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Identification of Equipment and Components Predicted as Significant Contributors to Severe Core Damage

Harry W. Heiselmann

May 1984

Prepared for the

U.S. Nuclear Regulatory Commission

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IDENTIFICATION OF EQUIPMENT AND COMPONENTS PREDICTED AS SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

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ABSTRACT

The Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, sponsored the Equipment Qualification Research Program which performed a survey of applicable severe accident study reports to aid in focusing the program efforts. The objective of the survey was to identify, where possible, equipment and components that have been predicted to be significant contributors to high probability accident sequence resulting in severe core damage. A summary of the results of the survey is presented in Tables 1 and 2 of this report. Future updates of this report are anticipated as applicable risk study reports become available.

NRC FIN No. A6322—Equipment Qualification Research Program

SUMMARY

To aid in focusing the efforts of the Nuclear Regulatory Commission (NRC)-sponsored Equipment Qualification Research Program, a study (Identification of Equipment and Components Predicted as Significant Contributors to Severe Core Damage) of available reference reports pertaining to severe core damage situations was performed. The purpose of the study is to identify equipment and components which, based on the reference reports, are predicted to be significant contributors to the dominant severe core damage accident sequences.

No reference reports could be identified which address risk directly and also provide the capability to identify specific equipment and components. As a result, the available reports which were used in the study address dominant or most probable accident sequences up to the point of severe core damage. A listing of the reference reports utilized is presented in Section 6.

The Accident Sequence Evaluation Program (ASEP) and Accident Sequence Precursor (ASP) study reports are the primary references that were used. ASEP utilizes the probabilistic risk assessments (PRA) for various plants to determine the likelihood of severe core damage and evaluate the significant contributors for each dominant accident sequences (DAS) on a plant by plant basis. Approximately 10 to 14 DAS were analyzed for each of the six pressurized water reactor (PWR) plants and four boiling water reactor (BWR) plants included in the study.

The ASP studies utilize the licensee event report (LER) as the basis for the accident initiating (precursor) event. A series of subsequent events are then assumed, and an event tree is generated using the precursor as the initiating event. Those sequences resulting in predicted severe core damage were then used to identify the equipment and components contributing to each sequence.

A number of other references were also reviewed. Although the majority of these references provided support and confirmation of the equipment and components identified from the ASEP and ASP reports, they did not provide sufficient information to identify specific equipment and components.

Tables 1 and 2 present a summary listing of the equipment and components identified as predicted significant contributors to severe core damage. Categories of valves, pumps, electrical, instruments, maintenance and test, human error, and weather have been used to group the findings. The non-equipment categories of maintenance and test, human error, and weather have been included to provide interface information on how equipment and component performance can be affected. A discussion in this regard is provided in Section 4.

Tables 1 and 2 also provide a coarse comparison of the equipment and components identified by the ASEP and ASP studies. Similar equipment and components have, in general, been identified by both studies.

Because both the ASEP and ASP studies are limited to severe core damage, and because results of reference studies relating severe core damage to risk are not completed, certain equipment and components which would be identified between severe core damage and risk (release from containment), such as purge vent and containment isolation valves, have not been identified by this study. Additional work will be required on this study to include these types of equipment and components in updated versions of this report. This can be accomplished as soon as appropriate reports from such NRC sponsored programs as the Severe Accident Sequence Analysis (SASA) and Severe Accident Risk Reduction Program (SARRP) are available.

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EXHIBITS

ACRONYMS AND ABBREVIATIONS

AC	Alternating current
AFW(S)	Auxiliary feedwater (system)
ANO-1	Arkansas Nuclear-One
ASEP	Accident Sequence Evaluation Program
ASP	Accident Sequence Precursor
ATWS	Anticipated transient without scram
BF-1	Browns Ferry-One
BNL	Brookhaven National Laboratory
BWR	Boiling water reactor
CC-2	Calvert Cliffs-Two
CD/CM	Core damage and/or core melt
CR-3	Crystal River-Three
CV	Control valve
DAS	Dominant accident sequences
DC	Direct current
DG	Diesel generator
ΔP	Differential pressure
e/c	Equipment and component
EFS	Emergency feedwater system
EFW	Emergency feedwater
EGG	EG&G Idaho, Inc.
ESF	Engineered safety features
EQRP	Equipment Qualification Research Program
FW	Feedwater
GG-1	Grand Gulf-One
HPCI	High pressure coolant injection

HPI(S)	High pressure injection (system)
HPR(S)	High pressure recirculation (system)
HPSWS	High power service water system
ICM	Isolation condenser makeup
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operation
IREP	Interim Reliability Evaluation Program
LER	Licensee event report
LOCA	Loss of coolant accident
LOFW	Loss of feedwater
LOOP	Loss of offsite power
LPRS	Low pressure recirculation system
LRTR	Locked rotor
LWR	Light water reactor
MOV	Motor operated valve
MP-1	Millstone Point-One
MSLB	Main steam line break
N_2	Nitrogen
NNI	Non-nuclear instrumentation
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
OC-3	Oconee-Three
ORNL	Oak Ridge National Laboratory
P	Pressure
PB-1	Point Beach-One
PC(B)	Printed circuit (board)
PCS	Primary coolant system

PORV	Power operated relief valve
POV	Power operated valve
PP	Precursor probability
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RBCS	Reactor building cooling system
RBSIS	Reactor building spray injection system
RC	Recirculation
RCIC	Reactor coolant isolation cooling
RCS	Reactor coolant system
RHR	Reactor heat removal
RPS	Reactor protection system
SARRP	Severe Accident Risk Reduction Program
SASA	Severe Accident Sequence Analysis
SC	Significance category
SCD	Severe core damage
SE-1	Sequoyah-One
SLCS	Standby liquid control system
SRV	Safety relief valve
SSMRP	Seismic Safety Margins Research Program
SSWS	Standby service water system
SU-1	Surry-One
T&M	Test and Maintenance
TMI-2	Three Mile Island-Two
VAC	Volts alternating current
VDC	Volts direct current

IDENTIFICATION OF EQUIPMENT AND COMPONENTS PREDICTED AS SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

INTRODUCTION

This study is sponsored by the Nuclear Regulatory Commission to aid in focusing the Equipment Qualification Research Program (EQRP) efforts toward equipment and components (e/c) that have been predicted to be significant contributors to the probability of severe core damage (SCD) or core damage and core melt (CD/CM) situations for dominant accident sequences (DAS). This study has been limited to identifying equipment and components to SCD or CD/CM situations because reference reports relating e/c to consequences and/or risk are not complete and/or available. The CD/CM or SCD restrictions imposed on this study by the unavailability of risk-oriented reference reports and limited data for the reference reports will result in the omission of some critical containment integrity e/c items such as: containment purge-vent and isolation valves. Subsequent EQRP research efforts will use the results of this study to focus on specific equipment and develop recommendations for qualification studies aimed at upgrading qualification methods to improve e/c performance and reduce their contribution to DAS.

INFORMATION SOURCES

Available reports for severe accident studies were used to identify the e/c predicted to be significant contributors to CD/CM. A list of these reference reports is presented in Section 6. The Accident Sequence Evaluation Program (ASEP) draft report for work performed at the Idaho National Engineering Laboratory (INEL)[1] and the Accident Sequence Precursor (ASP) status reports[2,3] provided the major portion of the information for this study. These reports were used as the primary references because, in general, they included sufficient information to:

- Provide identification of e/c
- Provide a coarse comparison between the e/c identified by analysis (ASEP) and experience (ASP)
- Provide information for a sufficient number of plants to permit the results to be considered reasonably representative of all plants. For example, ASEP analysis was for six pressurized water reactors (PWRs) and four boiling water reactors (BWRs) as opposed to the Seismic Safety Margins Research Program (SSMRP) which is plant specific to Zion. The ASP studies address Licensee Event Reports (LERs) from a large cross section of plants.

Other reference reports listed in Section 6 were reviewed for content to appraise usefulness and applicability to this study and to see if they provided results essentially in agreement with the ASEP and ASP information. Most of these reports contained information supportive of the ASEP and ASP results but did not provide sufficient information to permit identification of specific e/c.

EQUIPMENT AND COMPONENT IDENTIFICATION METHOD

To aid in focusing the EQRP efforts, a composite summary comparing the significant e/c identified using the ASEP and ASP study results was prepared for PWRs and BWRs. This summary is presented as Table 1 (PWR systems) and Table 2 (BWR systems) of this report. The methods used by this study to identify the listed e/c are discussed in detail in Appendixes A and B.

Briefly, the e/c identified using the ASEP study results were determined on a plant by plant basis for each DAS identified for each plant. A numerical

Table 1. Summary: predicted significant contributors to core damage and/or core melt sequences for PWR systems

Equipment Component Category	Based on ASEP Analysis[a]	Based on Precursor (LERs) Studies[b]
Valves	• Motor operated valves (MOVs) (6)[c] • Primary relief valves (3) • Water isolation valves (1) • Boron injection control valves (1) • 3 way vaive (1)	• Power operated relief valves (PORVs) (1)[c] • Primary relief valve (stuck open) (1) • Sump isolation valves (3) • Main steam isolation valve (1) • Steam relief valve (1) • Steam dump valve operator (1) • Solenoid valves; failed windings (1)
Pumps	• Feedwater turbine (auxiliary and emergency) (3) • Centrifugal; service water (2) • Vacuum; service water (2) • High pressure injection (HPI) (1)	• Feedwater (tight packing and binding) (2) • Feedwater; turbine, (fail to start) (1) • Safety injection (failure) (1) • Reactor coolant (shaft broke and seal) (2)
Electrical	• Cables; open circuit and wire fault (2) • Circuit breakers; fault and operation (2) • Control circuit; MOVs, pumps (3) • Power busses; common mode (1) • Batteries; low power (1) • Backup alternating current (AC) (1)	• Cables; unconnected (1) • Circuit breakers; opens, fails, balance loading (3) • Relays; open, reset fails spurious operation (3) • Controllers; pressurizer feedwater and auxiliary feed (3) • Power busses; deenergized (1) • Transformer; site and recirculation (2) • Inverter; PORV, steam dump (2) • Power supply; HPIS + boron injection (2)
Instruments	• Thermostat fail to close (1) • Thermocouple failure (1)	• Non-nuclear instrumentation (NNI) insufficient auxiliary feedwater (AFW) flow; calibration (2) • NNI direct current (DC) power supply failure; buffer card (1)
Systems and miscellaneous equipment	• Diesel generators; fail to start or run (2) • Primary coolant system; fail to recover or run (2) • Reactor protection system; fail to terminate fission and hardware fail (2) • Chilled water system; fail to start/run (1) • Emergency recirculation system failure (1) • Boron inject heat system failure (1)	• Diesel generator trip load imbalance (1) • Clogged feedwater strainers (2) • Leaking service water pipe flange (1) • Steam generator tube break (1)

Table 1. (Continued)

Equipment Component Category	Based on ASEP Analysis[a]	Based on Precursor (LERs) Studies[b]
Test and maintenance (T&M)	• Motor operated valves T&M (10) • Pumps: vacuum and turbine T&M (6) • Electrical breakers closed T&M (1) • AFWS insufficient flow T&M (1)	• Discharge valve select switches improperly set • Cavitation emergency feedwater (EFW) suction port; wrong procedures • Transformer overload trip: other transformer out for inspection • LOFW; AFW pump out for repair, motor control center out for inspection
Human error	• Equipment alignment, position or actuation in; HPRS, valve position, low pressure recirculation system (LPRS), containment plugs, AFWS. • Improper timing (too soon, doesn't establish) on high pressure recirculation system (HPRS), feed and bleed, recirculation. • Insufficient emergency DC battery charge	• Operator opened main feed breaker erroneously • Operator opened main tie breaker instead of auxiliary • Relay opened when bumped by worker • Operator actuated breaker at wrong time • Failure to properly reset AFW pump breaker caused loss of feedwater (LOFW) • Loss of offsite power (LOOP) caused by spurious signal on maintenance switch of 115 VAC power • LOOP caused by procedural error; trip circuits not faulted • LOOP caused by premature manual trip • LOOP caused by improper switching • LOOP caused by incorrect under voltage setpoints for safety buss • Fuses not installed; cause feedwater (FW) pump to fail to start • Instrument dummy test loads cause erroneous signals and possible loss-of-coolant accident (LOCA) • HPI flow incorrectly reduced with PORV stuck open Three Mile Island Unit-2 (TMI-2) • PORV's stick open during heatup; technician removed recorder fuses • Misunderstood instructions cause erroneous containment spray actuation • Improper instrument route causes diesel generator (DG) to fail to start • Scram caused by accidental grounding of hydrogen analyzer • Steam dump valves stuck open; air dampers in wrong position.

