on the number of LER events reported by the plants. The report does not specifically reference the contacts with various contributors. Thus, it is difficult to ascertain the identity, content, and circumstances of those discussions. A reference list of the manufacturers, utilities, and suppliers that were contacted is not provided in the report. However, the B&R report does include a copy of the LER abstracts that were used as the primary data source.

2.38.2 Data Source Access

Contact:	National Technical Information Service Springfield, Virginia 22161
Report cost:	\$11.95
Report accessibility:	No restrictions

3. NONNUCLEAR DATA SOURCE DESCRIPTIONS AND ACCESS

Brief discussions of the fundamental features of selected nonnuclear data sources are provided in this section.^a The data sources are presented in alphabetical order based on their acronyms. A brief overview of each data source is provided. This is followed by information on the data source contents and its potential usefulness in nuclear applications. Finally, reference information for contacting the data source representative is provided.

3.1 Failure and Inventory Reporting System

The Failure and Inventory Reporting System (FIRS) program was developed by the Geological Survey Division of the U.S. Department of the Interior for safety and pollution prevention devices on offshore structures that produce or process hydrocarbons. The program contains data on mechanical and some electromechanical systems on offshore oil platforms. About 8,000 failure events are documented each year. Access has been limited to internal materials management system use. No real-time access or periodic output products have been available.

3.1.1 Contents

The system is composed of two programs. The safety device inventory reporting program provides information listing the number of safety and pollution devices by type, manufacturer, and model. The safety device failure reporting program provides information about the failures of these devices by failure causes, corrective actions, device type, manufacturer, model, and frequency of failure. Failure percentages, reliability and quality trends, mean time between repairs, and mean time between failures, along with other statistical information, are derived from these data.

a. This information is largely adopted from Reference 3, and is current as of late 1982.

3.1.2 Areas of Usefulness

The FIRS data could be applicable to nuclear plant reliability and risk assessments. However, as long as the data are unavailable to outside agencies, some administrative changes would be required to allow nuclear industry users access to the data.

Contact: L. E. Bennett
Minerals Management Services (OS-2)
U.S. Department of the Interior
P.O. Box 7944
Metairie, LA 70010

Telephone: 504-837-4720

3.2 Generating Availability Data System

The National Electric Reliability Council (NERC) Generating Availability Data System (GADS)⁹¹ covers cumulative information from 2,600 nuclear, fossil, hydro, and combined-cycle electric power plants. All systems and components whose failure caused an outage or partial derating during normal plant operating conditions are reported. Information gathered dates from the mid-1960s and is updated quarterly by status reports. These quarterly reports provide the electric utility industry with periodic data and information on the performance of power plants.

On January 1, 1979, NERC assumed responsibility for the operation of the GADS from EEI. The objectives of NERC GADS are coordinated through a Joint Advisory Committee comprised of representatives from the EEI Prime Movers Committee, EPRI, DOE, the NERC Engineering and Operating committees, and the NERC staff.

3.2.1 Contents

The NERC GADS encompasses 2,600 electric power units, including nuclear, fossil, hydro, and combined cycle. The data base is comprised of

safety or commercial quality components, 25 percent electronic and 75 percent mechanical. There are over a million individual and statistically reduced events, encoded in a structured extended text format. Historically, system transients and external initiating events have been grouped together.

Beginning in 1982, the categories started being reported separately. Data supplied to GADS have been recorded by the electric utilities participating in the GADS program. All generic categories of components are included if their failure caused a forced, scheduled, or partial plant outage. Noncurtailing outages of major components are also reported. Immediate recovery actions are not included but long-term solutions are indirectly available through the EEI committees. Approximately 80 percent of the data are verified.

The data are statistically reduced to derive failure frequency rates, modes, mechanisms, and intervals using calculated population and demands. Also considered are the mean repair time and equipment downtime. Access to unit-specific information is available only to utilities that own the unit and others with specific needs. General statistical information is available to the public in the form of quarterly or annual status reports. There is no charge to participants of the GADS program.

The NERC GADS ten-year review report for 1971-1980 on equipment availability presents statistical data sets on the performance of major types of electrical power generating units. For example, performance data on fossil plants identified that 126 utilities reported information on 1,129 individual units for a total of 9,179 unit-years. The equipment group statistics are computed by considering only those situations where components within that equipment group were the primary reason for the outage. Cumulative and unit-year averages are calculated on such quantities as service hours, available hours, scheduled outage rate, mean time between full forced outages, shutdown because of economic reasons, and probability of outage. The number of start demands and successful starts are included.

Additional information as well as illustrative examples are included in the appendixes to this NERC GADS report.

The ten-year review for 1971-1980 of the NERC GADS component cause code report documents all outages attributed to each of the component cause codes. The equations used to calculate the various quantities are included in the report. Each three-digit numeric cause code identifies a specific type of equipment, component, or system within the unit. A unique set of cause codes is provided for each of the major types of electric generating units. The calculated results include such quantities as average number of occurrences per year, average number of outage hours per year, and average duration per outage. The report notes that the quantities are not additive because of overlapping partial outages, or multiple work performed during an outage. Additional information as well as illustrative examples are provided in the appendixes to this ten-year NERC GADS report.

3.2.2 Areas of Usefulness

The NERC GADS is a comprehensive collection of component and system data that addresses outages of electric generating units. Both nuclear and nonnuclear units are considered, which limits the direct applicability of the data to reliability, risk assessment, and aging investigations of nuclear plants. The availability of unit-specific data is limited, but general statistical information in the form of quarterly and annual reports is free of charge. Approximately 80 percent of the data entries are verified. The management of the NERC GADS provides thorough and illustrative examples of the tabular summaries. Limitations on the use of the data are noted, such as the inhomogeneous nature of overlapping partial outages or of multiple work done during a full outage.

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3.3 Government-Industry Data Exchange Program

The Government-Industry Data Exchange Program (GIDEP)⁹² is funded and operated by the United States Government. GIDEP has approximately 650 participants from U.S. government agencies and various industries. All quality levels of components are represented with microcircuit, electronic, electrical, electromechanical, and mechanical systems included. The collected data specifically excludes both proprietary and confidential data; the emphasis is on off-the-shelf items. The data base is constructed from both recorded historical data furnished by the participants and from experimental or test data. Various environments, depending on the component's application, are included in the data base; the operating environment is flagged. The failure frequencies, failure rates, failure modes, failure mechanisms, and mean times to repair may be included.

Access is limited to contributors; no charge is levied except for some special services. The data base is continually updated; standard output products are generated either monthly or quarterly. Special outputs can be generated on demand. Real-time access is possible.

3.3.1 Contents

GIDEP is comprised of four main data banks (referred to as data interchanges):

1. The engineering data interchange contains engineering evaluation and qualification test reports, nonstandard parts justification data, parts and material specifications, and other related engineering data on parts, components, materials, and manufacturing processes
2. The reliability-maintainability data interchange contains failure rate/mode and replacement rate data on parts, components, and

materials based on field performance information and/or reliability demonstration tests of equipment, subsystems, and systems

3. The metrology data interchange contains metrology-related engineering data on test systems, calibration systems, and measurement technology and test equipment calibration procedures.
4. The failure experience data interchange contains objective failure information generated when significant problems are identified for parts, components, processes, fluids, materials, or safety and fire hazards.

Three special services are provided within GIDEP: The ALERT System, which notifies the participant of potential problem areas; The Urgent Data Request System, which allows a GIDEP participant to query all other GIDEP participants on specific problems; and the Metrology Information Service, which provides rapid response to a participant's queries related to test equipment and measuring services.

Two categories of data distribution are available. A full participant maintains microfilmed data banks and indexes within the participant organization, while a partial participant receives all program materials except the microfilmed data bank. A partial participant can request needed data from the GIDEP Operations Center.

If a participant has remote terminal equipment compatible with the Operation Center's computers, real-time access to the GIDEP data banks may be authorized.

3.3.2 Areas of Usefulness

GIDEP is a comprehensive data source that includes some data directly applicable to nuclear power plant risk assessment. The Urgent Data Request

system feature appears to be a potentially useful tool for exchanging data on components in a nuclear environment. The real-time access features and the data search capabilities are definite assets in accessing data rapidly.

A possible negative aspect would be the lack of verification of the data compounded with the nonuniformity of reporting requirements from the various organizations.

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3.4 Military Handbook 217E

Military Handbook 217E (MIL 217E)⁹³ establishes uniform methods for predicting the reliability of military electronic equipment and systems. It provides a common basis for reliability predictions and for comparing and evaluating reliability predictions of related or competitive designs. There are two methods of reliability predictions, namely parts count and parts stress analysis. The methods are applicable for early and late stages of design, respectively. Based on equipment environmental characteristics, piece part count, electronic stresses, submodule repair rates, and system configuration, the RADC-ORACLE computer program calculates piece part failure rates, equipment failure rates, mean-time-to-failure availability, and repair rates. The data base services are available to all military organizations or specific contractors.

The part stress analysis prediction section contains failure rate models for a broad variety of parts used in electronic equipment. This method includes the effects of part quality factors and environmental factors. The tabulated values of the base failure rate are "cut off" at the design temperature and stress of the part.

The parts count method is suitable for early design phase reliability prediction. The method uses information on generic types, quality levels, and environment. The latter two effects are considered with the application of specified factors. The failure rates for both methods are calculated using the same generic expressions.

3.4.1 Contents

The military electronic components are subdivided into 13 categories. Generic failure rates for these categories are derived by applying single-value factors, such as part quality level and operating environment. The tabulated base failure rates are terminated at the rated design temperature and material stress of the part. The handbook provides example failure rate calculations for all of the categories considered as well as for miscellaneous related parts. Maintenance information is covered by a separate program, which uses data from MIL 470 for equipment testing and MIL 471 for prediction of maintenance.

The data contained in MIL 217E is not computerized and status reports are generated every 12 to 18 months. Access to the data is not restricted, nor is there any charge for a copy of the handbook. The data compiled in the handbook is not verified by the data base and only on a voluntary basis by the equipment suppliers/subcontractors.

3.4.2 Areas of Usefulness

The data are comprehensive in detail, but limited in their application to nuclear plant risk assessment. The data are compiled from subcontractor documents according to various requirements. The access to information is limited to published reports. The handbook states that the methods of reliability prediction for military electronic equipment do not "apply to a nuclear survivability environment, nor do they consider the effects of ionizing radiation."

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3.5 Reliability Analysis Center Handbooks

The Reliability Analysis Center (RAC)⁹⁴ at Rome Air Development Center, Griffiss Air Force Base, maintains a comprehensive accumulation of electronic component reliability data and information representing the combined experiences of hundreds and government, industrial, and independent organizations.

Present data acquisition concentrates on microelectronic devices, high-technology components, discrete semiconductors, and nonelectronic parts. Data are solicited among all phases of device and system development, assembly, testing, and field operation. These activities are enhanced and extended through direct interaction between the RAC and the Rome Air Development Center reliability staff. Emphasis is given to failure modes and mechanisms; material, device, and process technology; quality assurance, reliability, and maintainability practices; specifications and standards; test results; and application experience.

Collected data are classified according to physical, material, design, and process control characteristics as well as applied stress environment. The RAC data service maintains databooks, handbooks, reliability studies, and synopses of symposium meetings. Consulting services are available to support user requested information as well as to assume the dominant role in a specific investigation. At additional cost, the reference material identified by the search can be analyzed, evaluated, summarized, or extracted in full. The RAC also conducts symposia and tutorial courses for instruction, familiarization, and use of the various data services.

The RAC is a Department of Defense Information Analysis Center operated by IIT Research Institute of Chicago, Illinois.

3.5.1 Contents

RAC publications include data summaries for specific component types, such as hybrid microcircuits, small, medium and large-scale integration digital devices, linear and interface devices, digital monolithic devices, and discrete semiconductors. In addition, there are reliability and equipment maintenance data books that provide the failure and repair time data on military electronic equipment by application such as subsystem. Handbooks published by RAC provide guidance on methods for analyzing failure data and for designing reliable equipment. Finally, there are publications describing RAC technical reliability studies, technology abstracts, and symposium proceedings related to the reliability of microcircuits and how to protect them from electrical overstress (electrostatic discharge).

3.5.2 Areas of Usefulness

The RAC handbooks are very comprehensive in detail regarding the reliability of military electronic components. In general, the data books concentrate on failure rates, modes, and mechanisms, operating environment, and quality and screening factors. The RAC interfaces extensively with other military data banks for principal data sources. For this reason, the RAC handbooks have limited applications to the reliability of nuclear power plants. The methods of reliability prediction for military electronic equipment are not specifically tailored to a nuclear survivability environment.

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4. FOREIGN DATA SOURCE DESCRIPTIONS AND ACCESS

Brief discussions of the fundamental features of selected foreign sources are provided in this section.^a The data sources are presented in alphabetical order based on their acronyms. The discussion addresses such topics as types of data collected from various sources, period of data collection, access restrictions, degree of data reduction performed, types of systems and equipment data, and generation of data base output reports. For data sources not strictly nuclear in origin, observations are made on the source's usefulness in nuclear applications.

Because of limitations in the amount of descriptive detail as well as the unavoidable subjectivity of the 1982 survey, the reader is encouraged to pursue topics of interest for clarification and a working knowledge of the data source. The discussion of each data source is followed by reference information for contacting the data source representatives.

4.1 Swedish Thermal Power Reliability Data System

The Swedish Thermal Power Reliability Data System (ATV) is maintained and managed by the Swedish State Power Board at Stockholm, Sweden. Engineering and reliability data have been collected from both nuclear and nonnuclear power generating plants. Nuclear data collection began in 1973. Collection of reliability data began in 1976. Over 30,000 events have been recorded in the data base.

All data recorded in the data base have been acquired from plant records. Statistical reductions of data for generation of reports or specific end use are available. Data are currently collected from four operating plants (eight units). Time clocks have been installed on

a. With the exception of Section 4.4 below, this information is largely adopted from Reference 3, and is current as of late 1982. It is included for completeness.

components to record actual exposure time. Event data are available on a broad variety of safety and commercial grade components including pumps, valves, transformers, instruments, diesels, filters, tanks (vessels), and heat exchangers.

Real-time access to the data base is possible through a data base management system. In addition to safety-related system data, data for interfacing (auxiliary) system components are also available. The data base management system provides in-depth flexibility for generation of specific output reports. Although access has been limited to members of the Swedish Utility Consortium and the Swedish Power Board Directorate, a Reliability Data Book ⁹⁵ is available with generic failure rates based primarily on this data base.

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4.2 European Reliability Data System

The European Reliability Data System (ERDS),⁹⁶ sponsored by the European Economic Community (EEC), is used by the European nuclear industry, licensing authorities, and utilities. It contains information on all U.S. light water reactors, and some 35 European reactors. Eighty percent of the

information is accepted from national authorities and utilities and is not internally reduced. The remaining 20% is data collected by the internal data base investigators. The data base provides information on equipment performance, repair and maintenance records, and human factors. The data base is updated every six months but regular output products are not presently available. Access to the data base is limited to data contributors, while some design details are confidential.

4.2.1 Contents

The ERDS is comprised of four data banks:

1. The Component Event Data Bank contains raw data on component failure, reliability, and operating modes. The content is similar to the NPRDS (see Section 2.24). The data are from EEC countries and also Spain and Sweden
2. The Abnormal Occurrences Reporting System contains unusual plant events and human factors associated with normal operation. The content is similar to pre-1984 LERs. Input is from the U.S., Sweden, and all relevant ECC countries except Germany (discussions are in progress there)
3. The Operating Unit Status Report contains production and outage log information on power reactor operation from EEC countries, Spain, and Switzerland, and is very similar to the Generating Availability Data System operated by the U.S. National Electric Reliability Council (see Section 3.2)
4. The Generic Reliability Parameter Data Bank is a bibliography of reliability information for plants with similar classes of components.

The ERDS attempts to establish a uniform method of encoding the types

of data submitted from the various entities. Inconsistencies of detail are monitored automatically by the data bank computer services.

4.2.2 Areas of Usefulness

The data are very comprehensive with direct applications to reliability, risk, and event analysis of nuclear power plants. Information has been assembled on failure frequency, modes, repair, and maintenance. Rate information is based on demands calculated. The time period covered varies from the early 1970s to the present. Using real time access, the output format of the event can be varied by selection of 20 generic and detailed categories.

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4.3 Gesellschaft für Reaktorsicherheit

The German Gesellschaft für Reaktorsicherheit (GRS) has a private arrangement with the Rheinische Westfälisches Elektrizitätswerke (RWE) to compile reliability data from an operating power plant, Biblis B. The data base contains failure rate, maintenance, and operational event data. External event data (floods, earthquake, fire, etc.) are compiled through a separate utility-sponsored data base. The data base provides information on repair and maintenance, and equipment performance. Data are available for 17,000 components, plus additional data on a "subcomponent" level. Output products are produced as needed.

4.3.1 Contents

The GRS Reliability Data Collection on Nuclear Power Plant, Biblis B, includes data from 37 safety-related peripheral systems. This data

collection effort was concentrated on the following components because of their extensive populations and repair action documentation: pumps, valves, electrical positioning devices, electric motors, and drives. For each component type, preface pages and data summary tables are provided. The prefaces provide descriptions of the respective degree of impact of eight influence parameters on different component failure modes. Separate data summary tables are provided for each component type and are structured in a format that allows for the inclusion of the number of pieces of operating equipment, the total number of operating hours, total number of failures, and hourly failure rates with upper and lower bounds.