Table 1. (Continued)

Equipment Component Category	Based on ASEP Analysis[a]	Based on Precursor (LERs) Studies[b]
Weather	(Category not included in ASEP Study)	Loss of offsite power caused by • Lightning strikes (3) • Electrical storms (2) • Hurricane or tornado (3) • External grid disturbance (1) • Ice storm (1) Loss of feedwater caused by ice (1)

a. Reference 1, the ASEP report on which this data is based considered six PWR plants (Arkansas Nuclear-1, Calvert Cliffs-2, Oconee-3, Crystal River-3, Sequoyah-1, Surry-1) and four BWR plants (Browns Ferry-1, Grand Gulf-1, Millstone Point-1, Peach Bottom-1).

b. The Precursor Studies are based on the 1969-1979 Status Report (NUREG/CR-2497)[2] and 1980-1981 Status Report (NUREG/CR-3591).[3]

c. Numbers in ( ) indicate the number of occurrences found for each item.

Table 2. Summary: predicted significant contributors to core damage and/or core melt sequences for BWR systems

Equipment Component Category	Based on ASEP (PRA Analysis)[a]	Based on Precursor Studies (LER Experience)[b]
Valves	• Safety relief valves (SRVs) (fail to reseat) • Hardwater failure outlet valves and pipes • LOOP; minimum bypass valve (fail to close) • MOV isolation condenser makeup (fail to open)	• Primary relief (5) • Main steam relief (2) • Auto depressurization system (2) • Steam supply (1) • Turbine stop (1) • Isolation (torque switch) (1) • Scram discharge vent (1) • Solenoid control valve and regulator (2)
Pumps	(No severe accident sequences involving pump failures predicted by ASEP)	• Feedwater pump failure (vibration) (1)
Electrical	• Circuit breaker fails (open or close) (7) • Circuit breaker fails (no output) (3) • Step down transformer (local fault) (2) • Undervoltage shutdown printed circuit (PC) board (3) • Failure to recover offsite power (5)	• Blown fuse; "no break" power panel (1) • Auxiliary turbine trips; improper alignment (1) • Electrical connector; reactor coolant isolation cooling (RCIC) turbine tachometer (1) • Tachometer limit switch faulty; RCIC (1) • Tachometer circuit resistor failed; RCIC (1)

Table 2. (Continued)

Equipment Component Category	Based on ASEP (PRA Analysis)[a]	Based on Precursor Studies (LER Experience)[b]
Instruments	• Reactor low pressure switch; calibration • Control circuit bypass valve; no output • Common mode logic failure; reactor protection system (RPS)	• Reactor water level signal; calibration
Systems and miscellaneous equipment	• Diesel generator fails to start (8) • Gas turbine generator fails to start (1) • Insufficient makeup to primary coolant system (PCS) (1) • Control rod failure; 3 adjacent assemblies (1)	• High pressure coolant injection (HPCI) auxiliary oil pump; broken oil line
Maintenance and test	• RCIC MOVs opened (or closed) T&M (8) • Standby service water system (SSWS) MOVs opened (or closed) T&M (5) • SSWS pump down for maintenance (1) • Failure to restore T&M fault (2)	• HPCI and RCIC improper calibration procedures (1) • Service water intake blocked; sea life (1) • HPCI and RCIC loss due to AC power loss (1)
Human error	Operator fails to; • Initiate standby liquid control system (SLCS) (2) • Initiate control rod insertion (2) • Open isolation condenser makeup valve (1) • Initiate torus cooling (1) • Initiate depressurization system (2) • Start any high pressure service water system (1)	• Leakage through control valve; N_2 pressure set too high • LOOP; worker bumped relay • LOOP; power busses not tied correctly • LOFW; candle flame leakage test; cable tray fire • Differential power relay set incorrectly
Weather	(Category not included in ASEP Study)	• LOOP; salt buildup on insulators and electrical lines.

a. Reference 1, the ASEP report on which these data are based considered six PWR plants (Arkansas Nuclear-1, Calvert Cliffs-2, Oconee-3, Crystal River-3, Sequoyah-1, Surry-1) and four BWR plants (Browns Ferry-1, Grand Gulf-1, Millstone Point-1, Peach Bottom-1).

b. The Precursor Studies are based on the 1969-1979 Status Report (NUREG/CR-2497)[2] and 1980-1981 Status Report (NUREG/CR-3591).[3]

c. Number in ( ) indicate the number of occurrences found for each item.

value based on DAS probability, component importance factor, and the number of DASs for a given plant that a particular component appeared in, was used to rank the importance of each e/c item (see Appendix A). Essentially all DASs identified by ASEP (Reference 1) were considered in this study. However, consideration was limited to only the main event items in the DASs with component importance factors greater than about 1×10^{-2} as derived from the ASEP. Table A-4 presents descriptive examples of main events and the basic events considered in each main event.

Identification of the e/c using the ASP study results has been based on the description and sequence-of-interest event tree for each precursor presented in the study reports (References 2 and 3). The precursor events included in this study were arbitrarily limited to a Significance Category of 40 or less (precursor probability of 1×10^{-4} or greater).

DISCUSSION OF RESULTS

The e/c identified by ASEP and ASP were tabulated under the equipment categories of valves, pumps, electrical, instruments and systems. Results for the non-equipment categories of human error, weather and maintenance were also included in Tables 1 and 2 to provide insight and reference interface information as discussed in the following paragraphs. Since the purpose of this study is to derive an e/c list to assist in focusing the efforts of EQRP, no attempt has been made to assign probability, unavailability, or similar rating values to the predicted significant contributors to CD/CM or SCD items listed in Tables 1 and 2.

Briefly, Tables 1 and 2 indicate that essentially the same equipment has been derived from ASEP and ASP as predicted significant contributors for both PWRs and BWRs. The exception to this is the pump category for which there are essentially no significant contributors predicted for BWRs. Although the results of this study were derived from data for specific plants, the summary presented in Tables 1 and 2 is intended to provide a generic listing of the e/c that have been identified as predicted significant contributors to CD/CM or SCD. A summary of the major e/c items identified from the ASEP and ASP studies are:

Valves

Motor or power operated [MOV or power operated valves (POVs)]

Safety or primary relief (SRV or PORVs)

Isolation (steam, sump, condenser, etc.)

Scram discharge vent valves [Identified by LERs for BWRs. Failure could lead to anticipated transient without scram (ATWS)].

Pumps (Primarily for PWRs)

Feedwater (auxiliary and emergency)

Service water (centrifugal and vacuum)

High pressure-safety injection

Reactor coolant (broken shaft and seal failure)

Electrical

Circuit breakers and relays

Power busses

Controllers and circuits

Transformers

Power supplies

Cables and connectors

Limit switches

Systems and Miscellaneous Components

Diesel generator (fail to start or run or trips due to load imbalance)

Feedwater system (clogged strainers)

Primary coolant system (steam generator tube rupture)

Boron injection heatup system

Although test and maintenance, human error, and weather may be considered to be outside the scope of EQRP, they were included in Tables 1 and 2 to provide insight and interface information between the e/c and the effects on equipment performance due to human and nature interactions. Occurrences such as (a) accidental grounding of hydrogen analyzer system causes scram or (b) instrument dummy test load causes erroneous signals and possible LOCA, are examples of situations which e/c must withstand without failure or degradation of performance. The potential occurrence of these situations should be included in the consideration for equipment qualification criteria. A summary of the major non-equipment observations identified from the ASEP and ASP studies which could influence equipment qualification research requirements is described in the remainder of this section.

Test and Maintenance (T&M): Virtually all of the equipment, components, and systems summarized in the major e/c items above appear in the T&M category. Some of the observed causes for predicted significant contributors due to test and maintenance (T&M) are:

Hypothesized failure of alternate component while primary component is out for T&M or repair.

Failure to provide function (such as LOFW) because multiple inspection and/or repair actions on required subsystems are being performed simultaneously.

Improper alignment of valves, switches, or similar equipment causes failure of given system to perform.

Improper procedures

Human Error: A large number of the predicted significant contributors to CD/CM situations can be attributed to human error. Taken from this study as an individual category of contributors, human error appears to be the largest single contributor to CD/CM. Some details for the major human errors identified by the ASEP and ASP studies are presented in Tables 1 and 2. A summary of these errors is;

Improper alignment of equipment (improperly positioned valves, incorrect generator alignment)

Improper timing and actuation (establishment of feed and bleed or recirculation too soon or late)

Improper application of procedures (battery charge low)

Failure to operate or erroneous operation of equipment by operator (failing to initiate control rod insertion)

Inadvertent equipment operation (bumping breaker, causing it to open)

Improper calibration, set points, or use of dummy loads (dummy test loads causing erroneous signals)

Incorrect wiring assembly of system or equipment (power busses tied incorrectly; LOOP)

Spurious signals (115 VAC maintenance switch causing spurious signals; potential LOOP)

Misunderstood instructions (cause undesired initiation of containment spray).

Weather: The category of weather was not specifically included in the ASEP study because ASEP was limited to internal events occurring regardless of the external initiation. For those weather related events appearing in the ASP precursor study, the major effects on the plants that were noted are:

Loss of offsite power (LOOP) caused by lightning, storms, and salt buildup on electrical insulators (caused by ocean water)

Loss of feedwater (LOFW) caused by ice formation

Steam dump valves fail to open due to cold weather; air dampers incorrectly positioned.

The information presented in this section to this point has been oriented toward the information presented in Tables 1 and 2 and derived from the ASEP (Reference 1) and ASP (References 2 and 3) studies. Appendix A provides a discussion of the selection methods used to identify the e/c based on the ASEP studies. Table A-4 presents a sample of the main events associated with the DAS as well as the probabilistic risk assessment (PRA) identification number for each e/c associated with each main event. In general, the PRA identification number, which identifies a basic event in the PRA, can be used to trace through PRA documentation and drawings to identify specific e/c details such as manufacturers, model numbers, and other pertinent information for the specific item identified. The PRA identification number for each e/c item identified by this study is given in the compilation of summary sheets (Table A-5).

Appendix B provides a discussion of the selection methods used to identify e/c based on the ASP studies. The Nuclear Safety Information Center (NSIC) accession number for each e/c is given in the compilation of summary sheets provided as Table B-2. The accession number provides traceability to specific plant, LER, and other details required for identification of specific e/c (see Exhibit B-2 for example).

Reference 4, summarizes and delineates the current major ASEP findings and insights regarding light water reactor (LWR) accident sequences and containment responses for both PWRs and BWRs. Briefly, Reference 4 addressed PRA accident sequences with revised baselines using PRA groupings by like characteristics and employing the most current accident sequence insights and progression uncertainties. It also addressed variations in containment design and the potential impact on containment response. However the accident sequences were only analyzed to the system level, and details pertaining to the identification of e/c were not provided. As a consequence Reference 4 report results were used for reference and to complement the results provided by Reference 1.

Reference 5, the Brookhaven National Laboratory (BNL) report, is a seismically oriented study and feasibility analysis that is intended to provide a list of seismically risk sensitive systems and equipment for representative plants. The plant models are intended to be for generic hybrid plants. The PWR model consisted of modified event trees from Surry-One (Wash 1400) and fragility data from Zion (SSMRP), and the BWR model consisted of event trees from Peach Bottom and fragility data for Oyster Creek. Based on the PWR and BWR hybrid plant models assembled from this information, the BNL study derived a list of seismically risk sensitive systems and equipment. A summary listing of the derived systems and equipment excerpted from Reference 5 is presented in Table 3. Almost all of the items listed in Table 3 also appear in Tables 1 and 2. A few items that are seismically sensitive exceptions are soil failure/slab uplift, storage tanks, and buried condensate pipes. Although the BNL study results do not provide a complete generic risk ordered list of equipment that can be applied directly to specific plants, a good comparison with the results derived from the ASEP and ASP studies as shown in Tables 1 and 2 is provided. In this regard the BNL results provide valuable confirmatory information and lend confidence to the items listed in Tables 1 and 2.

NUREG/CR-3428 identifies items that are important to risk: electrical equipment failure and the indirect failure of mechanical equipment (i.e., caused by building failure).[6] Based on discussions with Dr. Michael P. Bohn of the SSMRP the following items important to risk were identified:

- Pipe failures between buildings
- Electrical busses and relays in onsite emergency AC power systems
- Safety and relief valves
- Reactor protection system failures.[7]

References 8 and 9 were also reviewed as part of this study. These LER summaries for pumps and valves provided additional details to complement the information presented in the ASP reports (References 2 and 3). References 10 and 11 also provided information which essentially complemented the ASP reports. References 12 and 13 were reviews of the ASP studies and did not provide additional information to identify e/c.

RESULTS AND RECOMMENDATIONS

The results of this study are to focus the efforts of EQRP on e/c which have been identified as predicted significant contributors to SCD conditions. The desired objective, to identify significant contributors to risk, could not be fully achieved because the required risk reference studies are not completed, do not provide the required detail, or (as is the case with References 1, 2, 3 and 4) do not provide information beyond SCD. From this standpoint, future additions must be made to the e/c

Table 3. Seismically risk sensitive systems and equipment identified by Brookhaven National Laboratory feasibility study

Equipment Component Category	Based on BNL Feasibility Study
PWR Systems	• Emergency power systems; diesels, DC busses and batteries • Auxiliary feedwater systems; storage tanks • Containment heat removal systems; electric pumps • High pressure injection/recirculation; service water pumps
PWR Components and Equipment	• Service water pumps • Pipes; buried condensate • Pumps and turbines • Pipes • Tanks
PWR Events	• Loss of offsite power • Soil failure/slab uplift
BWR Systems and Associated Components	• Core spray injection; nozzles, valves and electrical relays, cables and breakers • High pressure coolant injection; valves and electrical cables, relays and pressure switches • Reactor core isolation cooling; valves and electrical cables relays and pressure switches • Reactor protection system; valves • High pressure service water; valves • Emergency service water; relays • Low pressure coolant injection; valves and relays.

listings presented in Tables 1 and 2 to include significant contributors that are identified between SCD and risk. Identification of containment-related items such as cooling fans, purge vent valves, isolation valves, and other release control e/c must be made by studies that are risk oriented.

The general categories of e/c resulting from this study, and listed in Tables 1 and 2, have been compiled from both analysis and experience studies (References 1, 2 and 3) which addressed specific plants on a plant by plant basis. Based on the compilation summary tables presented at the back of Appendixes A and B, traceability to specific e/c details such as manufacturer, model number, etc. is possible in most cases. Presentation of the study results in this fashion provides a general listing as well as a compilation of plant specific details which can be modified or made larger as more information becomes available (particularly risk oriented information). If properly updated, Tables 1 and 2 and the compilation summary tables can provide a means of focusing the EQRP efforts and provide a method to trace desired e/c specific details.

An abbreviated form of Tables 1 and 2 has been used to compile a list of e/c for which past qualification test details from commercial testing laboratory files are desired. As part of EQRP, a data search contract has been awarded to a commercial testing lab. The preliminary e/c list (Tables 1 and 2) were used in the preparation of the work scope for this contract. Similar use of the results of this study will be made for other identified EQRP research projects.

It is recommended that additional efforts be expended on this study to identify e/c that contribute to the probability of public risk due to containment release. As results of studies directed toward the public risk/containment release become available, identified e/c items should be added to or deleted from Tables 1 and 2. Currently EQRP is researching purge vent and containment isolation valves which are in this category.

It should also be noted that continuing efforts are in place for the ASEP and ASP studies. The ASEP work used for this study was based on the draft report which analyzed six PWRs and four BWRs. ASEP studies on additional plants are in process and, when they are completed, the results should be factored into Tables 1 and 2. A similar effort should be made with regard to the ASP study of LERs.

REFERENCES

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APPENDIX A
EQUIPMENT AND COMPONENT SELECTION METHODS FOR ACCIDENT SEQUENCE EVALUATION PROGRAM (ASEP) PREDICTED SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

APPENDIX A
EQUIPMENT AND COMPONENT SELECTION METHODS FOR ACCIDENT SEQUENCE EVALUATION PROGRAM (ASEP) PREDICTED SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

Presented in this appendix are details pertaining to the references and methods used for the identification of the equipment and components (e/c) determined to be predicted significant contributors to the likelihood of core damage and/or core melt (CD/DM) situations based on the Accident Sequence Evaluation Program (ASEP) results.[A-1] A summary of the identified e/c has been presented in Tables 1 and 2 of the main body of this report.

Figure A-1 presents a graphic representation of portions of the ASEP report results that were used and the summary listings developed to derive the e/c listings presented in Tables 1 and 2. Because of the large volume of records involved, only samples of some of the (ASEP) documentation are presented.

Table A-1 presents a listing of the dominant accident sequences (DAS) identified for the Arkansas Nuclear One plant (ANO-1) picked at random for this example. The probability to CD/CM values presented are based on the ASEP reports (References A-1 and A-2) and the probabilistic risk assessments as described in Reference A-1. Some differences in the probabilities predicted by these efforts is evident. ASEP essentially used the PRA information with some changes to discount recovery actions and other similar occurrences. ASEP used the same PRA information but included different baseline information as described in Reference A-2.

Because ASEP (Reference A-1) was the only study which provided sufficient information to permit the tracing and identification of individual specific e/c, the probability values presented for it were used in subsequent analyses as a means of selecting important events and equipment.

Table A-2 presents the ASEP events and associated event importance for the DAS example (1.2)D1 for the ANO-1 plant. A similar table is presented in Reference 1 for each of the 14 DASs identified by ASEP for ANO-1. For the purposes of this EQRP study, only the ASEP events with an importance of approximately 5×10^{-2} to 1×10^{-1} and greater were used for each DAS.

Table A-3 presents a composite tabulation of the DASs and ASEP main event importances as taken from Tables A-1 and A-2. The purpose of the Table A-3 tabulation is to provide a means of gathering the ASEP events associated with each DAS for ANO-1, and ranking the events based on the DAS sequence probability, the importance factor, and the number of DASs containing each ASEP event. The numerical value derived for the ranking has no significance other than the ranking of ASEP events for a given plant relative to one another. The ranking value shown at the right-hand side of Table A-3 is simply the sum of all the products of the DAS probability and importance factor for all the DASs contained in the ASEP event.

Table A-4 presents a sample of the ASEP events and associated e/c. The total unavailability and the unavailability for each e/c is also given. Using the ASEP event ranking as shown in Table A-3, the e/c associated with each ranked main event was identified and an equipment category assigned (see notes on Table A-4). The e/c categories assigned are; valves, pumps, electrical, instruments, systems, maintenance and test, and human error. Weather was not included in the ASEP analysis.

Table A-5 presents the summary listings for the e/c categories that were identified by the ASEP study for the six PWR and four BWR plants. The ASEP event identification and a general description of the e/c is presented. The event ID provides traceability to the ASEP study and, if required, the PRA identification number provides traceability to reference PRA documentation so that the specific component involved can be identified. The amount of information available for the specific component varies from PRA to PRA. However, for most PRAs the component manufacturer, model number and other specific information can be determined. The summary tabulations presented in Table A-5 were used to derive the e/c items presented in Tables 1 and 2 of the main body of this report.

References

A-1. W. H. Sullivan et al., *Accident Sequence Evaluation Program Catalog of PRA Dominant Accident Information*, NUREG/CR-3301, EGG-2259 (to be published).

A-2. A. M. Kolaczkowski et al., *Interim Report on Accident Sequence Likelihood Reassessment (Accident Sequence Evaluation Program)*, Draft Report Sandia National Laboratories, February 1983 (to be published as NUREG/CR-3801).

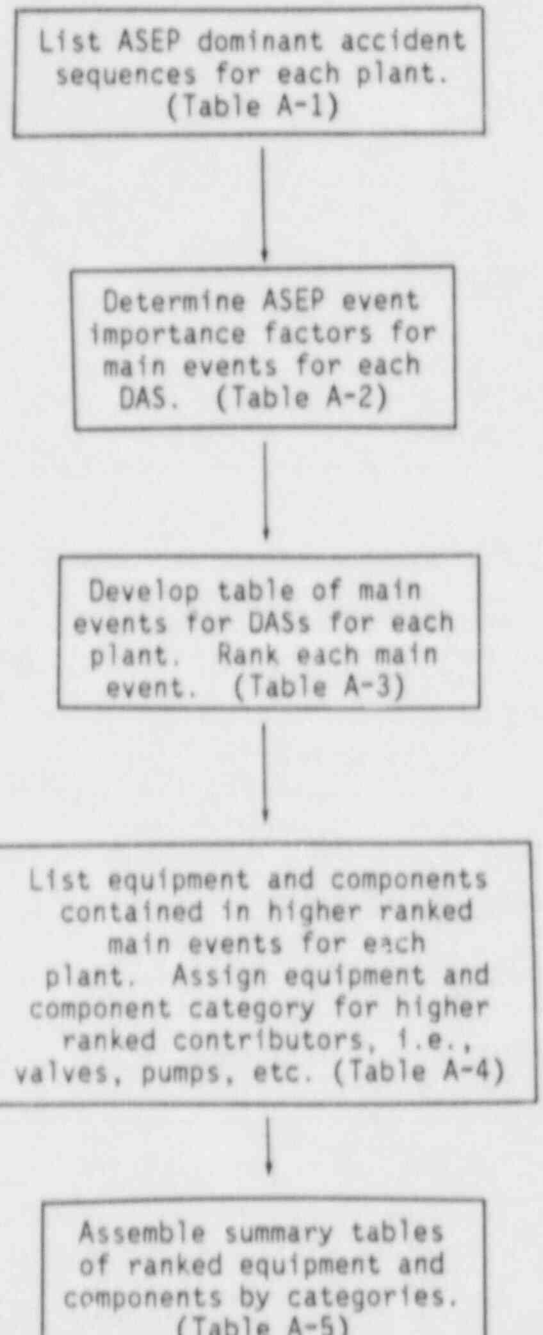


Figure A-1. Graphic representation—ASEP equipment and components selection method.