All of the information listed directly above is correlated to an all damages mode, individual subcategories of failure modes (internal leakage, for instance), and the influence parameters and their subdivisions. Real-time access is possible through a data base management system. Greater confidence has been achieved through use of actual operational data for computation of failure rate and significant event data. A companion data bank is maintained through the participating utility for external event data. There is no access cost to those having authorized use of the data base. Individual data requests are made directly to RWE with approval of GRS.

4.3.2 Areas of Usefulness

The Biblis data base project provides input for interesting comparisons on numerous levels: European versus U.S. plant reliability and component type variations, and the differing influence of various parameters on various component types.

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4.4 Offshore Reliability Data Handbook

The offshore oil industry faces numerous concerns in the exploration, drilling, and production of oil using platforms placed in extreme environments.² The realization that most of these concerns dealt with risk and reliability issues motivated a number of oil companies to pool their resources and efforts to gather the data necessary to address these issues through reliability analyses. The Offshore Reliability Data (OREDA) project began in 1982 as a Norwegian government effort to survey and evaluate existing data banks so that the best elements could be incorporated into a plan for offshore oil industry data collection. The operational phase of this project in 1983 was managed and funded by the OREDA Steering Committee, composed of oil company members. All the member companies contributed funds for the project which involved the monitoring, compilation, analysis, and presentation of data gathered from the individual companies. The results of the program were published in 1984 as the OREDA Handbook.

4.4.1 Contents

The OREDA Handbook is structured by the type of systems (safety, process, etc.). Components are cited beneath these headings and levels are designated by numeric subdivisions, such as 4.2.1 and 4.2.2, much like IEEE Std. 500-1984. A preface section precedes each system and each component type data page set. The system parameters to be considered are listed. A full structured taxonomy, including components and subclasses numerically identified, is also provided. Each component level preface describes a variety of parameters including operating mode, component specifications, and a diagram showing the component boundary. The data table pages all employ the same format in which the heading provides component information (operating hours, component population, etc.) and the body presents the failure and repair data. Mean values for failure data and upper and lower bounds are correlated to failure modes structured much like those in IEEE Std. 500-1984, except that the "Catastrophic" severity category is called "Critical" by OREDA. Repair time is given for each mode as well.

4.4.2 Areas of Usefulness

Although the data in the OREDA Handbook originate from the offshore oil industry, some components and their applications correspond to nuclear plant scenarios. In these cases the component failure data can be useful for incorporation into aggregated values of PRA model input. The key to employing OREDA data would be the close examination and comparison of the offshore oil component features, operating mode, and operating environment with the nuclear plant component of interest to evaluate the degree of correspondence.

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4.5 Systeme Recueil Donnee' Fialilite'

The System Recueil Donnee' Fialilite' (SRDF) data system is managed and maintained through the Department of Central Functions, Electricite de France (EDF). Since its inception in 1978, data have been collected from two operating nuclear power units. The data base was in 1982 being expanded to assemble and evaluate data from six nuclear units. Collected data are being used by a French user group consisting of French utilities and government agencies.

The information contained in the data base is specific to PWRs (data input source). Reliability data are available for 800 components and 3,000 events. Verbal event descriptions are included in the form of extended text. The event data (including maintenance events) are statistically reduced as needed to generate specific outputs.

Data access is available to all members of the French user's group. Information access to others is presently not available, but exchanges of information are given favorable consideration by EDF.

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4.6 System Reliability Service

The Systems Reliability Service (SYREL) Data Base is used by the United Kingdom Atomic Energy Authority (UKAEA) Systems Reliability Service (SRS). It contains information from all industries, of which 20% is nuclear related. Information is submitted by utilities, licensing authorities, and manufacturers in addition to inspection data gathered by the SRS staff. The data base provides information to contributors on equipment performance, reliability, and availability, although certain aspects of manufacturing design are confidential. The data base is updated continuously and status reports are published every two months on 450 generic items.

4.6.1 Contents

Membership in SYREL entitles the user to the services of four other data banks:

1. The Accident Data Base
2. The Maloperation Data Base, covering events of less than accident severity
3. The Event Data Store, applying specifically to nuclear power plants but discontinued because of high cost (approximately 1000 times normal rates)

4. The Piping Data Base, with a limited amount of detailed information which SYREL wishes to supplement with data from U.S. facilities.

The annual membership fee entitles a user to five days of free consulting, regular status reports, and in-house data reduction services. There are approximately 100 international members.

Data on failure, repair, and maintenance are reduced if the information has been collected for a period of over four years. The nonuniform quality of available data complicates verification. Approximately 10 to 15% of nuclear items are verified by SYREL investigators.

4.6.2 Areas of Usefulness

The data contained in this bank are very comprehensive with direct applications to the risk assessment of nuclear power plants. Much information has been assembled on failure, repair, and maintenance. Special reports can be generated at the user's request. The mean turnaround response time is two to three weeks for a written request. Real-time access is available for SYREL members only. At the user's request, SYREL will contact the data supplier and arrange for a confidential exchange of information.

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5. PROBABILISTIC RISK ASSESSMENT DESCRIPTIONS

In 1975 the first PRA of a nuclear power plant, the Reactor Safety Study (WASH-1400),⁴⁰ was completed. A PRA on a nuclear power plant can provide invaluable insights into basic design and operational features that pose a potential risk to the public. This information is useful not only to the analyzed plant (i.e., mitigating potential causes of core damage) but also can be extrapolated to a varying degree for use with plants of similar design.

A survey was made of PRAs published to date. For each PRA identified, a review was made that focused on the type of data sources used as input to the analysis. In this section, a description of each PRA is given. Most of the descriptions are divided into two parts. First, background information on the PRA is provided. Second, the data sources used in the analysis (and where and how these data are presented in the reference document) are discussed.

In a later section of the report, background information on the plants having PRAs and on the PRAs themselves is presented in tabular form. Also, a table that identifies the data sources used in each analysis and how each source was used (i.e., failures, repair times) is included.

5.1 Arkansas Nuclear Unit 1

5.1.1 Background

The Arkansas Nuclear One Unit 1 (ANO1) nuclear power plant PRA³⁵ was performed as part of the Interim Reliability Evaluation Program (IREP). The IREP has several objectives, two of which are achieved by the analysis, presented in this report. These objectives are a) the identification, in a preliminary way, of those accident sequences that are expected to dominate the public health and safety risks, and b) the development of

state-of-the-art plant system models that can be used as a foundation for subsequent, more intensive PRA applications.

The ANO1 PRA considered core melt accidents initiated by LOCAs for six different break size ranges and considered eight different types of transients. The emphasis of the study was on the estimation of core melt accident sequence frequencies. Core melt accidents with the highest frequency were analyzed in terms of containment phenomenology, and associated radioactive material release categories were estimated.

5.1.2 Operational Data Contents

A brief discussion of the event data used in the analysis and the data's origin is presented in Section 2.1 of the PRA. A complete list of the data sources used in the analysis is provided in Table 2.1.

To identify initiating events and the initiating event frequencies, ATWS: A Reappraisal--Part III, Frequency of Anticipated Transients, EPRI NP-801, was used as the basic source.⁴⁴ Additional insight was obtained through reviewing LERs for the plant and for plants of similar design.

A mixture of generic and plant-specific data was used to quantify the fault trees. Basic hardware failure rate data were obtained from a modified WASH-1400 data base. For particular components, plant-specific data obtained from plant logs were used. Plant-specific test and maintenance frequencies obtained from plant logs were used in the analysis. Data for human error rates were obtained from the Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278.⁴⁶

In addition to the above documentation, the utility personnel participating in the study served as contacts with the plant to obtain more information when needed.

Unfortunately the report does not identify the components for which plant-specific data were used. Consequently, the usefulness of the data in other applications has not been assessed.

5.2 Big Rock Point

5.2.1 Background

The Big Rock Point (BRP) nuclear power plant is unique relative to the majority of U.S. commercial nuclear plants from the standpoint of size, design, operating experience, and site location. The small size of BRP (240 MW_e) limits the capital that can be economically spent on plant modifications. Regulatory requirements imposed on nuclear plants on a generic basis after the accident at Three Mile Island made continued operation of BRP an unattractive alternative from an economic perspective. However, on the basis of the long, successful operating experience of BRP, the relatively remote site location, and the fact that the plant was only halfway through its projected lifetime, Consumers Power Company elected to find a better way of evaluating and enhancing plant safety. It was judged that use of PRA³⁶ would allow consideration of the above factors while focusing attention on those features of the plant where modifications are necessary and effective in reducing public risk. The PRA approach also allows evaluation of several different modifications addressing the same problem so that the most cost-effective modification can be pursued. This is crucial if the capital expenditures are to be kept within economic limits.

5.2.2 Operational Data Contents

The BRP PRA, Appendix III, provides a detailed description of the reliability data used in event tree and fault tree quantification. Because of its extensive operating experience and the uniqueness of the BRP design, BRP plant-specific data was used whenever possible. Plant-specific data sources included plant maintenance orders, control room log books,

surveillance tests, LERs, event reports, deviation reports, plant review committee meeting minutes, and USNRC correspondence. The plant-specific data used spanned the period from 1970 to 1979. Data before 1970 did not include maintenance orders or surveillance tests and therefore were excluded. The plant-specific data collected for BRP is presented in detail in Appendix XIII. Table III-4 summarizes 30 plant-specific component failure rates and Table II-06 contains plant-specific maintenance unavailabilities for 20 components. These tables are a summary of the BRP component failure and maintenance outages.

The generic data sources used in the BRP data base originate from the nuclear industry.

A systematic approach was undertaken for the BRP PRA to identify all potential sources of common mode failure. The first step in the treatment of common mode failures was a compilation of a detailed list of common mode initiators. To achieve this, available literature on common mode failure analysis was reviewed. The next step was to qualitatively assess the potential effects of these initiators on BRP systems. The initiator categories and the systems selected for examination are presented in Table VI.1. of the BRP PRA.

The failure data base from the BRP PRA is suitable for PRA applications because it was developed and structured for that purpose. In addition, maintenance frequency data can be applied to obtain average unavailability determinations that can be used on a comparative basis with other plant-specific sources or combined to develop broader-based averages.

5.3 Browns Ferry Unit 1

5.3.1 Background

The first Browns Ferry Unit 1 PRA was made as part of the USNRC IREP.³⁷ Specific goals of the study were to identify the dominant

contributors to core melt, develop a foundation for more extensive use of PRA methods, expand the cadre of experienced PRA practitioners, and extend IREP analyses procedures to other domestic light water reactors.

Event tree and fault tree analyses were used to estimate the frequency of accident sequences initiated by transients and loss of coolant accidents. External events such as floods, fires, earthquakes, and sabotage were beyond the scope of this study and were, therefore, excluded. From the accident sequences, the dominant contributors to the frequency of probable core melt were chosen. Uncertainty and sensitivity analyses were performed on these sequences to better understand the limitations associated with the estimated sequence frequencies. Dominant sequences were grouped according to common containment failure modes and corresponding release categories on the basis of comparison with analyses of similar designs rather than on the basis of detailed plant-specific calculations.

5.3.2 Operational Data Contents

Table C-4 of Appendix C³⁷ contains the IREP generic data set used for the majority of the failure data for quantification of fault trees. Most of this information comes directly from WASH-1400; because failure data were not available for every component in the fault trees, other sources of data were occasionally used. Table C-5 lists 10 components' failure rates that either differ in value or represent an extension of the IREP data set application; the rupture disk leakage probability is based on information from Tennessee Valley Authority (TVA). Repair times from WASH-1400 were used. Preventive maintenance and testing contributions to system/component unavailabilities were derived from information provided by the utility company. Initiator frequencies came from plant-specific data found in EPRI NP-801.⁴⁴

Human errors of omission were included where appropriate in the fault tree models for errors involving test and maintenance, and those involving errors in response to an accident situation. Emergency operating

instructions were reviewed with regard to potential accident sequences to determine the required human interactions with mitigating systems in response to the accidents. Section 3.2 of Appendix C describes in more detail these operator response errors. Explicit human error models were developed, based on the procedures found in NUREG/CR-1278.⁴⁶ It was especially important to create these models for human error events that affected multiple systems. These human error models can be found in Section 4 of Appendix B.

The Reliability Analysis System (RAS)⁹⁷ computer code provided unavailability calculations based on the fault trees and failure data. The code calculates time-dependent system unavailabilities using one algorithm to generate minimal cut sets and another to evaluate the unavailability associated with these cut sets. The code also ranks the cut sets from highest to lowest in terms of contribution to system unavailability.

A limited amount of plant-specific data was used in the analysis. Consequently, there is minimal potential use for the data sources. The generic data sources used in the analysis can support other risk assessments when plant-specific data do not exist.

5.4 Browns Ferry Unit 1 Second Evaluation

A second PRA was performed on Browns Ferry Unit 1.⁹⁸ Sponsored by TVA, it was an analysis that included modeling of the containment and the environmental transport of radioactive materials as well as identifying sequences of failures that could lead to core melt. Several plant-specific data sources were used to support the analysis; these are described in Appendix D of Reference 98. However, this reference document was not available for review. Information obtained from a questionnaire on the sources used is included in a table in Section 6.2, PRA Comparisons.

5.5 Calvert Cliffs Unit 1

5.5.1 Background

The first PRA analysis for Calvert Cliffs Unit 1³⁸ was performed as part of the IREP. Two of the IREP objectives are addressed by this analysis. They are a) the identification of those accident sequences which can be expected to dominate the risk related to the operation of Calvert Cliffs Unit 1, and b) the development of system models that can be used for future, more extensive PRAs of Calvert Cliffs Unit 1.

The scope of this analysis does not include external events (earthquakes, fires, etc.) and the assignment of accident sequences to release categories was performed in a subjective manner with limited plant-specific calculations. Thus, this portion of the study relied heavily on analyses performed previously on similar facilities. Other limitations are discussed in detail in Chapter 8 of NUREG/CR-3511. While accident sequence and release category frequencies were quantified, they are of value primarily in comparative analyses, and the absolute values determined should not be used without a clear appreciation of their inherent uncertainties. The principal product obtained is the integrated engineering logic presented in the plant and system models and insights into plant features contributing significantly to risk, not the specific values computed for accident frequencies.

5.5.2 Operational Data Contents

A mixture of generic and plant-specific data was used to quantify the fault trees. Basic hardware failure rate data were initially obtained from a modified WASH-1400 data base⁹⁹ assembled by the USNRC. Later, the generic data base given in the IREP procedures guide²⁴ was used in the Calvert Cliffs study. For particular components, plant-specific failure data obtained from plant logs were used. Plant-specific test and maintenance frequencies were also obtained from plant logs and used in the analysis. Data for human error rates were obtained from NUREG/CR-1278.⁴⁶

In addition to the above documentation, the utility personnel participating in the study served as contacts with the plant to obtain more information when needed. The PRA team visited the plant to view particular equipment and to discuss questions with plant personnel. The utilities also reviewed periodic reports to ensure accuracy of information.

5.6 Calvert Cliffs Unit 1 Second Evaluation

5.6.1 Background

Another evaluation of the risk to the Calvert Cliffs Unit 1 nuclear power plant, because of concerns about pressurized thermal shock (PTS),¹⁰⁰ was performed by ORNL with the assistance of several other organizations. This evaluation was part of a USNRC program designed to study the PTS risk to three nuclear plants. The other two plants were Oconee Unit 1¹⁰¹ and H. B. Robinson Unit 2.¹⁰² The specific objectives of the program were to a) provide a best estimate of the frequency of through-the-wall cracks (TWCs) in the pressure vessel at each of the three plants, together with the uncertainty in the estimated frequency and its sensitivity to the variables used in the evaluation; b) determine the dominant overcooling sequences contributing to the estimated frequency and the associated failures in the plant systems or in operator actions; and c) evaluate the effectiveness of potential corrective measures.

The program did not consider the effects of external events, such as earthquakes, fires, floods (both outside and inside the containment), and sabotage. ORNL suspects that the effect of excluding such events is not serious because of a) the low probabilities that the events will occur and b) the likelihood that failures of systems caused by external events would cause undercooling situations rather than overcooling situations.

5.6.2 Operational Data Contents

The three PTS analysis reports do not contain information on the input data sources. The input data preparation and analysis was performed by SAI,

Oak Ridge. The data was transmitted to ORNL from SAI by informal letter correspondences. Joe Minarick of SAI led the data collection effort.

5.7 Calvert Cliffs Unit 2

5.7.1 Background

The Calvert Cliffs 2 PRA is the third in a series of four that presents the results of the analyses performed in the USNRC's Reactor Safety Study Methodology Applications Program (RSSMAP). The results are documented in NUREG/CR-1659.¹⁰³ The RSSMAP analyses were an attempt to use insights from the relatively detailed and elaborate WASH-1400 analysis to perform a meaningful plant risk analysis with minimum manpower and economic impacts. It was also desired that the study of plants with differing reactor and containment designs would broaden the class of nuclear power plants explicitly analyzed in terms of risk.

The results presented in the subject document should be considered with caution and placed in their proper context. As was true of all the RSSMAP plants studied, the Calvert Cliffs analysis was conducted primarily with information available in the FSAR, technical specifications, and selected plant procedures. This approach does imply some limitations in the depth of the analysis because as-built systems often differ from those depicted in FSAR drawings. Also, FSAR analysis and technical specifications generally indicate more conservative criteria and guidelines than are actually required for system success.