Table A-1. ASEP dominant accident sequences

Plant: ANO Type: PWR PRA Method: Interim Reliability Evaluation Program (IREP)

Dominant Accident Sequence Acronym	Brief Description	Probability[a] ASEP/EGG[b]	ASEP/SNL[c]	PRA	Comment
B(1.2)D_1	Seal or pipe rupture 0.38 to 1.2 in. and HPIS failure	1.7×10^{-5} (5)	3×10^{-6} (7)	2.8×10^{-6} (7)	Small LOCA
B(1.2)D_1C	Seal or pipe rupture 0.38 to 1.2 in. HPIS and reactor building spray injection system (RBSIS) failure	1.6×10^{-5} (6)	4.5×10^{-6} (2)	4.4×10^{-6} (2)	Small LOCA
T(LOP)LD_1Y	Transient; LOOP; failure power conversion, emergency feedwater system (EFS) HPIS, reactor building cooling system (RBCS) and RBSIS systems	1.0×10^{-5} (8)	4.0×10^{-6} (4)	9.9×10^{-6} (1)	
B(4)H_1	Piping rupture 1.7 to 4 in. and high pressure recirculation system (HPRS) failure	3.0×10^{-5} (1)	1.5×10^{-5} (1)	1.4×10^{-6} (12)	Small LOCA
T(D01)LD_1YC	Transient; 125 VDC power bus, EFS, HPIS, RBCS, and RBSIS failures	2.3×10^{-5} (3)	3.0×10^{-6} (8)	3.1×10^{-6} (5)	
T(D02)LD_1YC	Transient; 125 VDC power bus, EFS, HPIS, RBCS, and RBSIS failures	2.0×10^{-6} (14)	2.5×10^{-6} (9)	2.5×10^{-6} (8)	
B(1.66)H_1	Rupture 1.2 to 1.66 in. plus HPRS failure	2.5×10^{-5} (2)	3.0×10^{-8} (14)	1.2×10^{-6} (13)	Small LOCA
T(D01)LQ-D_3	Transient; 125 VDC power bus, EFS, 2 of 3 HPIS failures, one SRV fails to close	2.0×10^{-5} (4)	4.0×10^{-6} (3)	4.0×10^{-6} (3)	
T(A3)LQ-D_3	Same as above only 4160 VAC engineered safety feature (ESF) power bus fails	5.6×10^{-6} (10)	3.5×10^{-6} (5)	3.3×10^{-6}(4)	
T(FIA)KD_1	Transient; anticipated transient without scram (ATWS), failure to initiate HPIS	2.8×10^{-6} (12)	3.0×10^{-6} (6)	2.8×10^{-6} (6)	ATWS
T(D01)LD_1	Transient 125 VDC, ESF power bus, EFS, HPIS failures	1.4×10^{-5} (7)	2.0×10^{-6} (10)	2.2×10^{-6} (9)	
T(A3)LD_1	Transient 4160 VAC, ESF power bus, EFS, HPIS failures	4.2×10^{-6} (11)	1.0×10^{-6} (13)	1.0×10^{-6} (14)	
T(D01)LD_1C	Transient 125 VDC, ESF power bus, EFS, RBSIS failures	8.5×10^{-6} (9)	2.0×10^{-6} (11)	1.8×10^{-6}(10)	
T(A3)LD_1C	Transient 4160 VAC, ESF power bus, EFS, HPIS, RBSIS failures	2.4×10^{-6} (13)	1.5×10^{-6} (12)	1.4×10^{-6} (11)	

a. Numbers in () are sequence ranking order numbers for each study based on highest probability.

b. EG&G Idaho, ASEP.

c. Sandia National Laboratory ASEP.

Table A-2. ASEP events and associated event importance

	ASEP Event	Importance
1.	PSSWHP08	3.3335E-01
2.	HSECSX42	2.4857E-01
3.	ACHPIC08	1.9498E-01
4.	ACSWHP08	1.8869E-01
5.	HSHPIC08	1.5095E-01
6.	HSSWXH38	1.2097E-01
7.	HSSWXA37	1.2097E-01
8.	FSHPCM07	1.1820E-01
9.	HSHPLP09	8.6170E-02
10.	ACHPLP09	5.302BF-02
11.	ACSWXH38	5.1371E-02
12.	ACECSX42	4.9713E-02
13.	ACSWXA37	4.9713E-02
14.	ACSWXA36	4.9713E-02
15.	HSSWXA36	3.1485E-02
16.	TMHPIC08	1.3837E-02
17.	TMECSX42	7.2913E-03
18.	PSECSW42	6.6285E-03
19.	TMSWXA36	3.1485E-03
20.	ENECSX42	2.9828E-03
21.	TMHPLP09	2.6514E-05
22.	TMSWXA37	2.6514E-05

Table A-3. Tabulation of ASEP DAS ASEP events of importance

Plant: Arkansas Nuclear

ASEP Events	B(1.2)D1	B(1.2)D.C	T(Lop)LD.VC	B(4)H1	T(DO1)LD1YC	T(DO2)LD1YC	B(1.65)H1	T(DO1)LO-03	T(A3)LD03	T(F3A)KD1	T(DO1)LD1	T(A3)LD1	T(DO1)LD1C	T(A3)LD1C	
Probability	1.7E-05	1.6E-05	1.0E-05	3.0E-05	2.3E-05	2.0E-06	2.5E)5	2.0E-05	5.6E-06	2.8E-06	1.4E-05	4.2E-06	3.5E-06	2.4E-06	Ranking Values[a]
HSSWHF08	0.33										0.38	0.37			(5) 1.24E-05
HSECSX42	0.25														4.25E-06
ACHPIC06	0.19										0.22	0.22			(11) 7.23E-06
ACSWHF08	0.19										0.21	0.21			(12) 7.05E-06
HSHP2C06	0.15										0.17	0.17			(15) 5.64E-06
HSSWXH38	0.12			0.09			0.09								(14) 6.99E-06
HSSWXA37	0.12			0.09			0.09								(14) 6.99E-06
ESHPSM07	0.12														2.04E-06
HSHPLP09		0.29													4.64E-06
HSHPL010		0.29											0.62	0.62	(6) 1.14E-05
HSECSX41		0.22		0.18	0.57		0.18								(1) 2.65E-05
HSECSX42		0.22		0.18			0.18								(3) 1.34E-05
ACHPLD10		0.18											0.38	0.38	(13) 7.02E-06
ACHP1P09		0.18													2.88E-06
HSHPCM13		0.17													2.72E-06
CMEDCX26			0.73												(10) 7.30E-06
HSDGX234			0.20												2.0E-06
HSDGX133			0.20												2.0E-06
ACEFSS24			0.13									0.54			3.57E-06
ACSWXH38				0.04			0.04								2.20E-06
ACECSX41				0.04	0.11		0.04								4.73E-06
ACEFSS2T					0.21								0.20		(4) 1.30E-05
ACEFSS25					0.18			0.19			0.19		0.17		(7) 1.10E-05
ACEFSS22					0.12			0.16			0.16		0.11		(9) 7.44E-06
ACSWX235					0.11			0.11			0.11				2.53E-06
HSEESS15					0.10										2.30E-06
HSDCX127						0.65									1.30E-06
TMDCX127						0.27									5.40E-07
HSRVXX46								1.00							(2) 2.50E-05
HSEFSS15								0.10			0.10		0.10		4.25E-06
HSEFSS20								0.09			0.09		0.09		3.83E-06
ACEFSS23								0.09			0.09		0.09		3.83E-06
ACEFSS24								0.08	0.55		0.08		0.08	0.55	(8) 7.80E-06
HSRVXX46									1.00						(15) 5.60E-06
HSEFSS17									0.38		0.06	0.39		0.38	(16) 5.52E-06
ESHPCM48										1.00					2.80E-06
ACRPSX01										0.50					1.40E-06
ACRPSX02										0.50					1.40E-06
ACRPSX03										0.25					7.00E-07
ACRPSX04										0.25					7.00E-07
ACRPSX05										0.25					7.00E-07
ACRPSX06										0.25					7.00E-07
HSEFCM18												0.04		0.04	2.64E-07

a. Number in ( ) ranking for main event. Main events without ( ) were not used in this study.

Table A-4. ASEP listing of equipment and components and unavailability for ASEP events

Main Event	Subordinate Event	Component	Unavailability
HSSWHP08	--	--	Σ = 5.3E-03
	*SWS0318CX-XOC-LF--manual valve-OC, plug	--	1.0E-04
	*SWS3180B-VCC-LF--motor operated valve-CC, failure to operate	Valves	4.0E-03
	SWS3180B-VCC-LF--motor operated valve,plug	--	1.0E-04
	*6214B-CBL-LF--cable, open circuit	Cables	1.1E-03
HSRVXX46	--	--	Σ = 1.0E-02
	*Primary relief valve failure to reseat	Valves	
HSECSX42	--	--	Σ = 1.5E-02
	*ECSCH4BA-CWU-LF--chilled water unit, failure to start	Systems	2.3E-03
	*ECSCH4BA-CWU-LF--chilled water unit, failure to run	Systems	1.4E-03
	*5254A-CBL-LF--cable, open circuit	Cables	1.1E-03
	*ECS5254A-B-AASF--thermostat failure to close	Instrument	5.4E-03
	EC5602BX-XOC-LF--manual valve-OC, failure to remain open (plug)	--	1.0E-04
	ESC604BX-CCC-LF--check valve-oc, failure to remain open (plug)	--	1.0E-04
	ECS601BX-XOC-LF--manual valve-oc, failure to remain open (plug)	--	1.0E-04

* Asterisk indicates events used in the study.

Table A-5. Summary: equipment and components identified as significant contributors based on ASEP report

Plant Type: PWR

Category: Valves

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System: General Description	PRA Identification Number	Portion of Event Unavailable	Notes
ANO-1	HSSWHP08	1	5.3E-3	Motor operated valve failure to operate	SWS 3180B-VCC-LF	4.0E-3	
ANO-1	HSRVXX46	2	2.0E-2	Primary relief valve failure to reseat	(none)	2.0E-2	
ANO-1	HSECSX42	3	1.5E-2	3 way valve failure to operate	ECS 6036A-DPC-LF	4.0E-3	
ANO-1	HSHPLP10	6	5.2E-3	Motor operated valve failure to operate	LPI 1408B-VCC-LF	4.0E-3	
CC-2	HSPORV20	6	8.0E-2	Power operated relief valve failure to reseat	(none)	8.0E-2	
CC-2	EOPORV24	7	1.0E-1	Block valve failure to close given PORV stuck open	(none)	1.0E-1	
OC-3	HSSRVX08	1	5.0E-2	Failure and pressurizer safety relief valve to re-seat	(none)	5.0E-2	
OC-3	HSHPIC22	6	1.4E-2	Motor operated valve failure	HP-27 HP-25	1.0E-3	
SE-1	HSAFWX05	1	3.1E-3	Failure of any single isolation valve between containment system and Aux feedwater pumps	(none)	3.1E-3	
SE-1	HMHPIX04	5	1.7E-3	Multiple control valve failures for boron inject and refuel water storage supplies	(none)	1.7E-3	
ANO-1	HSSWHP08	1	5.3E-3	Motor operated valve failure to operate	SWS 3180B-VCC-LF	4.0E-3	
ANO-1	HSRVXX46	2	2.0E-2	Primary relief valve failure to reseat	(none)	2.0E-2	
ANO-1	HSECSX42	3	1.5E-2	3 way valve failure to operate	ECS 6036A-DPC-LF	4.0E-3	
ANO-1	HSSWHP08	5	5.?E-3	Motor operated valve failure to operate	SWS 3180B-VCC-LF	4.0E-3	
ANO-1	HSHPLP10	6	5.2E-3	Motor operated valve failure to operate	LPI 1408B-VCC-LF	4.0E-3	
CC-2	HSPORV20	6	8.7E-2	Power operated relief valve failure to reseat	(none)	8.0E-2	
CC-2	EOPORV24	7	1.?E-1	Block valve failure to close given PORV stuck open	(none)	1.0E-1	
OC-3	HSSRVX08	1	5.0E-2	Failure of pressurizer safety relief valve to re-seat	(none)	5.0E-2	
OC-3	HSHPIC22	6	1.4E-2	Motor operated valve failure	HP-27 HP-25	1.0E-3	
SE-1	HSAFWX65	1	3.1E-3	Failure of any single isolation valve between containment system and Aux feedwater pumps	(none)	3.1E-3	
SE-1	HMHPIX04	5	1.7E-3	Multiple control valve failures for boron inject and refuel water storage supplies	(none)	1.7E-3	

Table A-5. (Continued)

Plant Type: PWR

Category: Pumps

				Equipment/Component/System			
Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	General Description	PRA Identification Number	Portion of Event Unavailable	Notes
CC-2	HSAFW203 (AFW)	2	2.1E-2	Turbine pump failure (fails to operate after 24 hours)	TP-22	2.0E-2	
CR-3	HSEFSA08 (emergency feed)	6	2.2E-2	Turbine pump fails to start	EFP-2	2.0E-2	
OC-3	HSLSWB01 (low pressure service water)	5	1.4E-3	Centrifugal pump fails to run for 24 hours, vacuum pump fails to run for 24 hours	LPSW-P3B VP1	7.2E-4 7.2E-4	
OC-3	HSHPIC22 (high pressure injection)	6	1.4E-2	Pump lube oil too viscous, pump hardware	HP-PIC HP-PIC	1.0E-2 1.0E-3	
OC-3	HSLSWA02 (low pressure service water)	17	2.0E-3	Centrifugal pump hardware, vacuum pump hardware	LPSW-P3A VP2	1.0E-3 1.0E-3	
SU-1	HSTPUM02 (AFW)	4	1.2E-3	No flow from turbine pump signals No flow from turbine signals No flow from turbine pump signals	PCV0142C PPMTURBF PXV0153C	1.0E-4 1.0E-3 1.0E-4	

Table A-5. (Continued

Plant Type: PWR

Category: Electrical: Cables, Circuit Breakers, Relays, Power Busses

				Equipment/Component/System			
Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	General Description	PRA Identification Number	Portion of Event Unavailable	Notes
ANO-1	HSSWHP08	1	5.3E-3	Cable: open circuit	6214 B-CBL-LF	1.1E-3	
ANO-1	HSECSY42	3	1.5E-2	Cable: open circuit	5254 A-CBL-LF	1.1E-3	
ANO-1	ACEFSS21	4	1.1E-2	Cables: (8 ea) open circuit	EFCAC04 X-CBL-LF (typical)	1.1E-3 (ea)	
ANO-1	HSSWHP08	5	5.3E-3	Cable: open circuit	6214 B-CBL-LF	1.1E-3	
ANO-1	ACEFSS25	7	9.3E-3	Cables: (7 ea) open circuit	EFVVD41 X-CBL-LF (typical)	1.1E-3 (ea)	
ANO-1	ACEFSS24	8	4.5E-3	Cables: (4 ea) open circuit	EFVVD25 X-CBL-LF (typical)	1.1E-3 (ea)	
ANO-1	ACEFSS22	9	6.4E-3	Cables: (5 ea) open circuit	EFCRS22 B-CBL-LF	1.1E-3 (ea)	
ANO-1	ACHPIC08	11	3.1E-3	Control breakers: local fault and failure to transfer	HPIA406 B-BOO-CC HPIA406 B-BOO-LF	2.0E-3 1.0E-3	
ANO-1	ACSWHP08	12	3.0E-3	Control breakers: local fault and failure to transfer	SWS6214 B-BOO-CC SWS6214 B-BOO-LF	2.0E-3 1.0E-3	
ANO-1	ACHPLP10	13	3.2E-3	Control breakers: local fault and failure to transfer	LPI6164 B-BOO-CC LPI6164 B-BOO-LF	2.0E-3 1.0E-3	
CC-2	ACCC2234	15	1.3E-2	Control circuit: control valve circuit failure	CV-5162 CV-5208	6.4E-3 6.4E-3	
OC-3	ACHPIC22	6	1.4E-2	Control circuit: motor operated valve control circuit	HP-25 HP-27	6.4E-3 6.4E-3	
OC-3	ACHPAB21	9	6.4E-3	Control circuit: motor operated valve control circuit	HP-24	6.4E-3	
OC-3	ACHPIA23	10	6.4E-3	Control circuit: motor operated valve control circuit	HP-26	6.4E-3	
OC-3	ACLSWA02	12	3.6E-3	Control circuit: low pressure centrifugal pump and vacuum pump	LPSW-P3A VPA	1.8E-3 1.8E-3	
OC-3	ACLPIB14	15	6.4E-3	Control circuit: motor operated valve	LP-20	6.4E-3	
OC-3	ACLPIA13	15	6.4E-3	Control circuit: motor operated valve	LP-19	6.4E-3	
SU-1	HSRPSA06	2	9.7E-4	Cables: (9 ea) wire faults	ITM0009Q (typical)	1.08E-4 (ea)	
SU-1	HSRPSB06	2	9.7E-4	Cables: (9 ea) wire faults	IWR0008Q (typical)	1.08E-4 (ea)	
SU-1	CMBUAB07	6	1.0E-4	Power busses: common mode failure	JA00-JB00	1.0E-2	
CR-3	HSDCXB06	5	3.0E-6	Batteries: insufficent power	(none)	3.0E-6	Could be considered human error: charging
CR-3	UNEACX01	3	3.6E-1	Backup AC power	(none)	3.6E-1	Need to define better
CR-3	HSDG3B03	8	6.2E-2	Circuit breaker: fails to open fails to close	 3210 3212	 1.0E-3 1.0E-3	

Table A-5. (Continued)

Plant Type: PWR

Category: Instruments

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System General Description	PRA Identification Number	Portion of Event Unavailable	Notes
ANO-1	HSECSX42	3	1.5E-2	Thermostat failure to close	ECS 5254 A-B-AASF	5.4E-3	
SU-1	HSHT3414	11	2.2E-2	Gas thermocouple fails	TIC 1934 B	1.1E-2	

Table A-5. (Continued)

Plant Type: PWR

Category: Systems

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System: General Description	Equipment/Component/System: PRA Identification Number	Equipment/Component/System: Portion of Event Unavailable	Notes
ANO-1	HSECSX42	3	1.5E-2	Chill water system: fail to start	ECSCH4 BA-CWU-LF	2.3E-3	
ANO-1	HSECSX42	3	1.[illegible]E-2	Chill water system: fail to run	ECSCH4 BA-CWU-LF	1.4E-3	
CC-2	RCPCSR09	3	1.0E-1	Primary coolant system: failure to recover PCS within 30 min. after trip caused by PCS interruption	(none)	1.0E-1	
CC-2	UNPCSX14	8	1.0E-2	Primary coolant system: failure to continue operation following trip	(none)	1.0E-2	
CC-2	HMRPSX19	11	4.0E-6	Reactor protection system: failure to terminate fission process multiple: hardware fail	(none)	4.0E-6	
CR-3	HSDG3A02	4	6.1E-2	Diesel generator system: fails to run, fails to start	DG-3A	3.0E-2 3.0E-2	
SE-1	UNERCW05	1	1.0E-2	Emergency recirculation cooling water system: failure	(none)	1.0E-2	
SE-1	EBIX04	7	7.6E-4	Boron injection tank heating system: undetected failure	(none)	7.6E-4	

Table A-5. (Continued)

Plant Type: PWR

Category: Maintenance and Test

				Equipment/Component/System			
Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	General Description	PRA Identification Number	Portion of Event Unavailaole	Notes
CR-3	TMEFS810	12	5.5E-3	EFP and MOVs out for maintenance	MOV EFV-14 MOV EFV-33 MOV EFV-7 EFP-1	9.7E-4 9.7E-4 9.7E-4 2.6E-3	
OC-3	TMLSWA02	7	8.0E-3	Low pressure pump maintenance vacuum pump maintenance	LPSW-P3A VP2	4.0E-3 4.0E-3 2.6E-3	
OC-3	TMLPIA11	8	8.2E-3	Motor uperateu valve maintenance pump test	LP-5 LP-12 LP-17 LP-PIA	2.1E-3 2.1E-3 2.1E-3 1.9E-3	
OC-3	TMHPIC22	11	8.2E-3	Motor operated valve maintenance pump test and maintenance	HP-25 HP-27 HP-PIC	2.1E-3 2.1E-3 4.0E-3	
OC-3	TMHPAB21	16	2.1E-3	Motor operated valve maintenance	HP-24	2.1E-3	
OC-3	TMHPAB23	16	2.1E-3	Motor operated valve maintenance	HP-26	2.1E-3	
OC-3	TMESA124	16	2.1E-3	Actuation train test and maintenance failure	ESPS	2.1E-3	
OC-3	TMEPAC03	17	2.1E-3	Actuation train test and maintenance failure	ESPS	2.1E-3	
SF-1	TMAFWX05	4	8.4E-7	Insufficient AFWS flow test and maintenance	(none)	8.4E-7	
SU-1	TMBKRB05	3	6.1E-3	Breaker closed due to maintenance	1CB0002X	6.1E-3	
SU-1	ENTPUM02	5	6.0E-3	No flow from turbine signals maintenance fault	PXV4041Y PXV0153Y	3.0E-3 3.0E-3	
CC-2	TMRPSX19	13	1.6E-5	T&M failures cause reactor protection system failures	(none)	1.6E-5	
CC-2	TMCC2234	14	1.2E-2	Maintenance on control valves	CV-5162 CV-5208	5.8E-3	

Table A-5. (Continued)