5.7.2 Operational Data Contents

The Calvert Cliffs 2 PRA almost exclusively used WASH-1400 data. These data came primarily from the nuclear and military industry. Expert opinion is the foundation for portions of the data base. There is little actual operational data in this study.

The failure data base from the Calvert Cliffs 2 PRA is of limited use for PRA applications because it was developed and structured before a number of more current data sources.

5.8 Connecticut Yankee

5.8.1 Background

Northeast Utilities (NU) has been committed to developing and maintaining living PRA models for all company-operated nuclear power plants for use in in-house safety assessments, design engineering evaluation of hardware and procedural changes, and operations policy decisionmaking. In support of this objective the Connecticut Yankee (Haddam Neck Plant) PRA was performed by personnel in the NU Safety Analysis Branch, PRA Section.

The PRA study¹⁰⁴ resulted in the discovery that certain size breaks in the reactor coolant system loop, where a single charging header connected, could not be mitigated. The PRA study was then used to identify modifications to eliminate this problem. Temporary modifications were made during the 1986 refueling outage. These will be supplemented by permanent hardware during the next scheduled refueling outage. The PRA model was used to obtain a temporary exemption to General Design Criteria 35 from the USNRC and to justify continued operation until the next refueling outage.

The existing PRA models for core melt probability are being expanded to cover fire events and flooding. Current use of the models has been for evaluating the safety improvements of approximately 50 individual proposed plant modifications under the Integrated Safety Assessment Program (ISAP) in which NU is voluntarily participating with the USNRC.

5.8.2 Operational Data Contents

The Connecticut Yankee PRA data base contains component failure and maintenance unavailability data, and initiating event frequency data

including typical PWR anticipated operational occurrences, LOSP, and RCP seal failures. Section 4.1.1, "Component Data Collection," describes the process used to gather component failure history, demand history, and run time experience over a 10-year period. Included in the process was the use of the Baseline Events Analysis Reliability Data System (BEARDS), a proprietary NU data base that includes failure and maintenance reports. Section 4.1.2, "Component Reliability Analysis," shows the updated component failure data. The results reflect a Bayesian update of WASH-1400, IEEE-500, and Westinghouse Nuclear Technology Division means and variances, using plant-specific experience where available. Failure-on-demand rates were modeled using beta-distributed priors. Hourly failure rates were modeled using gamma-distributed priors.

The pump data base is broken down into pump types (each is treated separately) and failure modes (start versus run). Limited motor-operated valve (MOV) data were analyzed for critical, infrequently tested MOVs inside the containment. No plant-specific breaker data were obtained. Good plant-specific data exist for batteries, chargers, inverters, motors, and diesels.

Maintenance data are treated by computing an average maintenance unavailability for each component type or system train.

Event frequency data are analyzed in Section 2.6, "Initiating Event Frequency Quantifications," based on plant trip reports and shift supervisor's log book entries for a one-year time period. The first two years of plant experience were discarded as they appear to represent experience typical of early plant operation and tests which are not typical of operation in later years. The plant experience was used to perform a Bayesian update of EPRI NP-2230¹⁸ reactor trip experience.

The failure data base from the Connecticut Yankee PRA is suitable for PRA applications because it was developed and structured for that purpose. A large amount of plant-specific data was used in the analysis and the data could be useful in other applications.

5.9 Crystal River Unit 3

5.9.1 Background

The first PRA of the Crystal River Unit 3 nuclear power plant¹⁰⁵ is an IREP study that provides a quantitative assessment of certain aspects of the public risk associated with its operation.

The assessment includes estimates of the frequency (or probability per year) of radioactivity releases in each of seven discrete categories stemming from LOCAs and various anticipated transients. The primary objective is to identify and estimate the probabilities of those types of accidents that are most likely to cause releases in these seven categories, and to identify and quantify the combinations of hardware and human faults that contribute most to these probabilities.

5.9.2 Operational Data Contents

The component failure data sources and maintenance unavailability data sources used in the Crystal River 3 analysis are discussed in Section 5.1 of the report. The principal data base employed in the Crystal River 3 analysis for component failure rates was the IREP data base.

Plant-specific data were used in the analysis for a) emergency power from the on-site fossil units 1 and 2, b) emergency diesel generators, c) emergency feedwater turbine pumps, and d) maintenance outage frequency and duration of active mechanical components. Section 5.1 also includes discussion of the approach and data sources used in addressing common cause failure, maintenance outage frequency and duration, and human error rates.

The amount of plant-specific data used in the analyses was minimal and would be of limited use in other applications.

5.10 Crystal River Unit 3 Second Evaluation

5.10.1 Background

The second study¹⁰⁶ was done in response to a request by its utility, Florida Power Company. The study, completed in February 1986, was performed by SAIC. It contains a detailed analysis of sequences that could lead to core melt. It does not include an extensive analysis of external events, but the analysis of plant-specific data is thorough.

5.10.2 Operational Data Contents

The Crystal River 3 plant has been operating since March 1977, but it was considered essential to review the daily operating history through 1984 as well as the USNRC monthly reports to determine a start and end date for data evaluation. Because of consideration of the plant's operating history, September 19, 1978, was set as the start date and June 30, 1984 was set as the end date.

Plant records, NPRDS files, and LERs from this period were all used as plant-specific input to the data base, and were supplemented by generic data to narrow the uncertainty bounds. The majority of the generic data used for this base was obtained from nuclear power-related sources; however, a few of the generic sources used to produce the final aggregated failure rates contain data completely or partially from nonnuclear sources, such as the military or the offshore oil industry.

The data base consists of system-by-system folders displaying the various stages of data compilation the analyst went through to calculate the failure rates, a table of generic component failure rates by failure mode, and a partial write-up of the methodology employed for data base development. Each system folder contains component folders for each component type called out on the relevant system fault trees. Within each of these component folders is a checklist of items needed to develop the

data that are indicated by a reviewer as either being included, not available, or not applicable. Such items include computer history file by tag number, plant operating hours and cold shutdown list, and assumptions made for determining interfacing demands. The input to demand-related and time-based failure rate calculations is all included in these folders and final calculations are summarized on failure summary sheets. This method was used to allow the data development process to be traceable, even years after its completion. Rates compiled in this manner have been combined with generic data summarized across component types in a table, which was incorporated into the final quantification of the PRA. Test and maintenance were considered in the construction of component demand histories, and the unavailability rates caused by these situations have been calculated.

This data base employs a thorough approach to the development and documentation of failure information. This data source will provide valuable input to PRA model quantification and component unavailability studies.

5.11 Grand Gulf Unit 1

5.11.1 Background

The Grand Gulf PRA¹⁰⁷ was performed under the USNRC's RSSMAP. The objectives of the study were to use the methods developed in WASH-1400 to determine the sensitivity of dominant accident sequences and risk to plant-to-plant variations.

The results of this analysis should be considered in the proper context. As was true of all the RSSMAP plants studied, the Grand Gulf analysis was conducted primarily with information available in the FSAR.

5.11.2 Operational Data Contents

Plant-specific data, technical specifications, and plant procedures were not available for Grand Gulf because it was still under construction at

the time of the analysis. Consequently, the Grand Gulf analysis was performed using generic data. Because of their similarities in design, Peach Bottom Unit 2 data from WASH-1400 was used in the analysis. All failure rates, failures per demand, and repair time estimates came from WASH-1400. Thus, this PRA is not useful as a source of operational data.

5.12 H. B. Robinson Unit 2

See Calvert Cliffs Unit 1 Second Evaluation (Section 5.6).

5.13 Indian Point Units 2 and 3

5.13.1 Background

The Indian Point Units 2 and 3 PRA study includes a discussion of probabilistic risk assessment methodology; plant, containment, and site analyses; an analysis of initiating events including events outside the plant; an identification of the dominant contributors to risk; and a quantitative statement of the level of safety at the Indian Point nuclear power plants.

The study was prepared by Pickard, Lowe & Garrick, Inc.; Westinghouse Electric Corporation; and Fauske & Associates under the supervision of the utilities.

5.13.2 Operational Data Contents

The review of the data portion of the Indian Point 2 (IP2) and 3 (IP3) PRA (a 1982 internal document prepared by Consolidated Edison and the New York Power Authority) is confined to the site-specific and generic component failure and service hour data sections because these were the only segments available to the reviewers. The LERs produced during a ten-year span of IP2's operation were evaluated to determine their applicability to the PRA data needs. It was eventually decided to use only the LERs generated after IP2 became critical (from May 23, 1973 to December 31, 1979) for the

component data base development, based on the availability of failure event information and more uniform operability, testing, and reporting criteria.

The IP3 data portion presentation parallels the IP2 data section. The discussion below applies to both plants. Distinctions are made for the location of information in the PRA as it applies to IP2 versus IP3.

Opening segments of the IP2 PRA data analysis section (1.5.1), (IP3: 1.6.1) describe the definitions of terms and concepts employed, the assumptions made, and limitations recognized during the data base construction. Separate treatment of these discussions is given for component failure data, component operability test and service hour data, component failure modes, data uncertainty, and data applicability. A set of 39 (IP3: 37) plant-specific component failure mode summaries established the basis for component service hour determinations, the number of failures, and the test data source for each failure mode given for each component. Generic data from WASH-1400, IEEE Std 500, and the LER data summaries on valves, pumps, and diesels were combined with plant-specific failure data to produce "updated" failure information. All the IP2 specialized component hardware failure data, both generic and updated, are contained in Table 1.5.1-4 (IP3: 1.6.1-4). This table contains (by system, component, and failure mode) plant-specific data on the number of failures and service hours or demands. Additional updated and generic mean failure rates and variances are provided. Component types and their failure modes are correlated to the number of failures, service hours or demands, and updated mean and variance estimate for failure rates. For some components, it was determined that specification of the system was warranted because of its impact on the data values; these system-specific cases are also noted in Table 1.5.1-4 (IP3: 1.6.1-4). Tables providing IP2 cold shutdown dates, durations, reasons, and component periodic test frequency are included as 1.5.1-1 and 1.5.1-2 (IP3: 1.6.1-1 and 1.6.1-2), respectively.

Potential users of LER data for reliability analyses will find the experience of IP2 beneficial in noting how and where assumptions and interpretations were made. In addition, the failure-mode-specific

explanations of service hour determinations are insightful for anyone calculating the denominators for hourly (or demand, using the basic methodology concepts) failure rates.

5.14 Limerick Units 1 and 2

5.14.1 Background

At the request of the USNRC, Philadelphia Electric Company, through its contractor (the General Electric Company) and prime subcontractor (SAI), performed a PRA¹⁰⁸ for the Limerick Generating Station. The purpose of the analysis was to assess the risk of operating Limerick Station, specifically with regard to its location near a high population density area. These risks were evaluated to determine whether they represent a disproportionately high segment of the total societal risk from postulated nuclear reactor accidents. The USNRC requested that the Limerick analysis employ the methodology of WASH-1400, with modifications to account for both the design differences and the site-specific differences between Limerick and the WASH-1400 reference plants and sites.

In response to USNRC direction, the Limerick analysis accounted for a revised list of accident initiators based on the Limerick plant design and a more detailed analytical modeling of event sequences following each accident initiator. Plant-design-specific and site-specific data were also included in the analysis of the Limerick Mark II containment and in the meteorology and demography input to the evaluation of accident consequences.

5.14.2 Operational Data Contents

The Limerick Units 1 and 2 PRA, Appendix A, provides a detailed discussion of the input data used in the evaluation of accident sequences. Included are discussions of accident sequence initiators (Section A.1) and equipment and component failure rate data (Section A.2).

The component failure rate data used as input to the fault tree model came from four basic sources: plant records from Peach Bottom (a plant of similar design to Limerick), actual nuclear plant operating experience data as reported in LERs (to produce demand failure rates evaluated for pumps, diesels, and valves), General Electric BWR operating experience data on a wide variety of components (e.g., SRV valves, level sensors, containment pressure sensors), and WASH-1400 assessed median values.

The failure data for the Limerick PRA are of use for other risk analyses. Although the data do not include any plant-specific values generated specifically from Limerick, some plant-specific data from Peach Bottom are included.

5.15 Millstone Unit 1

5.15.1 Background

The first Millstone Unit 1 nuclear power plant PRA¹⁰⁹ was performed as part of the USNRC's IREP. Two of the IREP objectives have been achieved by this analysis. They are a) the identification of those accident sequences that can be expected to dominate the risk related to the operation of Millstone 1, and b) the development of system models that can be used for future, more extensive PRAs of Millstone 1.

The IREP analyses represent integrated plant systems analyses. Detailed analyses were performed on those systems required to respond to a variety of initiating events and on those systems supporting the responding systems. The analysis included unavailabilities during test and maintenance activities, human errors that could arise in restoring the systems to operability following test and maintenance and in response to accident situations, and a thorough investigation of support system faults that could affect operation of more than one system.

5.15.2 Operational Data Contents

The sources of information used in the Millstone 1 analysis are listed in Table 2.1 of the referenced document. The FSAR and plant system descriptions and drawings provided the basic information base for the analysis. This was supplemented by more detailed information in support of particular aspects of the analysis.

To identify initiating events and initiating event frequencies, EPRI NP-801⁴⁴ was used.

Data for quantifying the fault trees were a mixture of generic and plant-specific data. Basic hardware failure rate data were obtained from a modified WASH-1400 data base assembled by USNRC personnel participating in the study. For particular components, plant-specific data obtained from LERs for Millstone 1 and similar plants were used. Plant-specific test and maintenance intervals and duration were obtained whenever possible from discussions with plant personnel and from reviewing plant logs. Data for human error rates were obtained from NUREG/CR-1278.⁴⁶

In addition to the above documentation, the utility personnel participating in the study served as contacts with the plant to obtain more information as needed.

A more current PRA has been performed on Millstone 1 (see the next section). This study included the development of a data base with more reliable and accurate estimates; thus, the IREP PRA is not recommended for further reference for acquiring data.

5.16 Millstone Unit 1 Second Evaluation

5.16.1 Background

NU has been committed to developing and maintaining living PRA models for all company-operated nuclear power plants for use in in-house safety

assessments, design engineering evaluation of hardware and procedural changes, and operations policy decisionmaking. The Millstone Unit 1 IREP PRA was not capable of supporting these needs because of its lack of detail. Thus, a determination was made to perform a complete reanalysis in-house using personnel in the NU Safety Analysis Branch, PRA Section.

The resulting PRA study¹¹⁰ uncovered inadequately sized RHR and long-term decay heat removal in both LOCA and non-LOCA accident sequences. The study has been used to evaluate possible improvements and modifications.

The existing models for core melt probability were expanded to cover fire events and flooding. Current use of the models has been for evaluating the safety improvements of approximately 75 individual proposed plant modifications under the ISAP that Northeast Utilities is voluntarily participating in with the USNRC.

5.16.2 Operational Data Contents

The Millstone Unit 1 PRA data base contains component failure and maintenance unavailability data, and initiating event frequency data, including typical BWR anticipated operational occurrences and LOSP. Section 3.1.1 describes the plant data collection process used to gather system and component failure history, demand history, and run time experience over a 13.5-year period. Section 3.1.2, "System and Component Reliability Analysis," shows the updated system and component failure data. The results reflect a Bayesian update of WASH-1400 means and variances using plant-specific experience. Failure-on-demand rates were modeled using beta-distributed priors. Hourly failure rates were modeled using gamma-distributed priors.

The pump data base is broken down into pump types (each is treated separately) and failure modes (start versus run). The MOV data base is separated into MOVs inside the drywell versus those outside the drywell (statistically significant differences exist in observed failure rates). The MOV data base also reflects failure to open versus failure to close.

The large electrical breaker (4160 V, 480 V) data base shows significant differences from both the WASH-1400 and IEEE-500 data bases, i.e., 3 failures in 34,333 demands for 4160 V breakers and 6 failures in 11,238 demands for 480 V breakers. Good data are also available for diesels and the emergency gas turbine generator.

Maintenance data are treated by computing an average maintenance unavailability for each component type or system train and fitting the data to a beta distribution. This is because maintenance outages are logged on a train basis in many cases.

Event frequency data were developed from a detailed review of plant trip reports and shift supervisor's logbook entries. The first two years of plant experience were discarded as they appear to represent experience typical of early plant operation and tests that are not typical of operation in later years. The plant experience was used to perform a Bayesian update of EPRI NP-2230 reactor trip experience. These calculations are summarized in Section 1.2 of the PRA report.

The failure data base from the Millstone Unit 1 PRA is suitable for PRA applications as it was developed and structured for that purpose. The data set contains an extensive amount of plant-specific data that can be used in other applications.

5.17 Millstone Unit 3

5.17.1 Background

In 1981, the USNRC requested NU to submit a PRA to identify, analyze, and quantify the additional potential risk contributed by the Millstone Unit 3 PWR to the surrounding population. This request was made in part because of USNRC concerns about high population density sites and desires to have a PRA available to assist the USNRC staff in reviewing the FSAR and operating license application. The scope of the study was to include internal and external events, and the determination of core melt frequency,

containment failure probability, and hypothetical public consequences. The study was performed by Westinghouse, the reactor supplier for Millstone Unit 3.

5.17.2 Operational Data Contents

The Millstone Unit 3 PRA¹¹¹ relied on existing generic reliability data bases because the study was to be submitted three years before plant operation. Appendix 2-A, "Unavailability Model and Failure Rate Data Base for Random Component Failures," documents the failure data used. The data sources included Westinghouse Nuclear Technology Division data (proprietary), WASH-1400, NREP Procedures Guide, and IEEE-500.

Event frequency data was based on EPRI NP-2230 reactor trip experience. LOSP experience was based on a Bayesian update of ORNL experience using 13 years of site-specific experience. These calculations are summarized in Section 1.1.3, "Quantification of Internal Initiating Event Frequencies."

The PRA predates the operation of the unit by three years and provides no plant-specific data. Future updates and revisions to the Millstone Unit 3 PRA models will incorporate plant-specific data when it becomes available in statistically significant levels. At that time, the data would become more useful.

5.18 Oconee Unit 1

See Calvert Cliffs Unit 1 Second Evaluation (Section 5.6).

5.19 Oconee Unit 3

The first PRA analysis of Oconee Unit 3 was performed for the USNRC RSSMAP.¹¹² In this analysis, a limited amount of plant-specific data was used. In contrast, the second Oconee Unit 3 PRA used a large volume of

plant-specific data. Also, this second analysis was far more comprehensive (Level 3). Refer to the second Oconee PRA for the type of plant-specific data for Oconee Unit 3 that has been produced.

5.20 Oconee Unit 3 Second Evaluation

5.20.1 Background

Following both WASH-1400 and the Three Mile Island incident, it became increasingly apparent that risk assessments were to become a crucial tool for the investigation and evaluation of nuclear power plant accident risks. During the four years between the two events cited above, however, the sponsorship for risk evaluation efforts originated strictly from regulatory bodies. It became clear that a more direct industry involvement in reliability and risk assessment was needed to allow for the maximum benefit from the insights gained and to enable utilities to be conversant with USNRC licensing evaluators in the terms of the new technology. To enable the utilities to initiate risk evaluation projects on their own, EPRI's Nuclear Safety Analysis Center (NSAC) undertook the management of a PRA project¹¹³ to be performed in cooperation with a utility. Oconee Unit 3 was selected for this study because of the strong project support available from Duke Power, the access to detailed design and prior RSSMAP study information, the Oconee operating experience, and the USNRC's original intent to select Oconee as one of the first plants for the IREP study.

5.20.2 Operational Data Contents

The Oconee PRA data base includes component failure and maintenance duration data, initiating event data, and a special section on "rare-event" initiators such as LOCF and steam generator tube ruptures (SGTRs). Section 5.1.6, "Updated Component-Failure Data," displays aggregated values of generic and plant-specific data as "updated" values in a four-page table. Component types and failure modes are correlated to mean, median, 5th and 95th percentile, and variance values. There is no hierarchical structure of component types and failure modes, and components are listed in

a generic sense unless it has been determined that the failure rate is sufficiently different to warrant the notation of a particular system. Maintenance frequency data, for the same five parameters as the failure data, are associated with specific components that are noted by their system and train letter identifiers (e.g., LPI Pump C). Initiating events are dealt with on an individual basis (pipe-break, small LOCA, etc.) through an explanatory section on the treatment of these events. A final table in this section summarizes these initiating event categories and their associated frequencies.

The failure data base from the Oconee PRA is suitable for PRA applications because it was developed and structured for that purpose. In addition, the plant-specific data can be used on a comparative basis with other plant-specific sources or combined to develop broader-based averages.

5.21 Peach Bottom Unit 2

5.21.1 Background

Peach Bottom 2 and Surry 1 were the two plants whose risks were assessed in WASH-1400,⁴⁰ the first comprehensive risk study performed on nuclear power plants. It was performed to assess the risks associated with a typical PWR and a typical BWR and to place those risks into proper perspective with other societal risks. It was motivated by Congressional consideration for renewal of the Price-Anderson Act, a law which establishes, in part, an insurance pool for nuclear plant accidents and limits public liability for those accidents. Surry 1 and Peach Bottom 2 were selected to represent the typical industry PWR and BWR, respectively, primarily because of their stage of completion (they were readily accessible by the study team) and they were typical of current generation plants of the 1000 MWe class.

As noted in the Nuclear Data Bases section, the WASH-1400 data represent an aggregation of data from a number of nuclear and nonnuclear sources. Because it was the first of its kind, a considerable amount of

effort was used to collect and compile these data. Most of the failure data were compiled by searching through LERs, although numerous other sources were used. Other sources include data from foreign nuclear plants, fossil fuel plants, IEEE, military sources, and chemical plants. Technical specifications and discussions with plant personnel provided test interval and repair time estimates.

5.21.2 Operational Data Contents

Event occurrence frequencies, equipment failure rates, and appropriate time intervals (i.e., test intervals and repair times) are contained in Appendix III of WASH-1400. Although there was considerable uncertainty in the data because of their sparsity, subsequent data collection and evaluation have tended to validate, or confirm, the values selected.

Most PRAs performed since WASH-1400 to some extent have used the data compiled in WASH-1400 and/or subsequent refinements of that data. It is used primarily where data specific to the plant studied are not available. Some refinements to the data are recommended.¹¹⁴

A major drawback in the use of most of these data is the lack of sufficient resolution for event descriptions. WASH-1400 event categories tend to be larger blocks than an analyst would like to specify in fault tree models, particularly for electrical and control system components. For example, the WASH-1400 may list "control device" as a component, but an analyst may find it necessary to break that "control device" into a large number of components.

5.22 Seabrook Units 1 and 2

5.22.1 Background

The PRA¹¹⁵ performed on the twin nuclear power plants that comprise the Seabrook generating station in New Hampshire was done during the

construction phase of these units. A full assessment, including seismic and consequence analyses, was performed from 1982 to 1983.

5.22.2 Operational Data Contents

Nuclear industry data sources, including the LER data summaries and WASH-1400 that are described fully in this review document, were used as input to the data documented for the Seabrook PRA. Proprietary data from the consultant responsible for the effort was also used, but was not reviewed during this study to determine its origin. Plant-specific sources were not available because the plant had not been licensed to operate.

The Seabrook PRA Data Analysis Section (Section No. 6 in Volume Two) contains component failure rates, common cause failure parameters (beta factors), component maintenance frequency and duration, human error rates, and initiating event frequency data. A seven-page table of component failure data includes mean, median, and 5th and 95th percentile values associated with specific component types and failure modes. A table of common cause failure related beta factors is given for a more select and system-specific list of component types. Generic component maintenance frequency distributions have been derived. Components, such as diesel generators and startup feed pumps, were correlated to their appropriate maintenance frequency distribution type and values (mean, 5th and 95th percentile). A similar structuring method was used for the component maintenance mean duration distribution table. Basic human error rates per demand are provided in a table that categorizes them either as errors of commission or omission. Finally, the Seabrook PRA data base displays mean, median, 5th and 95th percentile, and variance values in a summary table for 24 initiator categories. The development processes for all of these tables are explained in preceding sections, including the equations and calculational and data references that were used.

For Seabrook, the risk analyses predate the plant operation date. Therefore, while the analysis provides insights to different approaches for modeling and quantification it cannot supplement the available body of

plant-specific reliability data. The values included should, therefore, be used in full recognition of their scope and limitations.

5.23 Sequoyah Unit 1

5.23.1 Background

The PRA for Sequoyah Unit 1 is one of four that present the results of analyses performed in the RSSMAP. Other plants included in the analyses were Grand Gulf, Oconee Unit 3, and Calvert Cliffs Unit 2. The RSSMAP analyses were an attempt to use insights from the relatively detailed and elaborate WASH-1400 analysis to perform a meaningful plant risk analysis with minimum manpower and economic impacts. It was also desired that the study of plants with differing reactor and containment designs would broaden the class of nuclear power plants explicitly analyzed in terms of risk.

5.23.2 Operational Data Contents

The Sequoyah 1 PRA¹¹⁶ was performed before the plant's commercial operation. Consequently, no plant-specific failure data was included in the analysis. WASH-1400 data were used almost exclusively in the analysis. Thus, this PRA is not recommended as a source of data.

5.24 Shoreham

5.24.1 Background

Before its operation, a PRA was performed on the Shoreham Nuclear Power Station¹¹⁷ to assist in identifying potential accident sequences and determining whether minor plant alterations could reduce their chances of occurring. An overall idea of the areas of risk significance before plant start-up may also help to focus the preventive maintenance programs on important components and systems. The data used to quantify the Shoreham PRA is not specific to Shoreham operating experience but is taken from

generic sources and modified by data from studies on similarly designed plants that were already operating (e.g., Fitzpatrick).

5.24.2 Operational Data Contents

The input data appendix includes data on accident sequence initiators, component failure, human failure, system unavailability because of maintenance, diesel unavailability, LOSP, and dependent failure analysis. There is no uniform format used for data presentation on these varied subjects but data of interest within sections is organized into tables, such as "Cross-reference between BWR Anticipated Transient Categories and Applicable Operating Experience." The initiating event categories and initiating event frequencies were based on EPRI NP-801. The component failure rate section has the greatest amount of structuring, using a six-page table to compare rates extracted from four basic sources: LER summaries for pumps, diesels, and valves; a source which uses General Electric BWR data, WASH-1400, and IEEE Std. 500-1977. Rates from each of these sources are correlated to a general grouping (highest level component type, such as valves), a component type (relief valves, for example), and as many failure modes as were available from these sources. Limerick PRA data were used to estimate unavailability and maintenance actions. Error indicators such as upper and lower bounds or error factors are included where available. Data selection was based on a hierarchy indicated by the order of source listing above.

From the standpoint of BWR accident sequence and initiating event evaluation, the Shoreham PRA may provide useful insights report and their usefulness has been addressed. Because Shoreham was not operational at the time of the analysis no new plant-specific data were produced.

5.25 Surry 1

See Peach Bottom 2 (Section 5.21).

5.26 Yankee Rowe

5.26.1 Background

The Yankee Nuclear Power Station (YNPS) was designed during the 1950s, and was first licensed for operation in 1961 by the U.S. Atomic Energy Commission. The plant is at a remote location in western Massachusetts, in the town of Rowe. It is relatively small, compared with contemporary nuclear power plants, reliably producing about 185 MW electric power, with a lifetime capacity factor exceeding 70%.

A probabilistic safety study (PSS) of the YNPS was performed by Energy Incorporated and Yankee Atomic Electric Company to provide additional insight into the design and operation of the plant and to use the latest analytical tools in support of the decisionmaking process. A spectrum of events ranging from turbine trips to large-break LOCAs was examined. Plant-specific data and modeling were used in assessing the likelihood of core melt and risk to the public. The analysis is documented in the YNPS PSS.¹¹⁸

Because of YNPS's age, an extensive history of plant experience was available for use for the comprehensive risk analysis performed in 1983. Plant information records, maintenance department information (such as surveillance schedules and machinery history cards), instrument and controls department item identification index, and reactor engineering department operating data report and statistics all provided valuable data resources for the PRA. LERs and NPRDS data were also used to reduce the data reduction workload to a manageable effort. Special generic data studies on component-specific issues were accessed to supplement this plant-specific information.

5.26.2 Operational Data Contents

Section 7 of the PRA describes the Yankee PSS Data Development and Use. The major data areas addressed in this section are: initiating

events, sequence data, top event data components, and human error data. Subsections describe the data development process for each of these cases, and the resulting values are presented in a series of data tables. Twelve event tree descriptions of initiators are correlated to a means and variances for rates occurrence in Table 7-1. An alphabetical presentation of component types (mechanical, then electrical), subtypes where deemed necessary, and failure rate data in terms of mean, median, range factor, and variance values are logged in Table 7-2. Table 7-3 organizes NUREG/CR-1278 human error probability data into a matrix format, listing task descriptions along the side and stress levels along the top with mean and variance error rates at their junctions. These human error data are also given in Table 7-6, but is applied to specific event tree-identified manual actions. Special consideration was given in this PRA to piping and tube failure rates; therefore, mean and variance values are cited for small, intermediate, and large LOCAs and secondary system piping failures in Table 7-7.

The component hardware data are well-based because they are derived from 22 years of records. It is also useful because of the structure and depth of the presentation; for example, the inclusion of data on pumps in different systems (emergency feedwater, condensate, service water). Such detail is very helpful for PRA model applications for similar plants.

5.27 Zion Station

5.27.1 Background

After the Three Mile Island accident, the staff of the USNRC examined the risk posed by a number of nuclear power plants, concentrating on those that are located close to major population centers; one of these plants was the Zion station. The USNRC staff assessment, which was of limited scope, was performed by postulating that the PWR plant analyzed in WASH-1400 was located at the Zion site.

The detailed risk assessment conducted for the Zion station as a result of this assessment considered both Units 1 and 2. A comprehensive data

base, covering topics similar to those dealt with in the Shoreham and Oconee Unit 3 PRAs, is discussed and presented in PRA Section II.4.4 on Data Base Development of the report.³⁹

5.27.2 Operational Data Contents

The Zion PRA data base includes generic, plant-specific, and combined "updated" component failure data, maintenance frequencies for components, initiating event data, human error rates, and component operability, test, and service hour data. A nine-page component failure data table specifies mean values and 60% confidence interval error factors for generic data and updated mean values and variances for particular component types and failure modes. Most of the component failure rates were applicable to all systems, but exceptions are noted in some cases. Tables with maintenance frequency mean and variance values for selected components; tables with initiating event occurrence probability mean, median, and 90% confidence bound values; and fluid systems unavailability values are among the Zion PRA Data Base tables with features. A series of graphs shows the distribution of probability density versus occurrences per year for each initiating event; for example, loss of RCS flow, core power excursion, and turbine trip. A System Description and Analysis Summary section provides brief descriptions of the safety systems essential to core damage prevention.

Information from this PRA is useful for analysis of performance, availability, and reliability of similar nuclear plants and for calculating aggregations of plant-specific values.

6. DATA SOURCE COMPARISONS

A summary of the description and contents of the nuclear and PRA data sources is presented in the form of tables in this section. The two groups are addressed separately.

For both groups, the tables provide an easy means of evaluating one specific source, or for making comparisons between different sources.

6.1 Nuclear Data Source Comparison

In this section, tabular comparisons of the nuclear data sources are presented. The tables provide a quick index to the origin, contents, accessibility, and areas of potential application for each data source.

The entries in the following tables should be reviewed with some caution. In some cases, a data source may contain a certain type of information, but it may not be as comprehensive as required to fulfill a specific data need. This type of evaluation must be made on a case-by-case basis. However, when an assessment was made that a data source marginally included a specific feature, L for "limited" was used in the tables.

All the nuclear data sources are included in each of the tables except for the EPRI reports source. The variety present in the EPRI reports precluded evaluating them collectively. Also, as pointed out earlier the reliability of ac power systems and diesel generator availability sources have been combined for these comparisons. If the contents of a table do not apply to a data source, then this is indicated by an "X" in the "none included" column of the table.

A discussion of each of the tables follows. The tables are presented consecutively at the end of this discussion.

Table 1, a data source overview, includes a brief statement of the focus of each source. Also the type of data source (report versus

computerized data base) is indicated and whether or not the source is updated. The remaining columns of Table 1 provide an approximate idea of the scope of the source. Design information relates to the static or planned operation of the plants, their component layouts, and operating procedures. Operational scope relates to events that occur during operation; it includes testing as well as operation abnormalities. The "aggregated" column refers to data that are combined or summed up over a set of events or time period, generally to produce statistics for particular applications. Finally, derived data are data that result from further manipulation of aggregated data. Counts may be divided by exposure times to produce occurrence rates, or more complicated analyses may be performed. Each of these refinements of the data is described in further detail in the remaining tables.

The type of pedigree data each source contains appears in Table 2. Pedigree data consist of information on characteristics or features that are static with time, or that are not anticipated to change rapidly with time (e.g., component type and population, design characteristics). In cases where only limited design information such as plant type and vendor is contained in the data source, an L is used.

The scope of the data sources relative to the operating data contents is presented in Table 3. This information includes basic off-nominal events and descriptions, the duration of unavailabilities, and initiators (events that lead to a reduction in power and/or challenge safety systems). In the description of individual events, the failure mode and failure cause columns are marked only if the source identifies these explicitly for each event described. The event environment column is marked if the source describes physical parameters or other environmental factors for each event. Although the emphasis in Table 3 is on individual events, sources citing initiating event frequencies are also included.

Table 4 is used to identify any special-purpose areas of operational data captured in a data source (i.e., human reliability, root cause). The "operational" column of Table 1 is checked for a source if any items from

Tables 3 or 4 are marked for a data source. Possible application areas for the data sources are reflected in these tables.

Information on aggregated data appears in Table 5. The level which the data are aggregated over (i.e., plant, component) is indicated. Also indicated are some of the possible types of aggregates. For example, the "design" column is checked if the data source contains tabulations of component failure by component design or manufacturer. Many data sources have the capability of readily producing aggregated data sets or products but do not actually contain them. Parenthesis around entries in Table 5 are used to indicate when a data source has the capability of producing aggregates.

Table 6, Derived Data, is the last table closely related to Table 1. It shows whether the source contains calculated data. In most cases, it refers to contents of reports associated with a source. As with the previous table, parentheses indicate that a capability to calculate derived data exists. The derived data may be rates or probability estimates, as mentioned above, or they may be the result of statistical evaluations or other modeling.

Tables 7, 8, and 9 provide a map of the origin of the data in each source. Only the immediate origin is described; there is no attempt to trace data back to their ultimate in-plant source. Knowing the data from which a source evolved can be used in evaluating merit. These tables are not intended to be all inclusive; in some cases only the primary input data sources are identified.