Plant Type: PWR

Category: Human Error

				Equipment/Component/System			
Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	General Description	PRA Identification Number	Portion of Event Unavailable	Notes
ANO-[illegible]	CMF[illegible]X26	10	2.6E-5	Combined batteries: improper charging	BATCM	2.6E-5	
CC-2	ESAFWS10	4	1.0E-1	Operator error to restore AFWS after total loss of AC power	(none)	1.0E-1	
CC-2	ESAFWS0[illegible]	5	1.0E-3	Operator fails to manually initiate AFWS	(none)	[illegible]0E-3	
CC-2	[illegible]NBTCM15	9	4.0E-4	Common mode battery failure, improper charging	(none)	4.0E-4	
CC-2	ESCC2232	6	1.0E-1	Operator fails to open control valve (CV)	CV-3824	1.0E-1	
CC-2	ENAFWS04	12	1.0E-4	Operator error; manual valve	[illegible]3 & C4	1.0E-4	
CR-3	ESHPCM13	1	8.0E-2	Operator reconfigures for HPR incorrectly	(none)	8.0E-2	
CR-3	ESLPCM17	2	[illegible]E-2	Operator switchs to recirculation too soon	(none)	5.0E-2	
CR-3	ES[illegible]CM14	7	1.4E-2	Operator fails to establish feed and bleed	(none)	1.4E-2	
CR-3	E[illegible]HPCM13	9	8.0E-2	Operator reconfigures for HPR incorrectly	(none)	8.0E-2	
CR-3	ESLPCM[illegible]	10	5.0E-2	Operator switchs to recirculation too soon	(none)	[illegible]E-2	
CR-3	ENHPIB12	11	[illegible]E-2	Manual and stop valves in wrong position by operator	(none)	1.0E-2	
OC-3	ESLPCM18	2	3.0E-3	Failure of operator to open sump valves at start of recirculation common to spray and core cooling	(none)	3.0E-3	
OC-3	ENLPSX10	3	3.0E-3	LPRS failure: test valves in wrong position	(none)	3.0E-3	
OC-3	E[illegible]SCM09	4	3.0E-3	HPRS suction alignment improper: operator failure	(none)	3.0E-3	
SE-1	ESCMHP03	2	6.0E-3	Common mode HPRS failures: operator align HPRS	(none)	6.0E-3	
SE-1	ENPLUG01	3	3.0E-3	Operator failure to remove plugs between containment chambers	(none)	3.0E-3	
SE-1	ESCMLP03	6	6.0E-3	Common mode failures of HPRS: operator failure to realign	(none)	6.0E-3	
SS-1	ENM[illegible]01	7	3.0E-6	Normally open manual valves left open inadvertently	PXVTESTY	3.0E-6	
SU-	ESHPCM50	8	3.0E-3	Operator failure to align HPR suction to LPRS	(none)	3.0E-3	
SU-1	ESLPCM01	9	3.0E-3	Operator failure to align LPRS suction to pump	(none)	3.0E-3	
SU-1	ESRCCM55	12	3.0E-3	Operator failure to align recirculation to hotleg after 24 hours	(none)	3.0E-3	
SU-1	ESLPCM[illegible]	12	3.0E-3	Operator failure to align LPRS suction to sump	(none)	3.0E-3	

Table A-5. (Continued)

Plant Type: BWR

Category: Valves

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System General Description	PRA Identification Number	Portion of Event Unavailable	Notes
BF-1	HSRBA007	12	1.0E-3	Loop 1 min. bypass valve does not close	RVM0071N	1.0E-3	
BF-1	HSRBA302	12	1.0E-3	Loop 2 min. bypass valve does not close	RVM0302N	1.0E-3	
GG-1	HSSRVR16	6	1.0E-1	Failure of safety/relief to reseat	(none)	1.0E-1	
MP-1	HSSRVR08	2	1.8E-2	Failure of safety/relief to reseat	(none)	1.8E-2	
MP-1	HSICMU20	13	1.7E-2	Isolation condenser make-up MOV fails to open	(none)	1.7E-2	
PB-1	HSESW506	4	1.0E-4	Hardware failure in outlet piping and valves	(none)	1.0E-4	

Table A-5. (Continued)

Plant Type: BWR

Category: Pumps

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System: General Description	Equipment/Component/System: PRA Identification Number	Equipment/Component/System: Portion of Event Unavailable	Notes

(No pumps were identified by ASEP/EG&G for BWRs)

Table A-5. (Continued)

Plant Type: BWR

Category: Electrical

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System General Description	PRA Identification Number	Portion of Event Unavailable	Notes
BF-1	ACEPD01D	9	5.8E-3	Circuit breaker continuous circuit: no output	ACK8160G	2.9E-3	
BF-1	ACEPD01D	9	5.8E-3	Shutdown board D under-voltage circuit: no output	ACK100DG	2.9E-3	
BF-1	ACEPC01C	11	5.8E-3	Circuit breaker continuous circuit: no output	ACK812CG	2.9E-3	
BF-1	ACEPC01C	11	5.8E-3	Shutdown board C under-voltage circuit: no output	ACK100CG	2.9E-3	
BF-1	ACEP3B30	11	5.8E-3	Circuit breaker continuous circuit: no output	ACK842BG	2.9E-3	
BF-1	ACEP3B30	11	5.8E-3	Shutdown board 3EB under voltage circuit: no output	ACK300BG	2.9E-3	
GG-1	RCEOSP14	7	2.0E-1	Failure to recover offsite power within 1 hour	(none)	2.0E-1	
GG-1	RCLOSP15	9	1.0E-1	Failure to recover offsite power within 30 hour given	(none)	1.0E-1	
MP-1	RCLOSP10	1	4.3E-1	Failure to recover offsite power within 1/2 hour	(none)	4.3E-1	
MP-1	HSEAC113	15	5.1E-2	Breaker fails to close	AC-14CT-16-FTC	1.7E-2	
MP-1	HSEAC113	15	5.1E-2	Breaker fails to open	AC-1514A-2-FTO	1.7E-2	
MP-1	HSEAC113	15	5.1E-2	Breaker fails to open	AC-1514C-1-FTO	1.7E-2	
MP-1	RLCOSP27	7	5.0E-2	Failure to recover offsite power within 20 hours	(none)	5.0E-2	
MP-1	RCLOSP19	8	2.4E-1	Failure to recover offsite power within 2 hours	(none)	2.4E-1	
MP-1	CMHIAC29	12	4.0E-1	Local fault step-down transformer	AC-IV-XFR-LOF	2.0E-1	
MP-1	CMHIAC29	12	4.0E-1	Local AC power breaker fails to stay open	AC-IV-3-FRC	1.0E-1	
MP-1	CMHIAC29	12	4.0E-1	Local AC power breaker fails to stay open	AC-IV-IF-FRC	1.0E-1	
MP-1	HSIACX24	12	4.0E-1	Local AC power breaker fails to stay open	AC-IV-IF-FRC	1.0E-1	
BF-1	TMRAIS0B	6	2.1E-3	Reactor heat removal (RHR) reactor low pressure switch out of calibration	RPS128BJ	2.1E-3	
BF-1	TMRAIS0A	6	2.1E-3	RHR reactor low pressure switch out of calibration	RPS128AJ	2.1E-3	
BF-1	ACRBA007	8	3.3E-3	Flow by pass valve control circuit no output	RCK0071G	3.3E-3	
BF-1	ACRBA302	8	3.3E-3	Flow by pass valve control circuit no output	RCK0302G	3.3E-3	
PB-1	ENRPSLCM	5	1.9E-6	Common mode logic failure RPS due to incorrect calibration	(none)	1.9E-6	

Table A-5. (Continued)

Plant Type: BWR

Category: Systems

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System General Description	PRA Identification Number	Portion of Event Unavailable	Notes
BF-1	HSEPD01D	1	3.0E-2	Diesel generator fails to start	ADL001DR	3.0E-2	
BF-1	HSEP3A3A	2	3.0E-2	Diesel generator fails to start	ADL003AR	3.0E-2	
BF-1	HSEP3B3B	2	3.0E-2	Diesel generator fails to start	ADL003BR	3.0E-2	
BF-1	HSEPC01C	3	3.0E-2	Diesel generator fails to start	ADL001CR	3.0E-2	
BF-1	HSEPB01B	4	3.0E-2	Diesel generator fails to start	ADL001BR	3.0E-2	
BF-1	HSEPA01A	7	3.0E-2	Diesel generator fails to start	ADL001AR	3.0E-2	
GG-1	UNPCS119	13	7.0E-3	PCS fails to remove heat within 28 hours	(none)	7.0E-3	
GG-1	HMCRSX22	12	5.8E-4	Failure of any 3 adjacent control rods to insert	(none)	5.8E-6	
GG-1	HSEAC202	14	3.0E-2	Diesel 2 fails to start	(none)	3.0E-2	
GG-1	IEPPCS21	16	1.0	Failure of PCS to provide makeup water within 1/2 hour given TPQE sequence	(none)	1.0	
MP-1	HSEAC103	4	6.0E-2	Gas turbine generator fails to start	(none)	6.0E-2	
MP-1	HSEAC214	11	6.0E-2	Diesel generator fails to start	(none)	6.0E-2	
PB-2	PCSNOREC	2	--	No recovery of primary coolant system	(none)	--	
PB-2	HMCRDMAB	3	5.8E-6	Failure 3 adjacent control rod drives	(none)	5.8E-6	

Table A-5. (Continued)

Plant Type: BWR

Category: Maintenance and Test

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System: General Description	Equipment/Component/System: PRA Identification Number	Equipment/Component/System: Portion of Event Unavailable	Notes
GG-1	RC2ANY18	2	2.1E-1	Failure to restore maintenance of test fault within 28 hours	(none)	2.1E-1	
GG-1	TMRCIC12	3	4.0E-2	RCIC normally open motor operated valve closed for maintenance	F064-A	5.8E-3	
GG-1	TMRCIC12	3	4.0E-2	RCIC normally open motor operated valve closed for maintenance	F063-A	5.8E-3	
GG-1	TMRCIC12	3	4.0E-2	RCIC normally open motor operated valve closed for maintenance	F045-A	5.8E-3	
GG-1	TMRCIC12	3	4.0E-2	RCIC normally open motor operated valve closed for maintenance	F068-A	5.8E-3	
GG-1	TMSWVA08	4	1.7E-2	SSWS motor operated valve normally open is closed for maintenance	F005A-A	5.8E-3	
GG-1	TMSWUB09	5	1.7E-2	SSWS pump down maintenance	C001B-B	5.8E-3	
GG-1	TMSWUB09	5	1.7E-2	SSWS motor operated valve normally open: closed for maintenance	F001B-B	5.8E-3	
GG-1	TMSWUB09	5	1.7E-2	SSWS motor operated valve normally open: closed for maintenance	F005B-B	5.8E-3	
GG-1	RC1ANY17	8	2.3E-1	Failure to restore maintenance or test fault within 28 hours	(none)	2.3E-1	
GG-1	TMSWXB07	10	1.2E-2	SSWS motor operated valve (normally closed, must open) is closed for maintenance	F014B-B	5.8E-3	
GG-1	TMSWXB07	10	1.2E-2	SSWS motor operated valve (normally closed, must open) is closed for maintenance	F068B-B	5.8E-3	
GG-1	TMRHRB05	13	1.74E-2	RHR motor operated valve closed for maintenance	F024B-B	5.8E-3	
GG-1	TMRHRB05	13	1.74E-2	RHR motor operated valve closed for maintenance	F003B-B	5.8E-3	
MP-1	TMICMV15	6	7.71E-3	MOVs closed for test and main	IC-1-MOV-TMC IC-2-MOV-TMC	2.3E-3 2.3E-3	
BF-1	ENRBA01D	12	1.0E-3	Operator error: manual initiation torus cooling	RRB0001D	1.0E-3	
GG-1	ESMRPS22	11	1.0E-1	Operator failure to initiate SLCS or manually insert control rods	(none)	1.0E-1	
GG-1	ESADSM23	15	1.5E-3	Operator failure to manually initiate ADS	(none)	1.5E-3	