Finally, the access instructions, computerized features, and overall availability of each source are presented in Table 10.

TABLE 1. DATA SOURCE OVERVIEW

Data Source Title	Acronym or Abbreviation	Type ^a			Scope ^b			Page
		Report	Data Base	Design	Operational	Accumulated	Refined	
Accident sequence precursor	ASP	U	U	U	X	X	X	Significant initiating events and accident precursor.
Aging root cause	BC-aging	F	F	—	X	X	X	Aging-related root causes for component failures in selected systems.
Component root cause	BC-comp	F	F	—	X	X	X	Root cause fractions for failures of selected components.
Dependent failure/harsh environment	Harsh env.	F	—	—	X	X	X	Identifications and quantification of dependent failures, especially in harsh environments.
CRD DG	CRD DG	U	U	X	X	X	X	Data on the availability of components of liquid metal and other advanced reactors.
Electric motor aging	El. motor aging	F	—	X	X	X	—	Detailed failure design and aging data for electric motors.
Diesel generator availability	DG avail.	F	—	X	X	X	X	Diesel generator availability, reliability, and risk significance.
Diesel generator reliability	DG reliab.	F	—	X	X	X	X	Diesel generator reliability and unavailability.
Diesel generator test intervals	DG test	F	F	—	—	—	X	Example and methodology for assessment of DG TI and its impact on unavailability.
Diesel generator performance	DG perf.	F	—	X	—	X	—	Detailed analysis of DG performance specific to the subcomponent and manufacturer.
Engineered safety feature activation/scram data base	ESAF/Scram	U	U	—	X	X	—	All unplanned ESF activations and scrams in US BWR.
Evaluation and update of base-line data	Baseline	F	—	—	X	X	X	Updated generic data from several sources for screening event sequences in probabilistic risk assessments.
Human error in risk assessment	HE RR	F	F	—	—	X	X	Cataloging of human reliability analysis data from PRRs.

TABLE 1. (continued)

Data Source Title	Acronym or Abbreviation	Type ^a		Scope ^b				Focus
		Reports	Data Base	Design	Operational	Aggregated	Derived	
IIR data summaries	IIR sum.	I	I	I	X	X	X	Gross failure rates for pumps, valves, and other components. ^c
NRC reactor safety data bank	NRC-DRSS	U	U	X	X	X	X	Primarily instrument reading versus time for experiments in test reactors.
Initiating events	init. ev.	I	d	--	X	X	X	Events causing scrams in US NPP (1962-1983).
In-plant reliability data system	IPRDS	I	I	I	X	X	X	Failure and repair time data for pumps, valves, electrical components, based on in plant records. ^e
IEEE Std. 500-1984	IEEE-500	I	--	--	I	X	X	Electrical, electronic, sensing components, and mechanical equip. Data aggregated from many sources.
licensed Operating Reactors Status	Grey Book	U	d	I	X	I	--	Monthly operating status of US NPP, with daily power outputs and causes for outages and reductions.
IIR compilation	IIR compl.	U	--	--	X	--	--	Summary of abstracts for IIRs; with key words.
Maintenance data base	Maint. Db	U	U	X	X	{X}	X	Maintenance actions and plant outages (frequency, down time, relative performance).
Nuclear plant reliability data system	NPRDS	U	U	X	X	X	X	The major source for ongoing reporting of NPP equipment failures.
Nuclear power experience	NPE	U	U	X	X	X	--	S. W. Stoller Corp. NPP event descriptions organized by system and date, with keywords.
NSIC IIR data base	IIRs-NSIC	--	U	--	X	{X}	--	IIR abstracts since 1960.
OPEC-2	OPEC-2	U	U	X	X	X	{X}	Operating history and significant maintenance at NPP.
Pipe break Frequency	Pipe bk.	I	--	--	I	X	X	Frequency of pipe breaks involving greater than 50 gpm leakage.
Reactor safety study	WASH-1400	I	--	I	X	X	X	First attempt to gather data for NPP risk assessment.
RCP seal failures	RCP seals	I	--	X	I	I	X	Detailed analysis of RCP seal leaks and their impact on plant safety.

TABLE 1. (continued)

Data Source Title	Acronym or Abbreviation	Type ^a		Scope ^b				Focus
		Reports	Data Base	Design	Operational	Aggregated	Derived	
Safety system unavailability	Syst. U.	U	U	--	X	X	--	Actual or potential unavailabilities of systems in US NPP reported in LIRs.
Reportable events file	50.72 DB		U	1	X			To provide the NRC with immediate notification of events reported under 10 CFR 50.72, 72.71, 50.55, 20.402, and 20.403.
Sequence coding and search system	LIR SCSS		U	--	X	(X)	--	Licensee event reports stored in a computer-readable, searchable form.
Snubber performance	Snubbers	F	--	X	--	X	--	Detailed analysis of snubber failure in relation to design, vendor, manufacturing, installation, and operation.
System interaction	Syst. Inter.	F	F	--	X	X	--	Implications of adverse system interactions.
Stat. analysis of IPWDS data	IPWDS stat.	F	--	1	--	1	X	Determining factors that explain the variability in failure data.
Tech. spec. violations	tech. spec.	U	U	--	X	X	X	Tech. spec. violations and outages required by 1.5. in US NPP as reported in LIRs.
Boron & flow valve study	B&F valve	F	--	X	X	1	--	Causes of valve failures that cause shutdown in US NPP.

a. F - fixed, U - updated, -- - not applicable.

b. X - yes, -- - no, 1 - limited. Entries in parentheses indicate capabilities that exist just in software.

c. IWEI LIR data summaries included nine reports on the following components: batteries and battery chargers, control rods and drive mechanisms, diesel generators, instrumentation and control components, inverters, penetrations, protective relays and circuit breakers, pumps, and valves.

d. Magnetic tapes are obtainable, but no manipulation capabilities exist.

e. IPWDS includes reports on the following components: diesel generators, batteries, chargers, and inverters; pumps, and valves.

TABLE 2. PIGGEE DATA^a

Data Source	None Included	Component Type and Population	Operating Characteristics	Test Characteristics	Design Characteristics
ASP	--	1	--	--	L
RC aglog	X	--	--	--	--
RC comp.	X	--	--	--	--
Harsh env.	X	--	--	--	--
CR100	--	X	X	X	X
FL motor	--	L	X	--	X
D6 avail	--	X	X	L	X
D6 reliab.	--	X	X	X	LC
D6 test	X	--	--	--	--
D6 perf.	--	L	--	--	--
ESM/Scm	X	--	--	--	X
Baseline	X	--	--	--	--
HEBA	X	--	--	--	--
LIR sum.	--	X	L	--	--
MBC DRS	--	--	X	L	--
Init. ev.	X	--	--	--	--
IPBDS	--	X	L	L	L
1111-500	--	--	L	--	L
Gray Book	--	--	X	--	--
LIR comp.	X	--	--	--	--
Maint. Db	--	--	L	--	X
MPBDS	--	X	X	X	X
MPI	--	X	X	X	X
LIRs MSIC	X	--	--	--	--
OPIC-2	--	X	--	--	X
Pipe bh.	--	L	--	--	--
WASH-1400	--	L	X	X	X
KCP seals	--	X	X	--	X
Syst. U.	X	--	--	--	LC
50-72 BB	X	--	--	--	--
LIR-SCSS	X	--	--	--	--
smelters	--	L	X	--	X
Syst. Infr.	--	--	--	--	--
IPBDS stat.	X	--	--	--	--
Tech. spec.	X	--	--	--	LC

TABLE 2. (continued)^a

Data Source	None Included	Component Type and Population	Operating Characteristics	Test Characteristics	Design Characteristics
BAR value	X	?	...

a. X = yes, ... = no, L = limited.

b. Acronyms are defined in Table 1.

c. Design data only includes plant type and series.

TABLE 3. OPERATING DATA^a

Data Source ^b	None Included	Individual Events				Event Exposure		Downtime		Initiator ^d Data
		format ^c	Failure Mode	Failure Cause	Event Environment	Operating time	Demands	Testing	Maintenance	
ASP	--	C/E	X	X	--	X	--	--	--	X
RC-aging	--	C/S	X	X	L	--	--	--	--	--
RC-comp.	--	C/S	X	X	L	--	--	--	--	--
Harsh env.	--	C/S	X	X	X	--	--	--	--	--
CREEO	--	C/AB	X	X	X	X	L	--	X	L
El. motor	--	S	X	X	X	--	--	--	--	--
DG avail.	--	C/S	X	X	--	L	X	--	L	L
DG reliab.	--	C	--	--	--	--	X ^e	--	--	--
DG test	X	--	--	--	--	--	--	--	--	--
DG perf.	X	--	--	--	--	--	--	--	--	--
ESIA/Scm.	--	C	--	X	L	--	--	--	--	A
Baseline	--	--	--	--	--	--	--	--	--	L ^f
HERA	X	--	--	--	--	--	--	--	--	--
LER sum.	--	C/S	X	X	--	X	L	--	--	--
MRC-DRSS	--	C/AB	X	--	X	X	--	--	--	T/L0
Init. ev.	--	C/S	X	L	--	--	--	--	L	A
IPRDS	--	C	X	X	--	X	L	L	L	--
IEEE-500	X	--	--	--	--	--	--	--	--	--
Grey Book	--	--	--	--	--	X	--	--	L ^f	A
LER cnp1.	--	AB	X	X	--	--	--	--	X	T
Maint. Db	--	C/S	--	--	--	--	--	X	X	A
NPRDS	--	C/S	X	X	--	X	L	--	L	--
NPE	--	AB/E	X	X	L	X	L	--	L	A(L)
LERs-NSIC	--	C/AB	X	X	--	--	--	--	X	T
OPEC-2	--	C/AB/E	X	X	--	X	--	X	X	A(L)
Pipe bk.	--	C/AB	X	X	X	--	--	--	--	--
WASH-1400	--	S(L)	--	--	--	L	L	--	L	L
RCP seals	X	--	--	--	--	--	--	--	--	--
Syst. U.	--	S	X	X	L	--	--	--	L	A(L)
50.72 DB	--	C/S	X	X	L	--	--	--	--	X
LER-SCSS	--	C/AB	X	X	--	--	--	--	--	T
Snubbers	X	--	--	--	--	--	--	--	--	--
Syst. Infr.	--	--	X	X	X	--	--	--	--	X
IPFD stat.	X	--	--	--	--	--	--	--	--	--
Tech. spec.	--	C/S	--	X	L	--	--	L	L	--

TABLE 3. (continued)^a

Data Source ^b	None Included	Format ^c	Individual Events		Event Exposure		Downtime		Initiator ^e Data
			Failure Mode	Failure Cause	Event Environment	Operating Time	Demand	Testing Maintenance	
B&B valve	---	AB	---	---	---	---	---	---	---

a. X - yes, --- - no, I - limited.

b. Acronyms are defined in Table 1.

c. f - coded, S - structured text, AB - abstract, I - extended text.

d. T - transient's, LO - LOCA initiators, EL - external event initiators, O - other, A - all.

e. Has failure data for last 20 starts and for last 100 starts of DG at each plant.

f. Rates from other sources for events that cause scrams; also, aggregates of fire and flood occurrences from other sources are provided.

TABLE 4. SPECIAL PURPOSE OPERATIONAL DATA^a

Data Source ^b	None Included	Root Cause	Common Cause	Purpose	
				Errors ^c	Recovery Actions
ASP	--	--	--	--	--
RC-aging	--	X	--	--	--
RC-comp.	--	X	--	--	--
Harsh env.	--	--	X	--	--
CREED	--	X	X	IN	X
El. motor	X	L	--	--	--
DG avail	--	L	X	X	--
DG rellab.	--	--	--	--	--
GG test	X	--	--	--	--
GG perf.	X	--	--	--	--
LSA/Scm.	--	--	--	L	--
Baseline	X	--	--	--	--
HEMA	--	--	--	X	X
LIF sum.	--	L	X	L	--
MRC DBSS	--	--	X	--	X
Intl. ev.	--	--	--	--	--
IPADS	--	L	L	L	L
ITEE-500	X	--	--	--	--
Grey Book	X	--	--	--	--
LER comp.	--	L	L	L	L
Maint. Db	--	--	--	X	--
MPADS	--	L	L	L	--
MPE	--	L	X	L	L
LEIS-MSIC	--	L	L	L	L
OPEC-2	--	L	L	L	L
Pipe bk.	X	--	--	--	--
MASH-1400	--	--	--	IN/OP	X
RCP seals	--	X	X	--	--
Syst. U.	--	L	--	--	--
50.72 DB	X	--	--	--	--
LER-SCSS	--	L	L	L	--
Snubbers	--	L	--	L	--
Syst. Intr.	--	--	L	L	L
IPRO stat.	X	--	--	--	--
Tech. spec.	--	L	L	--	--

TABLE 4. (cont'd)

Data Source ^b	None Included	Root Cause	Purpose	
			Common Causes	Human Reliability Errors ^c Resolving Actions
Bad value	---	1	---	---

a. 1 = yes, --- = no, 1 = failed.

b. Acronyms are defined in Table 1.

c. HM = test and maintenance; DP = operator.

TABLE 5. AGGREGATED DATA^a

Data Source ^b	None	Level ^c	Aging	Event Sequencing	Root Cause	Design	Human ^d	Performance Indicators
ASP	--	P/S/C/PR	--	--	--	--	--	--
RC-aging	--	S/C	X	--	X	--	--	--
RC-comp.	--	C	--	--	X	--	--	--
Harsh env.	--	P/C/E ^e	--	--	L	--	--	--
CR100	--	P/S/I/C	X	X	X	X	TM/RE	X
EI. motor	--	S/C/SC	X	--	--	X	--	--
DG avail	--	P/S/PR	--	--	L	X	X	L
DG sellab.	--	P/S/C	--	--	--	--	--	X
DG test	X	--	--	--	--	--	--	--
DG perf.	--	P/S/C/SC	--	--	--	X	--	L
ESFA/Scm.	--	P/S	--	--	--	--	--	--
Baseline	--	P/S/C/SC	--	--	--	--	--	--
HERA	--	P/S/PR	--	--	--	--	TM/OP/RE	--
LER sum.	--	P/S/C/SC/PR	--	--	L	--	TM/OP	--
MRC-DRSS	--	P/S/C	--	X	--	L	--	X
Init. ev.	--	P	L	--	L	--	--	--
IPRDS	--	P/C	--	--	(L)	--	--	--
IEEE-500	--	C	--	--	--	--	--	--
Grey Book	--	P	--	--	--	L	--	L
LER compl.	X	--	--	--	--	--	--	--
Maint. Db	X	P/S/C/PR	--	--	--	(X)	(X)	(X)
NPRDS	--	P/S/C	L	--	L	X	L	L
MPE	--	A	--	--	--	(X)	(X)	--
LERs-NSIC	--	(P/S/C/PR)	L	--	(L)	(L)	(L)	(L)
OPEC-2	--	P/S/C/SC/PR	--	--	X	X	X	X
Pipe bk.	--	P/S/C	--	--	--	--	--	--
WASH-1400	--	C	--	--	--	--	--	--
RCP seals	--	P/C/SC	L	--	L	--	--	L
Syst. U.	--	P/S/(C)	--	--	(X)	(L)	--	--
SO.72 DB	X	--	--	--	--	--	--	--
LER-SCSS	--	(A)	--	(X)	(L)	(L)	(L)	--
Snubbers	--	P/S/C/SC	--	--	--	X	--	L
Syst. Intr.	--	P/S/C	--	(X)	L	(X)	(TM/OP)	(X)
IPRD stat.	--	P/S/C/SC	--	--	--	X	--	--
Tech. spec.	--	P/S/C/PR	--	--	--	(L)	--	--

TABLE 5. (continued)^a

Data Source ^b	None	Level ^c	Aging	Event Sequencing	Root Cause	Design	Human ^d	Performance Indicators
B&I valve	--	C	--	--	I	I	--	--

a. X = yes, -- = no, I = limited. Entries are in parentheses for cases where the capability to obtain aggregates exists (in software) although the aggregates themselves are not tabulated.

b. Acronyms are defined in Table 1.

c. P = plant, S = system, I = train, C = component, SC = subcomponent, PR = personnel, A = all.

d. IM = test and maintenance, OP = operations, RE = recovery.

TABLE 6. DERIVED DATA^a

Data Source ^b	None	Failure Rates	Demand Probabilities	Unavailability			Statistical Analysis			Other Derived Data	Comments
				Corrective Maintenance	Preventative Maintenance	Testing	Error Bounds/Uncertainties	Estimated Probability Distributions	Other		
ASP	--	--	--	--	--	--	X	X	--	X	Statistics on precursors.
RC-aging	--	--	--	--	--	--	--	--	--	X	Root cause fractions.
RC-comp.	--	--	--	--	--	--	--	--	--	X	Root cause fractions.
Harsh env.	--	--	--	--	--	--	X	--	--	X	0-factors for common cause failures.
CREDQ	--	X	X	X	X	--	X	X	--	--	--
El. motor	X	--	--	--	--	--	--	--	--	--	--
DG avail	--	X	X	X	X	X	X	--	--	X	DG failure data versus no test by TI. Overall unavailability data is presented.
DG reliab.	--	X	X	--	--	--	--	--	--	--	DG reliability estimates.
DG test	--	--	--	--	--	X	--	--	--	X	Optional TI.
DG perf.	X	--	--	--	--	--	--	--	--	--	--
ESHA/Scm.	X	--	--	--	--	--	--	--	--	--	--
Baseline	--	X	X	X	--	--	X	X	--	X	Time distributions for DG recovery and LOSP; DG common cause fractions.
HERA	--	--	--	--	--	--	X	--	--	X	Estimate of human error probabilities for specified personnel and specified actions are given. Also included are recovery probabilities.
LER sum.	--	X	X	--	--	--	X	--	--	--	--
NRC-DRSS	--	--	--	--	--	--	--	--	--	X	System performance parameters.