Table A-5. (Continued)

Plant Type: BWR

Category: Maintenance and Test

Plant[a]	Event Identification	Plant Event Rank	Total Event Unavailable	Equipment/Component/System General Description	PRA Identification Number	Portion of Event Unavailable	Notes
MP-1	ESMADS09	3	7.0E-2	Operator fails to manually depressurize reactor coolant system (RCS)	(none)	7.0E-2	
MP-1	ESICMU21		1.0E-2	Operator failure to manually open isolation condenser makeup (ICM) valve	(none)	1.0E-2	
PB-1	ESSBLC	1	1.0E-1	Operator fails to initiate standby liquid control system or manually initiate control rod insertion	(none)	1.0E-1	
PB-1	ESHPSWAC	6	3.0E-5	Operator failure to start any high pressure service water system (HPSWS) pumps in 24 hours	(none)	3.0E-5	

a. Arkansas Nuclear-One (ANO-1), Calvert Cliffs-Two (CC-2), Oconee-Three (OC-3), Sequoyah-One (SE-1), Crystal River-Three (CR-3), Surry-One (SU-1), Browns Ferry-One (BF-1), Grand Gulf-One (GG-1), Millstone Point-One (MP-1), Point Beach-One (PB-1).

APPENDIX B
EQUIPMENT AND COMPONENT SELECTION METHODS FOR ACCIDENT SEQUENCE PRECURSOR (ASP) PREDICTED SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

APPENDIX B
EQUIPMENT AND COMPONENT SELECTION METHODS FOR ACCIDENT SEQUENCE PRECURSOR (ASP) PREDICTED SIGNIFICANT CONTRIBUTORS TO SEVERE CORE DAMAGE

The ASP study results were used to identify the e/c determined to be predicted significant contributors to severe core damage (SCD) situations.[B-1,B-2] This appendix presents the details pertaining to the references and methods used to make this determination. A summary of the identified e/c actions has been presented in Tables 1 and 2 of the main body of this report.

Figure B-1 presents a graphic representation of the portions of the ASP results used, and the summary listings developed, to derive the e/c listing presented in Tables 1 and 2.

Table B-1 presents a sample of the ASP event listings for the 1969-1979 precursors (Reference B-1). A similar listing is presented in Reference 2 for the 1980-1981 precursors.

The importance ranking for each of the 169 precursors analyzed for the 1969-1979 period is the significance category (SC) as shown in one of the columns in Table B-1 which has been excerpted from Reference B-1. The lower the SC ranking, the higher the precursor probability (PP) that the event will occur (example; SC = 00, PP = 1 or SC = 20, PP = 0.01). No importance ranking, was given for the 1980-1981 precursors presented in a report to be published in 1984.[B-2] Mr. Wm. B. Cottrell of ORNL provided a preliminary ranking, listing the estimated top, middle, and bottom one-third of the precursors.[B-3] For the purpose of this study, the 1969-1979 precursors with a SC ranking of 40 or greater were used, and the top one-third (those most likely to occur) of the 1980-1981 precursors were used. It should also be noted that the PP must be multiplied by the frequency or failure probability to derive a probability number that has roughly the same meaning as the probability given in the ASEP results.

Exhibits B-1 and B-2 and Figures B-2 and B-3 have been excerpted from Reference B-2 to provide examples of the detailed precursor information provided by ASP. Exhibit B-1 and Figure B-2 essentially provide data and a description for the precursor event that actually occurred. Exhibit B-2 and Figure B-3 provide data and a description for the hypothesized event that was assumed to result from the precursor. The example presented is for the TMI-2 accident in which core damage did actually occur. Using the detailed information presented in the above tables and figures, the e/c considered to be primary contributors to severe core damage situations were identified.

Table B-2 presents a summary listing of e/c identified as primary contributors to SCD based on the precursor studies. Tables 1 and 2, presented in the main body of this report, present a summary of these lists.

References

B-1. J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents:1969-1979, A Status Report*, NUREG/CR-2497, June 1982.

B-2. J. W. Minarick et al., *Precursors to Potential Severe Core Damage Accidents:1980-1981, A Status Report*, NUREG/CR-3591, (to be published).

B-3. W. B. Cottrell, private communication, Union Carbide Corp., June 2, 1983.

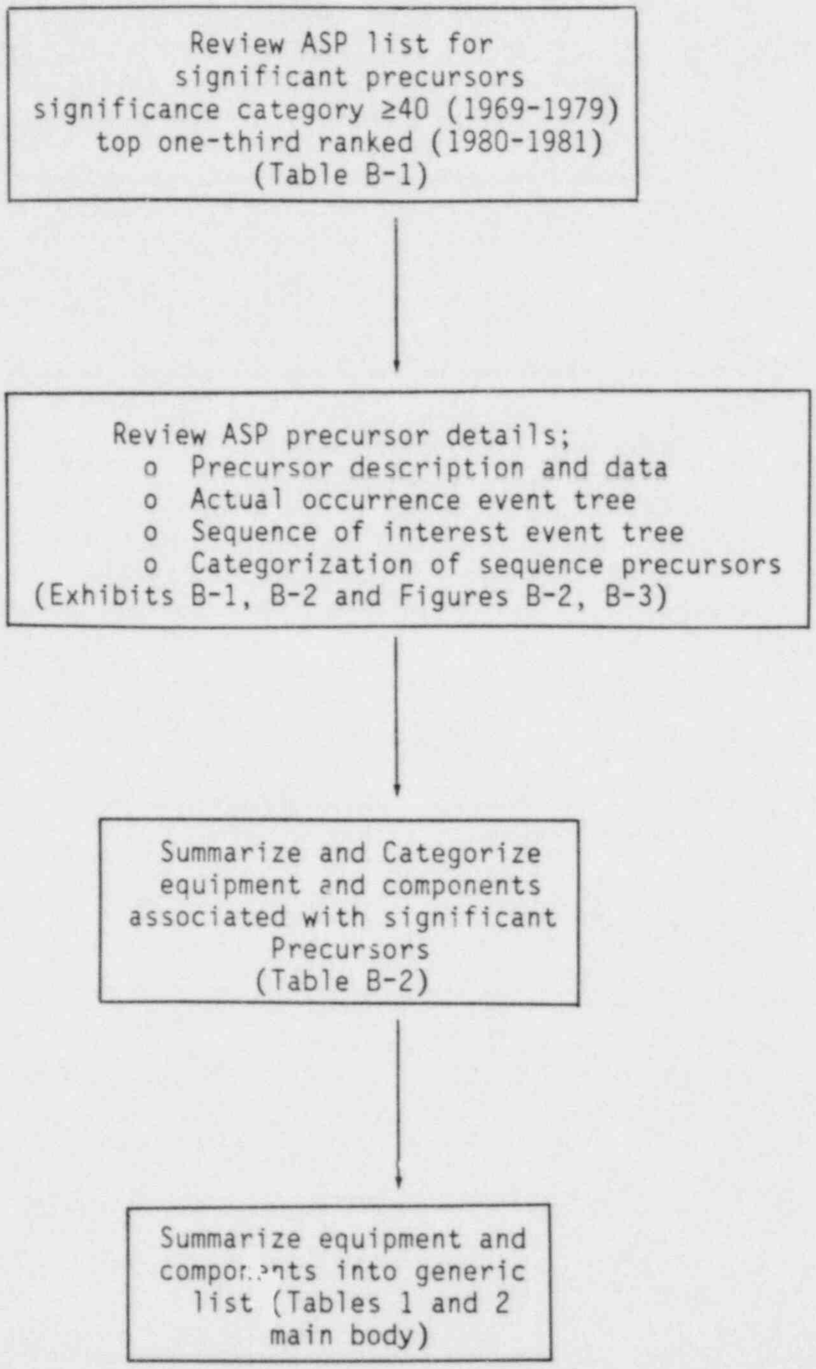


Figure B-1. Graphic representation—ASP equipment and components selection method.

Table B-1. Sample ASP precursors listing

Precursors listed by NSIC accession number

TABLE COLUMN HEADING ABBREVIATIONS

ACCESS: 6 DIGIT NSIC ACCESSION NUMBER
E DATE: EVENT DATE
SEQ: SEQUENCE OF INTEREST FOR THE EVENT
ACTUAL OCCURRENCE: DESCRIPTION OF EVENT
PLANT NAME: NAME OF PLANT AND UNIT NUMBER
DOC: PLANT DOCKET NUMBER
SY:SYSTEM ABBREVIATION:
COMPXX: SYSTEM COMPONENT CODE:
O: PLANT OPERATING STATUS:
D: DISCOVERY METHOD (O-OPERATIONAL EVENT, T-TESTING)
E: HUMAN ERROR INVOLVED (N-NO, Y-YES)

I: TRANSIENT/ACCIDENT INDUCED BY ACTUAL OCCURRENC (N-NO, Y-YES)
AGEX: PLANT AGE AT THE TIME OF THE EVENT IN DAYS
SC: SIGNIFICANCE CATEGORY
RATE: PLANT ELECTRICAL RATING IN MEGAWATTS ELECTRIC
T: PLANT TYPE (B=BWR, P=PWR)
V: PLANT NSSS VENDOR
AE: PLANT ARCHITECT ENGINEER
OPR: PLANT LICENSEE ABBREVIATION:
CRITXX: PLANT CRITICALITY DATE
SD: PLANT INDEFINATELY SHUT DOWN