TABLE 6. (continued)^a

Data Source ^b	Missing	Failure Rates	Demand Probability	Unavailability			Statistical Analysis			Other Derived Data	Comments
				Corrective Maintenance	Preventative Maintenance	Testing	Error Bounds/ Uncertainties	Estimated Probability Distributions	Other		
Init. ev.	--	X	--	--	--	--	X	X	--	X	Statistical data on initiating events. Some statistics can be generated by SAS.
IPROS	--	X	X	X	X	--	X	X	--	--	In some cases, environmental factors are provided.
IEEE-500	--	X	X	--	--	--	X	--	--	X	
Grey Book LER comp.	X	--	--	--	--	--	--	--	--	--	
Maint. Db	--	--	--	--	--	--	--	--	--	--	
IPROS	--	X	X	X	X	X	X	--	--	--	
NPE	X	--	--	--	--	--	--	--	--	--	
LEIS, MSIC	X	--	--	--	--	--	--	--	--	--	
OPEC-2	--	(X)	--	(X)	(X)X	(X)	X	X	--	--	Bayesian methods and analysis of variance.
Pipe Bk.	--	X	--	--	--	--	X	X	X	--	Hazard rates, reliability functions, and core-melt probabilities.
WASH-1400 RCP seals	--	X	X	X	X	X	X	X	--	X	
Syst. U.	X	--	--	--	--	--	--	--	--	--	
50.72 DB	X	--	--	--	--	--	--	--	--	--	
LEW-SCSS Snubbers	X	--	--	--	--	--	--	--	--	--	
Syst. Intr.	X	--	--	--	--	--	--	--	--	--	
IPRO stat.	--	X	X	--	--	--	X	X	--	X	Estimate the significance of different parameters on the failure rate.

TABLE 6. (continued)^a

Data Source ^b	Tech. spec.	None	Failure Rates	Demand Probabilities	Unavailability			Statistical Analysis			Comments
					Corrective Maintenance	Preventative Maintenance	Testing	Error Bounds/ Uncertainties	Estimated Probability Distributions	Other Derived Data	
					X	X	X				Forced and scheduled unit unavailabilities.
B&B valve	X										

a. X = yes, -- = no, L = limited. Entries in parentheses are cases when the capability to obtain the derived data exists (in structure) although the derived data itself is not tabulated.

b. Acronyms are defined in Table 1.

TABLE 7. DATA ORIGIN (OPERATIONAL SOURCES)⁴

Data Source	None Included	IER	MPRODS	IPRODS	Test Logs	Control Room Logs	Maint. Records	Operating History Log	LCO Reports	Instr. Rpts.	Tag List	Plant Walk Thru	Operator Interview	Expert Judgment	Lab. Tests
ASP	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
RC aging	--	--	X	--	--	--	--	--	--	--	--	--	--	--	--
RC comp.	--	X	X	--	--	--	--	--	--	--	--	--	--	--	--
Harsh env.	--	X	--	--	X	X	X	X	--	--	X	--	--	--	--
CREDO	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
El. motor	--	X	X	X	--	--	--	--	--	--	--	--	--	--	--
DG avail	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
DG reliab.	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
DG test	--	--	--	--	--	--	X	--	--	--	--	--	--	--	--
DG perf.	--	X	X	--	--	--	--	--	--	--	--	--	--	--	--
ESIA/Scm.	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Baseline	--	X	X	X	--	--	--	--	--	--	--	--	--	X	--
HERA	X	--	--	--	--	--	--	--	--	--	--	--	--	--	--
LER sum.	--	X	--	--	X	--	--	X	--	X	--	--	--	--	X
MRC DRSS	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Init. ev.	--	X	--	--	--	--	--	--	--	--	--	--	X	--	--
IPRODS	--	X	X	X	X	X	X	X	X	--	--	--	X	X	--
IEEE-500	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
Grey Book	--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
LER compl.	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Maint. Db.	--	X	--	--	--	--	X	--	--	--	--	X	--	--	--
MPRODS	X	--	--	--	--	--	--	--	--	--	--	--	--	X	--
NPE	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
LIBS WSIC	--	X	--	--	--	--	--	--	--	--	--	X	--	--	--
OPIC-2	--	X	--	--	--	--	--	X	--	--	--	--	X	--	--
Pipe db.	--	X	--	--	--	--	--	X	--	--	--	--	--	--	X
MASH-1400	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
RCP seals	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Syst. U.	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
SO 72 DB	X	--	--	--	--	--	--	--	--	--	--	--	--	--	--
LER SCSS	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Snobbers	--	X	X	--	--	--	--	--	--	--	--	--	--	X	--
Syst. Intr.	--	X	X	X	--	--	--	--	--	--	--	--	--	--	--
IPRO stat.	--	--	--	X	--	--	--	--	--	--	--	--	--	--	--
Tech spec.	--	X	--	--	--	--	--	--	--	--	--	--	--	--	--

TABLE 7. (continued)^a

Data Source	None Included	LFW	MPDS	IPDS	Test Log	Control Room Logs	Maint. Records	Operating History Log	LCO Reports	Instr. Edg.	Lag List	Plant Walk Thru	Operator Interview	Expert Judgment	Lab. Tests
B&R valve	--	X	--	--	--	--	--	--	--	--	--	--	--	X	--

a. X = yes, -- = no, L = some limited use.

b. Acronyms are defined in Table 1.

c. All data provided by utilities.

TABLE B. DATA ORIGIN (DESIGN SOURCES)^a

Data Source ^b	Non# Included	Test Proc.	Perf. Spec.	ESAR	Equip. List	Instr. Index	P&ID	Elec. Dwg.	Plant Descr. Manual	Vendor Manual	Design Manual	Fault Tree	Environ. Analysis	
													Int.	Ext.
ASP	--	--	--	--	--	--	--	--	--	--	--	X	--	--
RC-aging	X	--	--	--	--	--	--	--	--	--	--	--	--	--
RC-comp.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Harsh env.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
CREED	--	--	--	X	X	X	X	X	X	X	X	--	X	--
El. motor	--	--	--	--	--	--	--	--	--	--	--	--	X	X
OG avail.	--	--	--	--	X	--	--	X	--	X	X	--	--	--
OG rellab.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
OG test	X	--	--	--	--	--	--	--	--	--	--	--	--	--
OG perf.	--	--	--	X	--	--	--	--	--	X	--	--	--	--
ESFA/Scw.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Baseline	X	--	--	--	--	--	--	--	--	--	--	--	--	--
NERA	X	--	--	--	--	--	--	--	--	--	--	--	--	--
LER sum.	--	--	--	X	--	--	X	X	--	--	--	--	--	--
NRC-GRSS	--	X	X	X	--	--	X	--	X	X	--	X	--	--
Int. ev.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
IPRDS	--	X	X	--	X	--	X	X	X	X	X	--	--	--
IEEE-500	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Grey Book	X	--	--	--	--	--	--	--	--	--	--	--	--	--
LER compl.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Maint. Db	--	--	--	X	--	--	--	--	--	--	--	--	--	--
NPRDS	Note c	--	--	--	--	--	--	--	--	--	--	--	--	--
NPE	--	--	--	--	X	--	--	--	--	--	--	--	--	--
LERs-MSIC	X	--	--	--	--	--	--	--	--	--	--	--	--	--
OPEC-2	--	--	--	X	--	--	X	X	--	X	--	--	--	--
Pipe bk.	--	--	--	--	--	--	X	--	--	--	--	--	--	--
WASH-1400	--	--	--	X	--	--	X	--	--	--	--	X	--	X
RCP seals	--	--	--	X	--	--	X	--	--	--	X	--	--	--
Syst. U.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
50.72 DB	X	--	--	--	--	--	--	--	--	--	--	--	--	--
LER-SCSS	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Snubbers	--	--	--	--	--	--	--	--	--	X	--	--	--	--
Syst. intr.	--	--	--	X	--	--	--	--	--	--	--	--	--	--
IPRD stat.	X	--	--	--	--	--	--	--	--	--	--	--	--	--
Tech. spec.	X	--	--	--	--	--	--	--	--	--	--	--	--	--

TABLE 8. (continued)^a

Data b Source	None Included	Test Proc.	Perf. Spec.	ESR	Equip. List	Inst. Index	P&ID	Rec. Log.	Plant Descr. Manual	Vendor Manual	Design Manual	Fault Tree	Environ. Analysis Int. Est.
BAR valve										X			

a. X = yes, - = no, L = limited.

b. Acronyms are defined in Table 1.

c. All data provided by utilities.

TABLE 9. DATA ORIGIN (PRIMARY PUBLISHED SOURCES)

Data Source ^b	None Included	IEEE 500	IEEE Sum	Questionnaire	Other Sources
ASP	X	--	--	--	--
RC-aging	X	--	--	--	--
RC-comp.	X	--	--	--	--
Marsh env.	X	--	--	--	--
CR100	X	--	--	--	--
El. motor	X	--	--	--	--
DG avail	X	--	--	--	--
DG reliab.	X	--	--	--	--
DG test	X	--	--	--	--
DG perf.	X	--	--	--	--
ESIA/Scm. Baseline	X	X	X	--	EPRI documents. Init. ev. report, [PRI MP-2433 and 2301, PRA procedures guide NUREG/CR-2300, PRA of MPPs.
HERA	--	--	--	--	--
LER sum.	--	--	NA	--	--
MRC-DRSS	X	--	--	--	--
Init. ev.	--	--	--	--	EPRI/MP-2230.
IPROS	X	--	--	--	--
IEEE-500	--	NA	X	--	DAC Handbook on Microelectronic Parts, [PRI reports AP-2205, 2071, and 2321.
Grey Book	X	--	--	--	--
LER capt.	X	--	--	--	--
Maint. Db	--	--	--	--	MRC SALP reports, MRC Form 766 violations.
MPROS	X	--	--	--	--
MPE	--	--	--	--	Grey Book, power reactor events.
LIBs-MSC	X	--	--	--	--
OPIC-2	--	--	--	--	Grey Book.
Pipe bk.	--	--	--	X	MPE.
WASH-1400	--	X	--	--	--
RCP seals	--	--	X	--	MSC, MPE, [PRI MP-35]
Syst. U.	X	--	--	--	--
50-72 DB	--	--	--	--	Plant personnel reporting in accordance with CFR requirements.
LER SCSS	X	--	--	--	--
Snubbers	--	--	--	--	Vendor survey, IE Bulletins, and Information Notice.
Syst. Inir.	--	--	--	--	Additional MRC reports and other industry reports.
IPRO stat.	X	--	--	--	--
Tech. spec.	X	--	--	--	--

TABLE 9. (continued)

Data Source ^b	None		Questionnaire	Other Sources
	Included	HEI-500	HEI-500	
BBR valve	---	---	---	NPI, Grey Book; discussions with utilities and valve manufacturers.

a. X = yes, -- = no, NA = not applicable.

b. Acronyms are defined in Table 1.

TABLE 10. DATA ACCESS^a

Data Source ^b	Computerized Sources						Data Report Access Restrictions ^d
	Access Restriction ^c	Fees	Fee Exemption Available	Real Time Access	PC Disk	User's Manual ^d	
ASP	S	--	NA	--	X	S	--
RC-aging	S	--	NA	--	X	--	F
RC-comp.	S	--	NA	--	X	--	F
Marsh env.	NA	NA	NA	NA	NA	NA	S
CREDO	S	--	NA	--	X	X	--
El. motor	NA	NA	NA	NA	NA	NA	--
DG avail	NA	NA	NA	NA	NA	NA	--
DG rellab.	NA	NA	NA	NA	NA	NA	--
DG test	-- ^e	U	U	--	X	X	--
DG perf.	NA	NA	NA	NA	NA	NA	--
ES/A/Scm.	S	--	NA	--	X	X	F
Baseline	NA	NA	NA	NA	NA	NA	--
NERA	S	U	U	--	X	--	--
LER sum.	S	X	--	X	--	N	--
NRC DRSS	S	X	X	X	X	X	--
Intl. ev.	NA ^f	NA	NA	NA	NA	NA	--
IPRDS	R	U	U	--	--	--	--
IEEE-500	NA	NA	NA	NA	NA	NA	--
Grey Book	NA	NA	NA	NA	NA	NA	--
LER compl.	NA	NA	NA	NA	NA	NA	--
Maint. Db	--	--	NA	X	X	X	--
NPRDS	R	X	X	X	X	S	P/R
NPE	--	X	--	X	--	X	--
LERs-NSIC	X	X	--	--	X	X	NA
OPEC-2	X	X	--	X	X	X	P
Pipe bk.	NA	NA	NA	NA	NA	NA	F
WASH-1400	NA	NA	NA	NA	NA	NA	--
RCP seals	NA ^f	NA	NA	NA	NA	NA	--
Syst. U.	S	--	NA	--	X	X	F
SO. 72 DB	X	U	U	X	X	X	NA
LER-SCSS	X	X	--	X	X	P	NA
Snubbers	NA	NA	NA	NA	NA	NA	--
Syst. Intr.	S	NA	NA	--	X	U	--
IPRO stat.	NA	NA	NA	NA	X	NA	F
Tech. spec.	S	NA	NA	--	X	X	F

TABLE 10. (continued)^a

Data Source ^b	Computerized Sources						Data Report Access Restrictions ^d
	Access Restriction ^c	Fees	Fee Exemption Available	Real Time Access	PC Disk	User's ^d Manual	
B&R valve	NA	NA	NA	NA	NA	NA	--

a. X = yes, -- = no or none, NA = not applicable, U = unknown.

b. Acronyms are defined in Table 1.

c. S = approval of sponsor required, R = other restrictions apply.

d. P = proprietary, F = not formally published (few copies exist), S = obtain with approval of sponsor, R = other restrictions apply (see text), U = unknown.

e. This is not really a computerized source. Rather, analysis software exists for the data base and is transferable. The computerized source columns are filled out with regard to the software rather than the data base.

f. Magnetic tapes or other computerized forms of data are obtainable, but no manipulation capabilities exist.

6.2 Probabilistic Risk Assessment Comparisons

In performing a PRA, an enormous amount of input data must be assembled. Estimates of failure rates, unavailabilities, initiator frequencies, test intervals, and a number of other quantities for the plant are required to support the analysis. Just as the level of detail and scope of a PRA can vary, so can the input data. The more plant-specific the data are increase the likelihood that the data accurately characterize the daily operation of the plant.

A plant-specific data source developed to support a PRA has potential use in other PRA applications. The data might also be used in other applications such as screening occurrence frequencies, a detailed component reliability analysis, or a PRA on a similarly designed plant.

Table 11 summarizes PRA studies. Table 12 explains the containment types listed in Table 11. Table 13 describes the levels of analysis listed in Table 11. Most of the PRAs listed in Table 11 were discussed earlier. The table provides a reference for comparing the plants and the scope of the PRAs. As shown in Table 11, the analysis level indicates whether containment response is modeled and whether the propagation of source terms and their impact on the public is modeled.

The main feature of Table 11 pertinent to the bibliography of data sources is the index rating each PRA's operational data. High values of this index flag those PRAs that have the most thorough and traceable plant-specific data development. The index is backed up by the greater detail of Table 14, where the major sources used for data in the PRAs, and their use, are profiled.

In Table 15, the name of each data source entry in Table 14 is indexed to a formal referencing of the data source.

TABLE 11. SUMMARY OF PRA STUDIES

Name and Type of Plant	Thermal Power Rating (MW)	Type of Containment ^a	MSSS Vendor	Architect/Engineer	Month Commercial Operation Began	Sponsor of Study ^b	Performing Organization ^c	Analysis Level ^d	Data Index ^e	PRA Completion Date	Ref. ^f
Arkansas Nuclear 1 PWR/2-loop	836	Large dry (Type 3b)	B&W	Bechtel	12/74	NRC (IREP Phase II)	SNL	2	1	06/82	35
Big Rock Point BWR 2	71	Dry sphere (Type 1)	GE	Bechtel	12/65	Consumers Power Company	WEA	3	2	03/81	36
Browns Ferry 1 ^f BWR-4	1061	Mark I (Type 9)	GE	TVA	08/74	1. NRC (IREP Phase II)	INEL	2	1	06/82	37
						2. TVA	PL&G	3	3	12/84	98
Calvert Cliffs 1 ^f PWR/2-loop	845	Large dry (Type 3b)	CE	Bechtel	05/75	1. NRC (IREP Phase II)	SAI	2	2	03/84	38
						2. NRC (PIS study)	ORNL	h	1	07/85	100
Calvert Cliffs 2 PWR/2-loop	845	Large dry (Type 3b)	CE	Bechtel	04/77	NRC (RSSMAP)	SNL	2	1	05/82	103
Conn. Yankee (Haddam Neck) PWR/4-loop	582	Large dry (Type 3)	W	Stone & Webster	01/68	Northeast Utilities	NU	1	3	12/85	104
Crystal River 3 ^f PWR/2-loop	837	Large dry (Type 3b)	B&W	Gilbert Associates	02/77	1. NRC (IREP Phase I)	SNL	2	1	12/81	105
						2. FPC	SAIC	1	3	02/86	106
Grand Gulf 1 BWR 6	1250	Mark III (Type 5h)	GE	Bechtel	02/83	NRC (RSSMAP)	SNL	2	1	10/81	107
H.B. Robinson 2 PWR/3-loop	665	Large dry (Type 3a)	W	Ebasco	03/71	NRC (PIS study)	ORNL	h	1	08/85	102
Indian Point 2 PWR/4-loop	873	Large dry (Type 3)	W	UE&C	07/74	CONED	PL&G	3	2	12/83	1
Indian Point 3 PWR/4-loop	965	Large dry (Type 3)	W	UE&C	08/76	NYPA	PL&G	3	2	12/83	1

TABLE 11. (continued)

Name and Type of Plant	Thermal Power Rating (MW)	Type of Containment ^a	WSSS Vendor	Architect/Engineer	Month Commercial Operation Began	Sponsor of Study ^b	Performing Organization ^c	Analysis Level ^d	Data Index ^e	PRA Completion Date	Ref. ^f
Elmerick 1, 2 BWR-4	1055 (each)	Mark II (Type 5g)	GE	Bechtel	1985/—	Philadelphia Electric Co.	SAI	3	(1)	09/82	108
Millstone 1 ^f BWR-3	660	Mark I (Type 4g)	GE	Ebasco	12/70	1. NRC (IREP Phase II)	SAI	2	1	01/82	109
						2. Northeast Utilities	MU	1	3	04/85	110
Millstone 3 PWR/4-loop	1150	Sub-atmospheric (Type 3df)	M	Stone & Webster	04/86	Northeast Utilities	M	3	(1)	06/83	111
Oconee 1 PWR/2-loop	860	Large dry (Type 3b)	B&W	Duke Power and Bechtel	07/73	NRC (PTS Study)	ORNL	— ^h	1	1985	101
Oconee 3 ^f PWR/2-loop	860	Large dry (Type 3b)	B&W	Duke Power and Bechtel	12/74	1. NRC (RSSMAP)	SNL	2	1	05/81	112
						2. EPRI and Duke Power Company	NSAC (EPRI)	3	3	06/84	113
Peach Bottom 2 BWR-4	1065	Mark I (Type 4g)	GE	Bechtel	07/74	NRC (WASH-1400)	NRC	3	—	12/75	40
Seabrook 1, 2 PWR/4-loop	1150	Large dry (Type 3g)	M	UEAC	—	PSNH & YAEC	PLRG	3	(1)	12/83	115
Sequoyah 1 PWR/4-loop	1100	Ice condenser (Type 2ce)	M	IWA	03/77	NRC (RSSMAP)	SNL	2	(1)	02/81	116
Shoreham BWR-4	873	Mark II (Type 5g)	GE	Stone & Webster	10/85	ELECO	SAI	3	(1)	05/83	117
Surry 1 PWR/2-loop	775	Sub-atmospheric (Type 3d)	M	Stone & Webster	12/72	NRC (WASH-1400)	NRC	3	—	10/75	40
Yankee Rowe PWR/4-loop	1125	Dry (Type 1)	M	Stone & Webster	07/81	YAEC	EI	3	3	12/82	118
Zion 1, 2 PWR/4-loop	1040	Large dry (Type 3b)	M	Sargent & Lundy	10/73 11/74	Commonwealth Edison	PLRG	3	2	09/81	39

TABLE 11. (continued)

Name and Type of Plant	Thermal Power Rating (MW)	Type of Containment ^a	MSSS Vendor	Architect/Engineer	Month Commercial Operation	Sponsor of Study	Performing Organization ^c	Analysis ^d Level	Date Index ^e	PRA Completion Date	Ref. f
Oyster Creek BWR-2	620	Mark I (Type 4g)	GE	Burns & Roe and GE	12/69	GPU/New Jersey Central Power and Light	PL&G	3	1	08/79	119
Midland PWR/2-loop	467	Large dry (Type 3b)	B&W	Bechtel	Not scheduled	Consumers Power Company	PL&G	3	(1)	09/82	120
Three Mile Island 1 PWR/2-loop	752	Large dry (Type 3b)	B&W	Gilbert	09/74	GPU Nuclear	Unk.	3	3	12/86	Not Pub.

a. For a description of the types of containment see Table 12.

b. Acronyms: IREP, Interim Reliability Evaluation Program; IVA, Tennessee Valley Authority; NCSMAP, Reactor Safety Study Methods Application Program; PFS, Pressurized Thermal Shock; FPC, Florida Power Corp.; COMED, Consolidated Edison; NYPA, New York Power Authority; PSMH, Public Service Company of New Hampshire; YALC, Yankee Atomic Electric Co.; ILECO, Long Island Lighting Co.

c. PL&G, Pickard, Loew, and Garrick; W&A, Word-Leaver & Assoc.; MSAC, Nuclear Safety Analysis Center; SAI, Science Applications Incorporated; SAIC, Science Applications International Corp.; EI, Energy Incorporated. Additional contractors may have participated.

d. For a description of the levels of analysis see Table 13.

e. Index relates to use, and traceability of use, of plant-specific data. An index of 1 indicates little or no use of plant-specific failure data. (1) indicates no use because the plant was pre-operational when the PRA was done. 2 indicates some use of plant-specific data, and 3 indicates that the PRA and supporting documents appear to be a good source for plant-specific data.

f. Reference numbers refer to entries in the References section of this report.

g. Two PRAs were performed on Browns Ferry Unit 1, Calvert Cliffs Unit 1, Crystal River Unit 3, Millstone Unit 1, and Guinee Unit 3.

h. Process leading to vessel overcooling was modeled.

i. Internal document.

j. Because of a lack of detailed information, this PRA is not discussed in the main text.

TABLE 12. CONTAINMENT DESIGN EXPLANATION^a

Containment	Design	Explanation
Dry containment	Type 1	Steel sphere
	Type 2	Steel cylinder
	Type 3	Reinforced concrete cylinder with steel liner
Pressure suppression containment	Type 4	Steel drywell and wetwell
	Type 5	Reinforced concrete drywell and wetwell with steel liner
Primary vessel containment	Type 7	Reinforced concrete pressure vessel
Features	a	Posttensioned vertically only
	b	Posttensioned in three directions
	c	Ice condenser
	d	Subatmospheric
	e	Secondary containment, reinforced concrete shield building, for Types 1 and 2
	f	Secondary containment, steel enclosure building, for Type 3
	g	Secondary containment, concrete and/or steel, for Type 5, 6, and 7
	h	Primary containment following suppression, reinforced concrete with steel liner, for Type 5

a. Classifications obtained from Commercial Nuclear Power Plants, 121

TABLE 13. PRA ANALYSIS LEVELS^a

Level 1 PRA	A level 1 PRA consists of an analysis of plant design and operation, focused on the accident sequences that could lead to a core melt, their basic causes, and their frequencies. The analysis may or may not include external events. The quantitative results of this analysis consist of the frequencies of each core-melt accident. They can be used to derive the core-melt frequency by simply summing the frequencies of the individual sequences.
Level 2 PRA	A level 2 PRA consists of an analysis of the physical processes of the accident and the response of the containment in addition to the analysis performed in a level 1 PRA. Besides estimating the frequencies of core-melt sequences, it predicts the time and mode of containment failure as well as the inventories of radionuclides released to the environment. As a result, core-melt accidents can be categorized by the severity of the release. External events may or may not be included in the analysis. The quantitative results of a level 2 PRA represent an integration of the results obtained in system analysis and in containment analysis. Event trees reflecting consequence distinctions are constructed and quantified in this analysis. As in a level 1 PRA, the product of the sequence-quantification task is a frequency for each event-tree sequence. In addition, the frequency of each plant-damage state may be estimated.
Level 3 PRA	A level 3 PRA analyzes the transport of radionuclides through the environment and assesses the public health and economic consequences of the accident in addition to performing the analyses of a level 2 PRA. The quantitative results of this level of PRA integrate results from the systems analysis, the containment analysis, and the consequence analysis. Complementary cumulative distribution functions (CCDFs) are the most common integrated products of these analyses. The results are generally presented in the form of a CCDF accompanied by a table of sequences whose frequencies are grouped by release category.
Level 4 PRA	A level 4 PRA is like a level 3 PRA, except that in addition a detailed, site-specific analysis of external events is performed.

a. These definitions are from Reference 52.

TABLE 14. USE OF RELIABILITY DATA IN PRASP

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Duration)	Number of Maintenance Actions	Number of Dependent Failures	Failure Rate	Failure Per Demand	Repair Time
<u>Arkansas Nuclear 1^c</u>								
1. WASH-1400	--	--	--	--	--	Modified	Modified	Modified
2. Various NRC and utility letters	--	--	--	--	--	Reviewed	Reviewed	--
3. Plant records								
a) Operating logs	Counted	Estimated	Estimated	Counted	--	Calcu.	Calcu.	Estimated
b) Maint. reports	Counted	--	--	Counted	--	Calcu.	Calcu.	Estimated
c) Test and surveil. reports	Counted	Estimated	Estimated	--	--	Calcu.	Calcu.	--
d) AND IIRs	Counted	--	--	--	--	Calcu.	Calcu.	--
<u>Big Rock Point^c</u>								
1. WASH-1400	--	--	--	--	--	X	X	--
2. IIR-500 (1977)	--	--	--	--	--	X	X	--
3. IIR sum. (pumps, valves)	--	--	--	--	--	X	X	--
4. GE Data (proprietary)	--	--	--	--	--	X	X	--
5. Plant records								
a) Operating logs	Counted	Counted	Counted	Counted	--	Calcu.	Calcu.	Counted
b) Maint. reports	Counted	--	--	Counted	--	Calcu.	Calcu.	Counted
c) Test and surveil.	Counted	Counted	Counted	--	--	--	Calcu.	--
d) Event reports	Counted	--	--	--	X	Calcu.	Calcu.	--
e) Tech. spec.	--	--	X	--	--	--	--	X
<u>Browns Ferry 1 (PRA #1)</u>								
1. IREP update	--	--	--	--	--	X	X	X
2. IIR sum. reports	X	--	--	--	--	X	X	--
3. Transients (EPRI WP-801)	--	--	--	--	--	X	--	--
4. Plant records								
a) Test and surveil.	X	X	--	--	--	--	--	--
b) Tech. spec.	--	X	--	--	--	--	--	X
<u>Browns Ferry 1 (PRA #2)</u>								
1. Misc. generic sources ^b	--	--	--	--	--	X	X	--
2. Nuc. power experience	--	--	--	--	X	--	--	--
3. EPRI transients (WP-2230)	X	X	--	--	--	--	--	--
4. Plant records								
a) Monthly operating reports	X	--	--	--	X	--	--	--
b) Surveillance instructions	--	X	X	--	--	X	X	--
c) Hold tags	--	--	--	X	--	--	--	X
d) IIRs	X	--	--	--	X	--	--	--

TABLE 14. (continued)^a

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Duration)	Number of Maintenance Actions	Number of Dependent Failures	Failure Rate	Failure Per Demand	Repair Time
<u>Calvert Cliffs 1 (PRA #1)</u>								
1. IHERP (human error)	--	--	--	--	--	X	X	--
2. IREP update	--	--	--	--	--	X	X	--
3. Plant records								
a) IERs	Counted	--	--	--	--	X	--	--
b) Maintenance records	Counted	--	--	X	--	X	--	X
c) Tech. specs.	--	X	X	--	--	--	--	--
d) Testing procedures	--	X	X	--	--	--	--	--
e) Plant interviews	--	--	--	Interview	--	--	--	--
<u>Calvert Cliffs 1 (PRA #2--Pressurized Thermal Shock)</u>								
1. Baseline data	X	X	--	--	X	X	X	--
2. NUREG/CR-1460	X	X	--	--	X	X	X	--
3. 1980 operating exper.	X	X	--	--	X	X	X	--
4. Precursors	--	--	--	--	--	X	X	--
5. LOCA/FW trans.	--	--	--	--	--	X	X	--
6. Plant records	X	X	--	--	X	X	X	--
<u>Calvert Cliffs 2</u>								
1. WASH-1400	--	--	--	--	--	X	X	X
<u>Connecticut Yankee (Haddam Neck)</u>								
1. Westinghouse data (proprietary)	--	--	--	--	--	X	X	--
2. Plant records								
a) Incident reports	Counted	Counted	Counted	Counted	Counted	Calcu.	Calcu.	--
b) Shift logs	Counted	Counted	Counted	Counted	Counted	Calcu.	Calcu.	Counted
c) Maint. cards	Counted	Counted	Counted	Counted	--	Calcu.	Calcu.	Counted
d) BEARDS ^d	Counted	--	--	--	Counted	Calcu.	Calcu.	--
e) Plant procedures	--	X	X	X	--	--	--	--
f) Test records	--	Counted	Counted	Counted	--	--	--	--
<u>Crystal River 3 (PRA #1)</u>								
1. IREP update	--	--	--	--	--	X	X	X
2. IER summary pumps	--	--	--	--	--	X	X	--
3. Common cause (pumps, DG)	--	--	--	--	X	--	--	--

TABLE 14. (continued)^a

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Days/Year)	Number of Maintenance Actions	Number of Dependent Failures	Failure Rate	Failure Per Demand	Repair Time
<u>Crystal River 3 (PRA #1) (continued)</u>								
4. IHERP (human error)	--	--	--	--	--	X	--	--
5. Plant records	X	--	--	--	--	--	--	--
a) IIRs	--	--	--	X	--	--	--	X
b) tech. specs.	--	--	--	X	--	--	--	X
c) Interviews	--	--	X	--	--	--	--	--
<u>Crystal River 3 (PRA #2)</u>								
1. WASH-1400	--	--	--	--	--	X	X	--
2. IEEE-500 (1984)	--	--	--	--	--	X	X	--
3. LEM summary (pumps, valves)	--	--	--	--	--	X	X	--
4. Plant records	--	--	--	--	--	--	--	--
a) Maint. history file	Counted	Counted	--	--	--	Calcu.	Calcu.	--
b) NPSDS	Counted	Counted	--	--	--	Calcu.	Calcu.	--
c) IIRs	Counted	Counted	--	--	--	Calcu.	Calcu.	--
d) Questionnaire to FFZ	--	--	--	--	--	--	--	--
e) Tag file	--	--	--	X	--	--	--	--
<u>Grand Gulf 1C</u>								
1. WASH-1400	--	--	--	--	--	X	X	X
<u>M. R. Edlinson 2 (Pressurized Thermal Shock)</u>								
1. Baseline data	X	X	--	--	X	X	X	--
2. NUREG/CR-1460	X	X	--	--	X	X	X	--
3. Precursors	--	--	--	--	--	X	X	--
4. LOCA/TM trans.	--	--	--	--	--	X	X	--
5. 1980 operating experience	X	X	--	--	X	X	X	--
6. Plant records	X	X	--	--	X	X	X	--
<u>Indian Point 2 and 3</u>								
1. WASH-1400	--	--	--	--	--	X	X	--
2. IEEE-500 (1977)	--	--	--	--	--	X	X	--
3. NPSDS (1977)	--	--	--	--	--	X	X	--
4. EPRI transients (MP-601)	X	X	--	--	--	--	--	--
5. IIR sum. - pumps, valves, DG	--	--	--	--	--	X	X	--
6. IHERP (human error)	--	--	--	--	--	--	--	--
7. Plant records	--	--	--	--	--	--	--	--
a) IIRs	X	X	--	--	--	--	--	--
b) Other records	--	--	--	X	--	--	--	X

TABLE 14. (continued)^a

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Duration)	Number of Maintenance Actions	Number of Dependent Failures	Failure R _r	Failure Per Demand	Repair Time
<u>Time-1ck^c</u>								
1. Peach Bottom logs, IERs, maintenance records, specs	Counted	Estimated	Calcu.	Counted	--	Calcu.	Calcu.	Calcu.
2. IER sum. (pumps, DG, valves)	--	--	--	--	--	X	X	--
3. GE data (proprietary)	Counted	Estimated	Counted	Estimated	Reviewed	--	--	--
4. WASH-1400	--	--	--	--	--	X	X	X
5. ATWS (NUREG-0460)	--	--	--	--	--	--	X	--
<u>Milestone 1 (PRA #1)</u>								
1. IREP update	--	--	--	--	--	X	X	--
2. IHERP (human error)	X	--	--	--	--	--	X	--
3. Plant records								
a) Personnel interviews	--	X	X	X	--	--	--	X
b) test and maint. proc.	--	--	X	X	--	--	--	--
c) IERs	X	--	--	--	--	--	--	--
<u>Milestone 1 (PRA #2)</u>								
1. WASH-1400	--	--	--	--	--	X	X	--
2. Plant records								
a) Plant incident reports	Counted	Counted	Counted	Counted	Counted	Calcu.	Calcu.	--
b) Shift logs	Counted	Counted	Counted	Counted	Counted	Calcu.	Calcu.	Counted
c) Maint.	Counted	--	--	Counted	--	Calcu.	Calcu.	Counted
d) Brkr. cycle counters	--	Counted	--	--	--	--	Calcu.	--
e) BEARDS ^d	Counted	--	--	--	Counted	Calcu.	Calcu.	--
f) Plant procedures	--	--	X	X	--	--	--	--
g) test records	--	Counted	Counted	Counted	--	--	--	--
<u>Milestone 3</u>								
1. Westinghouse data (proprietary)	--	--	--	--	--	X	X	X
2. WASH-1400	--	--	--	--	--	X	X	--
3. Baseline data	--	--	--	--	--	X	X	X
4. IEEE-500 (1977)	--	--	--	--	--	X	X	--
<u>Oconee 1 (Pressurized Thermal Shock)</u>								
1. Baseline data	X	X	--	--	X	X	X	--
2. NUREG/CR-1460	X	X	--	--	X	X	X	--
3. 1980 operating exper.	X	X	--	--	X	X	X	--
4. Precursors	--	--	--	--	--	X	X	--
5. LOCA/FW trans.	--	--	--	--	--	X	X	--
6. Plant records	X	X	--	--	X	X	X	--

TABLE 1A. (continued)

Source of Reliability Data ^a	Number of Failures	Number of Demands or Exposures Time	Test Interval (hours)	Number of Maintenance Actions	Number of Independent Failures	Failure Rate	Failure Per Period	Repair Time
Process 2 (PRA #1)								
1. WSS-1400	—	—	—	—	—	X	X	X
Process 2 (PRA #2)								
1. Misc. generic sources ^b	—	—	—	—	—	—	—	—
2. Misc. power experience	X	—	—	—	—	X	X	—
3. Transients (EPRI NP-2750)	X	X	—	—	—	—	—	—
4. LERs	X	—	—	—	—	—	—	—
5. Plant records	—	—	—	—	—	—	—	—
a) Work requests	X	—	—	X	—	X	X	X
b) Incident reports	X	—	—	—	—	X	X	—
c) Periodic tests	—	X	X	—	—	X	X	—
d) Control room operating logs	—	X	—	—	—	X	X	—
Process Bottom 2								
1. WSS-1400 sources	—	—	—	—	—	—	X	X
Seawater								
1. Misc. generic sources ^b	—	—	—	—	—	—	—	—
2. Misc. power experience	X	—	—	—	—	—	X	—
3. Transients (EPRI NP-2750)	X	X	—	—	—	—	—	—
4. PIG-500 (proprietary)	X	X	—	X	—	X	X	—
5. Plant test procedures	—	—	X	—	—	—	—	—
Seawater 1								
1. WSS-1400	—	—	—	—	—	—	X	X
Seawater 2								
1. LERs (1973)	—	—	—	—	—	—	—	—
2. LERs (pumps, valves, etc.)	—	—	—	—	—	—	—	—
3. WSS-1400	—	—	—	—	—	—	X	—
4. Transients (EPRI NP-803)	X	X	—	—	—	—	—	—
5. ATRC (NRC 6-0446)	—	—	—	—	—	—	—	—
6. ATRC (WSS-1275)	X	—	—	—	—	—	—	—
7. Seams (ALB-78)	X	—	—	—	—	—	—	—
8. Pipe failures (EPRI NP-430)	—	—	—	—	—	—	—	—
9. Liner back PRA	—	—	—	—	—	—	—	—

TABLE 14. (continued)^a

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Duration)	Number of Maintenance Actions	Number of Dependent Failures	Failure Rate	Failure Per Demand	Repair Time
Surry 1								
1. WASH-1400 sources	--	--	--	--	--	X	X	--
2. Plant records	--	--	--	--	--	--	Calcu.	--
a) Tech. specs.	--	X	X	--	--	--	Calcu.	--
b) LFRs	Counted	--	--	--	Counted	Calcu.	Calcu.	--
c) Discussions at plant	--	X	X	X	--	--	Calcu.	X
Yankee Rowe								
1. WASH-1400	--	--	--	--	--	X	X	--
2. IEEE 500 (1977)	--	--	--	--	--	X	X	--
3. IEEE Std-493	--	--	--	--	--	X	X	--
4. NRPDS (1979)	--	--	--	--	--	X	X	--
5. GADS (NRC)	X	X	--	--	--	X	--	--
6. WCC (NPRD-1)	--	--	--	--	--	X	--	--
7. MIL-HBK-217C	--	--	--	--	--	X	--	--
8. Data Systems (EPRI NP-1064)	--	--	--	--	--	X	--	--
9. Nonelectronic rel. notebook	--	--	--	--	--	X	--	--
10. Plant records	--	--	--	--	--	--	--	--
a) Tech. ser. department	Counted	Counted	--	--	--	Calcu.	Calcu.	--
b) Maint. department	Counted	Counted	--	--	--	Calcu.	Calcu.	--
c) I&C department	Counted	Counted	--	--	--	Calcu.	Calcu.	--
d) Reactor engineering	Counted	Counted	--	--	--	Calcu.	Calcu.	--
Zion^c								
1. WASH-1400	--	--	--	--	X	X	X	X
2. NRPDS	--	--	--	--	X	--	X	X
3. LFR sum. pumps	Counted	Counted	--	--	--	X	X	--
DG	Counted	Estimated	--	--	--	X	X	Reviewed
valves	Counted	Estimated	--	--	--	X	X	--
4. IEEE-500 (1977)	--	--	--	--	--	X	X	--
5. Plant records	--	--	--	--	--	--	--	--
a) Zion technical specifications	--	Estimated	X	Estimated	--	--	--	Estimated
b) Test procs.	--	Calcu.	Counted	--	--	Calcu.	Calcu.	--
c) Testing records	--	Calcu.	Counted	--	--	Calcu.	Calcu.	--
d) LFRs	Counted	--	--	--	X	Calcu.	Calcu.	--
e) Deviation reports	Counted	--	--	--	X	Calcu.	Calcu.	--
f) Operating records	--	Calcu.	--	--	--	Calcu.	Calcu.	Calcu.
g) Out-of-service card log	Counted	--	--	Counted	--	Calcu.	Calcu.	Counted

TABLE 14. (continued)^a

Source of Reliability Data ^b	Number of Failures	Number of Demands or Exposure Time	Test Interval (Duration)	Number of Maintenance Actions	Number of Dependent Failures	Failure Rate	Failure Per Demand	Repair Time
Oyster Creek^c								
1. WASH-1400	--	--	--	--	--	X	X	X
2. IEEE 500 (1977)	--	--	--	--	--	X	X	--
3. NRCDS (1976)	--	--	--	--	--	X	--	--
4. feed pump (EPRI EP-154)	--	--	--	--	--	X	--	--
5. Rectic. pump (EPRI MP-351)	--	--	--	--	--	X	--	--
6. DG experience (ORR-ES-002)	--	--	--	--	--	X	--	--
7. Plant records	X	X	X	X	--	X	X	X
Midland^c								
1. Misc. generic sources ^b	--	--	--	--	--	X	X	--
2. Nuclear power experience	X	--	--	--	X	--	--	--
3. Transients (EPRI MP-2230)	X	X	--	--	--	--	--	--
4. PG-500 (proprietary)	X	X	--	X	X	X	X	X
5. Plant test procedures	--	--	X	--	--	--	--	--
Three Mile Island^c								
1. Misc. generic sources ^b	X	--	--	--	--	X	X	--
2. Nuclear power experience	X	--	--	--	X	--	--	--
3. PG-500 (proprietary)	X	X	--	X	X	X	X	X
4. Dependent (EPRI MP-3967)	--	--	--	--	X	--	--	--
5. Transients (EPRI MP-2230)	X	X	--	--	--	--	--	--
6. Plant records	X	--	--	--	--	--	--	--
a) Job tickets	X	--	--	--	--	--	--	--
b) Work request	X	--	--	--	--	--	--	--
c) Run time meter log	--	X	--	--	--	--	--	--
d) Maint. request	X	--	--	--	--	--	--	--
e) Switching and lagging order	X	--	--	X	--	--	--	X
f) Monthly operating records	X	X	--	X	--	--	--	X
g) Weekly reports	X	X	--	--	--	--	--	--
h) Test and operating procedures	--	--	X	--	--	--	--	--

a. X = yes, source used for application; -- = no, source not used for the application. In some cases, more detail about the use is provided in place of the "X".

b. See table 15 for more information on the sources other than plant records.

c. The data source information for this entry was obtained from EPRI MP-3265, 122.

d. BARSIS: Baseline Events Analysis/Reliability Data System (Northeast Utilities, 1975).

e. A description for this PRA was not available; therefore it does not appear in the PRA Descriptions section.

TABLE 15. PRA GENERIC DATA SOURCES

Source ^a	Subject	Reference ^b	Described Elsewhere in This Bibliography
WASH-1400	PRA of Peach Bottom 1 and Surry 1	40	Section 2.29
IEEE-500	Component failure rates	41, 61	Section 2.20 ^c
LER summaries	Component failure rates	28-34, 55, 56	Section 2.16
GE data	Component failure rates	Proprietary	--
IREP update	Component failure rates	26	See Section 2.14
Transients (EPRI NP-801)	Initiating events	44	x ^d
Misc. generic sources	See Note e	--	See NPRDS data section (Section 2.24); also RAC's NPRD-1 (Sec. 3.5) and MIL-HDBK-217 (Sec. 3.4) are under nonnuclear data bases.
Nuclear power experience	Events at NPP	45	Section 2.25
Transients (EPRI NP-2230)	Initiating events	18	Sections 2.12, 2.18
THERP (human error) (NUREG/CR-1278)	Human error	46	--
Baseline data (EGG-FA-5887)	Component failure rates	23	Section 2.14
NUREG/CR-1460	LWR safety research program	123	--
1980 operating experience (NUREG/CR-2378)	US NPP status in 1980	63	See Grey Book (Sect. 2.21)
Precursors (NUREG/CR-2497, NUREG/CR-3591)	Precursors to severe core damage accidents (1969-1979, 1980-1981)	4 5	Section 2.1
LOCA/FW trans. (NUREG/CR-0635)	Small-break LOCA and transients in CE plants	124	--
Westinghouse data	Component failure rates	Proprietary	--
Common cause	Common cause factors for pumps; for DG	125 50	--

TABLE 15. (continued)

Source ^a	Subject	Reference ^b	Described Elsewhere in This Bibliography
PLG-500	Component failure rates	Proprietary	--
NPRDS (1977)	Component failure rates	77	Section 2.24
ATWS (NUREG/0460)	Anticipated transients without scram	126	--
ATWS (WASH-1270)	Anticipated transients without scram	127	--
Scrams (ALO-7B)	Failures causing scrams	128	--
Pipe failures (EPRI NP-438)	Pipe system failures in LWRs	129	--
IEEE Std-493	Design of reliable power systems	130	--
NPRDS (1979) (NUREG/CR-1635)	Component failure rates	76	Section 2.24
GADS (NERC)	Plant availability	91	Section 3.2
RAC (NPRD-1)	Component failure rates	131	Section 3.5
MIL-HBK-217C	Component failure rates	93	Section 3.4
Data systems (EPRI NP-1064)	Utility-industry data systems	132	--
Nonelectronic reliability	Component reliability	133	Section 3.5
NPRDS (1976)	Component failure rates	77	Section 2.24
Feed pump (EPRI FP-754)	Feed pump outages	134	--
Recirc. pump (EPRI NP-351)	Recirculation pump seals	135	--
DG experience (OGE-ES-002)	DG experience before 1974	136	See more current DG studies (Sections 2.6-2.9)

TABLE 15. (continued)

Source ^a	Subject	Reference ^b	Described elsewhere in this bibliography
Dependent events (IPRI MP-3963)	Reactor experience involving dependent events	131	No, but see system interaction section (2.35)
<p>a. Sources are listed in the same order as they appear in Table 14.</p> <p>b. Reference numbers refer to entries in the References section of this report.</p> <p>c. IEEE Std-500 (1984), discussed with the nuclear sources, is an update of IEEE Std-500 (1977).</p> <p>d. IPRI MP-2230, discussed in the IPRI Reports Section of the bibliography, is an update of this document.</p> <p>e. The miscellaneous generic source list includes WASH-1400, the LTR summaries for pumps, valves, and diesel generators, IEEE-500 (1977), and the following reports:</p> <p>Equipment Availability Task Force of the Prime Movers Committee, Edison Electric Institute, <u>Equipment Availability Component Cause Code Summary Report for the Ten Year Period 1967-1976</u>, supplement to EII publication 77-64, January 1978.</p> <p>Southwest Research Institute, <u>Nuclear Plant Reliability Data System 1979 Annual Report of Cumulative System and Component Reliability</u>, NUREG/CR-1635, September 1980.</p> <p>Combustion Engineering, Inc., <u>Nuclear Power Systems, Component Failures at Pressurized Water Reactors</u>, AID-74, October 1980.</p> <p>Reliability Analysis Center, Rome Air Development Center, <u>Monoelectronic Parts Reliability Data</u>, MPRI-1, Summer 1978.</p> <p>Hughes Aircraft Company, <u>RADC Unanalyzed Monoelectronic Part Failure Rate Data Interim Report (HIDC9) No. 1</u>, AD-806546, December 1966.</p> <p>T. R. Moss and L. M. Youdell, <u>Reliability Data on Electronic Components from the Harwell Series of Nuclear Equipment</u>, National Centre of Systems Reliability, UK-1A, MCSB-83, 1975.</p> <p>Southwest Research Institute, <u>Generation and Component Data Usage in Availability Engineering</u>, IPRI MP-81-2 ID, February 1981.</p> <p>G. M. Hannaman, <u>GCR Reliability Data Bank Status Report</u>, General Atomic Company, GA-A14839, UC-77, July 1978.</p> <p>ARMIC Research Corporation, <u>Reliability Engineering</u>, W. H. Von Alven, ed., New York: Prentice-Hall, Inc., 1964.</p> <p>U.S. Department of Defense, <u>Military Standardization Handbook: Reliability Prediction of Electronic Equipment</u>, MIL-HDBK-217C, April 1979.</p>			

TABLE 15. (continued)

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7. FUTURE DATA SOURCES

A number of data sources were identified before this document was written but are not included for two reasons:

1. The data source was still in a developmental stage and had yet to be published
2. The information required to adequately describe the source was not obtainable before this document's publication.

Table 16 lists of these data sources with a brief description of each source's focus. It is anticipated that these data sources will be included in an update of this document.

TABLE 16. FUTURE BIBLIOGRAPHY DATA SOURCES

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1. Accident Sequence Evaluation Program (ASEP) data base--Data includes demand and time related component failure probabilities and unavailabilities; also included are initiator frequencies, common cause dependencies and human error estimates. Uncertainties are given for each point estimate.
 2. Risk Methods Integration Evaluation Program (RMIEP) Recovery Data--The source provides probabilistic estimates for diagnosis of operator recovery action following abnormal events. The data originates from eighteen simulator experiments conducted at the LaSalle plant simulator.
 3. Power Reactor Information System (PRIS)--A computerized data source of operational statistics from nuclear power plants in member countries of the International Atomic Energy Agency.
 4. OECD-Nuclear Energy Agency Incident Reporting System (NEA-IRS)--A computerized system in which abnormal occurrences are collected and analyzed.
 5. IAEA-IRS--A reporting system analogous to the NEA-IRS, but with a different set of participants. Included in the groups reporting are Soviet countries and East European countries.
 6. Union International des Producteurs et Distributeurs d' Energie Electrique Significant Event Reporting System--An on-line reporting and interrogation system for significant events occurring at nuclear power plants.
 7. RMIEP PRA (LaSalle)--A state-of-the-art PRA.
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BIBLIOGRAPHIC DATA SHEET

NUREG/CR-5050
EGG-REQ-7827

SEE INSTRUCTIONS ON THE REVERSE

2. TITLE AND SUBTITLE

Annotated Bibliography of Reliability
and Risk Data Sources

5. AUTHOR(S)

O. V. Hester
S. R. Brown
C. D. Gentillon

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH YEAR

September 1987

6. DATE REPORT ISSUED

MONTH YEAR

March 1988

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

EG&G Idaho, Inc.
Idaho Falls, ID 83415

8. PROJECT/TASK/WORK UNIT NUMBER

9. PIN OR GRANT NUMBER

A5237

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Safety Programs
Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11. TYPE OF REPORT

Technical

12. PERIOD COVERED (Indicate dates)

13. SUPPLEMENTARY NOTES

14. ABSTRACT (200 words or less)

This document is an annotated bibliography of nuclear, nonnuclear, and foreign data sources that are useful in nuclear power plant reliability and risk analysis applications. A brief description of the contents, areas of usefulness, access information, and the name and address of a contact is provided for data sources of all types. In addition, for nuclear data sources, tabular comparisons are made. These comparisons include the scope of the data sources; their operational, special-purpose operational, pedigree, aggregated, and derived data; the operational and design data each data source originates from; and access information.

Probabilistic risk assessments (PRAs) are profiled separately. For each PRA, background information that describes the PRA and the plant itself is provided. Also the input data sources used to support each PRA are identified. Of special interest is the identification of unique plant-specific data sources that have evolved from the PRA analyses. To the extent possible, how the data sources were used in the analysis is also discussed.

15. DOCUMENT ANALYSIS - KEYWORDS DESCRIPTORS

Data bases
Risk Assessment

16. IDENTIFIERS/OPEN-ENDED TERMS

17. AVAILABILITY STATEMENT

Unlimited

18. SECURITY CLASSIFICATION

This paper

Unclassified

This report

Unclassified

19. NUMBER OF PAGES

20. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, \$300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

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