ACCESS	E DATE	SEQ	ACTUAL OCCURRENCE	PLANT	DOC	SY	COMPXX	O	D	E	I	AGEX	SC	RATE	T	V	AE	OPR	CRITXX	S
32429	690313	MSLB	LOOP ISOLAT. VALVES FAILED TO SHUT	IND.POINT1	3	CB	VALVEX	G	T	Y	N	2415	75	265	P	B	UE	CEC	620802*	
36147	690715	LOOP	REACTOR TRIP WITH LOSS OF OFFSITE POWER	HAD.NECK	213	EB	CKTBRK	E	O	Y	Y	752	28	575	P	W	SW	CYA	670724	
39024	720125	LOOP	LOSS OF OFFSITE POWER	BG ROCK PT	155	EA	CKTBRK	E	O	N	Y	3406	39	72	B	G	BX	CPC	620927	
39380	711208	ECIT	SAFETY VALVE OPERATION AFTER LOFW	DRESDEN 3	249	CC	INSTRU	E	O	N	Y	312	36	794	B	G	SL	CWE	710131	
44751	700427	LOFW	3 OF 4 STEAM DRUM SCRAM SENSORS FAIL	DRESDEN 1	010	IC	INSTRU	U	T	N	N	3844	36	200	B	G	BX	CWE	591015	
47814	700505	ECIT	DEPRESSURIZATION INCIDENT	DRESDEN 2	237	CC	INSTRU	E	O	N	Y	150	36	794	B	G	SL	CWE	700107	
59484	710112	LOCA	SUMP ISO. VALVES CLOSED	PT.BEACH 1	266	SF	VALVEX	E	T	N	N	71	28	497	P	W	BX	WMP	701102	
60227	701204	MSLB	FAILURE OF SEVERAL MSIV'S TO CLOSE	DRESDEN 2	237	CC	VALVEX	E	T	N	N	331	59	794	B	G	SL	CWE	700107	
61043	710120	LOOP	LOSS OF OFFSITE POWER	LACROSSE	409	EA	RELAYX	G	O	Y	Y	1289	39	50	B	A	SL	DLP	670711	
61434	700717	LOOP	LOSS OF OFFSITE POWER	HUMBDLTBAY	133	EA	INSTRU	E	O	Y	Y	2708	21	65	B	G	BX	PGE	630216*	
61565	710902	LOOP	LOOP AND FAILURE OF A DIESEL GENERATOR TO LOAD	PALISADES	255	EA	RELAYX	U	O	N	Y	101	22	805	P	C	BX	CPC	710524	
63129	710324	LOOP	LOSS OF OFFSITE POWER	LACROSSE	409	EA	CKTBRK	E	O	N	Y	1352	18	50	B	A	SL	DLP	670711	
63144	710308	LOOP	FAILURE OF BOTH DIESEL GENER. DURING TESTING	ROBINSON 2	261	EE	ENGINE	E	T	N	N	169	29	700	P	W	EX	CPL	700920	
64600	701226	LOOP	SHUTDOWN OF BUCHANAN STATION	IND.POINT1	3	EA	RELAYX	U	O	N	Y	3068	32	265	P	B	UE	CEC	620802*	
65757	710205	LOOP	LOSS OF OFFSITE POWER	PT.BEACH 1	266	EA	CKTBRK	G	O	N	Y	95	28	497	P	W	BX	WMP	701102	
65757	710416	LOCA	FAILURE OF CONTAINMENT SUMP ISO VALVES	PT.BEACH 1	266	SF	VALVEX	C	T	N	N	165	35	497	P	W	BX	WMP	701102	
65969	710908	LOCA	OPEN ELECTROMATIC RELIEF VALVE	PALISADES	255	CB	VALVEX	G	O	Y	Y	107	49	805	P	C	BX	CPC	710524	
66996	711010	LOCA	TRANSIENT AND BLOWDOWN	MILLSTONE1	245	CC	VALVEX	E	O	N	Y	349	27	660	B	G	EX	NNE	701026	
71694	720517	LOOP	LOSS OF OFFSITE POWER	PALISADES	255	EB	RELAYX	G	T	N	Y	359	32	805	P	C	BX	CPC	710524	
73655	720720	LOOP	LOSS OF OFFSITE POWER	IND.POINT1	3	EA	ELECON	E	O	N	Y	3640	32	265	P	B	UE	CEC	620802*	
74242	711231	ECIT	HIGH COOLANT LEVEL	NINEMIPT1	220	CC	VESSEL	E	O	Y	Y	847	36	620	B	G	UX	NMP	690905	
75074	720817	LOOP	LOSS OF LOAD	LACROSSE	407	EA	RELAYX	C	O	Y	Y	1863	39	50	B	A	SL	DPL	670711	
77916	721229	LOCA	MALFUNCTION OF SEVERAL VALVES	OYSTER CRK	219	CH	VALVEX	E	O	Y	Y	1334	37	650	B	G	RB	JCP	690503	
78418	730202	LOCA	VALVE FAILURE TO OPEN DURING TESTING	MA.YANKEE	309	SF	VALVEX	E	T	N	N	102	3[illegible]	825	P	C	SW	MAY	721023	
79565	721201	LOOP	LOSS OF NORMAL STATION POWER	VT.YANKEE	271	EB	TRANSF	E	O	N	Y	252	32	514	B	G	EX	VYC	720324	
80138	720610	LOOP	FLOODING OF TURBINE BUILDING	QUAD-CTES1	254	CF	PUMPXX	G	O	Y	N	236	34	789	B	G	SL	CWE	711018	
81523	730618	LOFW	FAILURE OF AFW PUMPS TO AUTO-START	TKY.POINT4	251	SF	[illegible]TRU	B	O	Y	N	7	29	693	P	W	BX	FPL	730611	

Table B-1. (Continued)

153003	791024	LOCA	FAILURE TO PROPERLY POSITION VALVES AFTER TEST	OCONEE 2	270	SH	VALVEX	E	O	Y	N	2173	43	887	P	B	UX	DPC	731111
153164	790328	LOFW	LOSS OF FEEDWATER & OPEN PORV	TMI 2	320	CJ	VALVEX	E	O	Y	Y	365	00	906	P	B	BR	MEC	780328*
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 1	269	EE	ENGINE	E	O	N	N	2351	52	887	P	B	UX	DPC	730419
153167	790926	LOOP	LOSS OF STANDBY POWER SOURCE DURING PWR OPRT	OCONEE 2	270	EE	ENGINE	E	O	N	N	2145	52	887	P	B	UX	DPC	731111
153333	791115	LOFW	AFW UNAVAIL DUE TO MAINTENANCE AND INSPECTION	COOK 1	315	SF	PUMPXX	G	T	Y	N	1762	40	1054	P	W	AE	IME	750118
153338	791101	LOOP	INCORRECT DIESEL GENERATOR MAINT. WHILE REFUEL	COOK 2	316	EE	ENGINE	H	O	Y	N	601	67	1100	P	W	AE	IME	780310
153686	791127	LOOP	DG AND CHARGING PMP SUCT VALVE UNAVAILABLE	BVRVALLEY1	334	EE	ENGINE	E	O	Y	N	1296	71	852	P	W	SW	DLC	760510
153810	791120	LOFW	RCIC TURBINE TRIP WITH HPCI UNAVAILABLE	BRUNSWICK1	325	CE	MECFUN	E	O	N	Y	1139	25	821	B	G	UE	CPL	761008
154286	791128	LOCA	ONE PMP FAILED + 1 DEGRADED IN CLD SHTDN	SALEM 1	272	SF	PUMPXX	G	O	N	N	1082	35	1090	P	W	UX	PEG	761211
154639	791209	LOOP	BOTH DIESEL GENERATORS INOPERABLE	DVS-BESSE1	346	EE	ENGINE	G	T	N	N	849	73	906	P	B	BX	TEC	770812

ABBREVIATIONS:

ACCESS: 6 DIGIT NSIC ACCESSION NUMBER
E DATE: EVENT DATE
SEQ: SEQUENCE OF INTEREST FOR THE EVENT
- ECIT - EXCESSIVE COOLANT INVENTORY
- EQUK - EARTHQUAKE
- INAA - INADVERTANT ADS ACTUATION
- LOFW - LOSS OF FEEDWATER
- LOOP - LOSS OF OFFSITE POWER
- LOCA - LOSS OF COOLANT ACCIDENT
- LRTR - LOCKED ROTOR ACCIDENT
- MSLB - MAIN STEAM LINE BREAK
- RCPT - REACTOR COOLANT PUMP TRIP
- SGTR - STEAM GENERATOR TUBE RUPTURE

ACTUAL OCCURRENCE: DESCRIPTION OF EVENT
PLANT NAME: NAME OF PLANT AND UNIT NUMBER
DOC: PLANT DOCKET NUMBER
SY:SYSTEM ABBREVIATION:

STANDARD GENERIC CODE	SYSTEM DESCRIPTION
	REACTOR
RA	REACTOR VESSEL INTERNALS
RB	REACTIVITY CONTROL SYSTEMS
RC	REACTOR CORE

PRECURSOR DESCRIPTION AND DATA

NSIC Accession Number: 153164

Date: March 28, 1979

Title: Core Damage and Radioactivity Release Occurs at Three Mile Island 2

The failure sequence was:

1. A condensate pump trip while at 99% power resulted in a subsequent trip of both feedwater pumps and a consequent turbine trip and eventual reactor trip.
2. The pressurizer pilot operated relief valve (PORV) opened and stuck open.
3. Auxiliary feedwater flow to the steam generators was blocked by three improperly closed AFW system valves. (AFW was manually initiated 8 minutes after turbine trip.)
4. Decreasing RCS pressure due to the stuck open PORV initiated High Pressure Injection at 1600 psig.
5. Increasing pressurizer level indication caused the operator to infer an increasing pressurizer level. The operator erroneously throttled HPI flow (10-12 minutes after turbine trip).

(see attached sheet)

Corrective action:

1. The stuck open PORV was discovered and its isolation valve closed.
2. The RCS was repressurized and a reactor coolant pump started to provide core cooling.
3. An alternate cooling scheme was installed to permit eventual core cooling without using the decay heat removal system, which would have required pumping highly radioactive reactor coolant outside containment.

Design purpose of failed system or component:

1. The PORV provides for pressure relief for the RCS. It opens before the code safety valves open and prevents their lifting for small pressure excursions.
2. The AFW system provides for core cooling via the steam generators when the main feedwater system is inoperable.
3. The HPI system provides high pressure borated water to the RCS in the event of a small break in the reactor coolant system.

Unavailability of system per WASH 1400:* AFW: 2.5×10^{-4}/D (start + 8 hours)
HPI: 1.2×10^{-2}/D

Unavailability of component per WASH 1400:* PORV, failure to reseat: 10^{-2}/D

general human error: 5×10^{-3}/D

*Unavailabilities are in units of per demand D^{-1}. Failure rates are in units of per hour HR^{-1}.

Exhibit B-1. Precursor description and data

The failure sequence was: (continued)

6. Between one and two hours after turbine trip, the operator stopped all Reactor Coolant Pumps because of pump vibration. Due to the loss of forced core cooling and the inability to cool the core by natural circulation because of voids which existed in the RCS as a result of inadequate HPI flow, core damage occurred.

Exhibit B-1. (Continued)

CATEGORIZATION OF ACCIDENT SEQUENCE PRECURSORS

NSIC ACCESSION NUMBER: 153164

DATE OF LER:

DATE OF EVENT: March 28, 1979

SYSTEM INVOLVED. reactor coolant system, auxiliary feedwater, high pressure injection

COMPONENT INVOLVED: pilot operated relief valve

CAUSE: stuck open PORV, isolated AFW system, operator errors related to high pressure injection and RC pumps, human error

SEQUENCE OF INTEREST: loss of feedwater

ACTUAL OCCURRENCE: loss of feedwater and subsequent failed open PORV, degraded AFW system, failed high pressure injection, and RC pump trip result in core damage.

REACTOR NAME: Three Mile Island 2

DOCKET NUMBER: 50-320

REACTOR TYPE: PWR

DESIGN ELECTRICAL RATING: 906 MWe

REACTOR AGE: 1.0 yr

VENDOR: B&W

ARCHITECT-ENGINEERS: Burns & Roe

OPERATORS: Metropolitan Edison

LOCATION: 10 mi SE of Harrisburg, Pa.

DURATION: N/A

PLANT OPERATING CONDITION: 99% power

SAFETY FEATURE TYPE OF FAILURE: (a) inadequate performance; (b) failed to start; (c) made inoperable; (d) ______

DISCOVERY METHOD: operational event

COMMENT: A formal LER on this event has not been received at NSIC.

Exhibit B-2. Categorization of accident sequence precursors

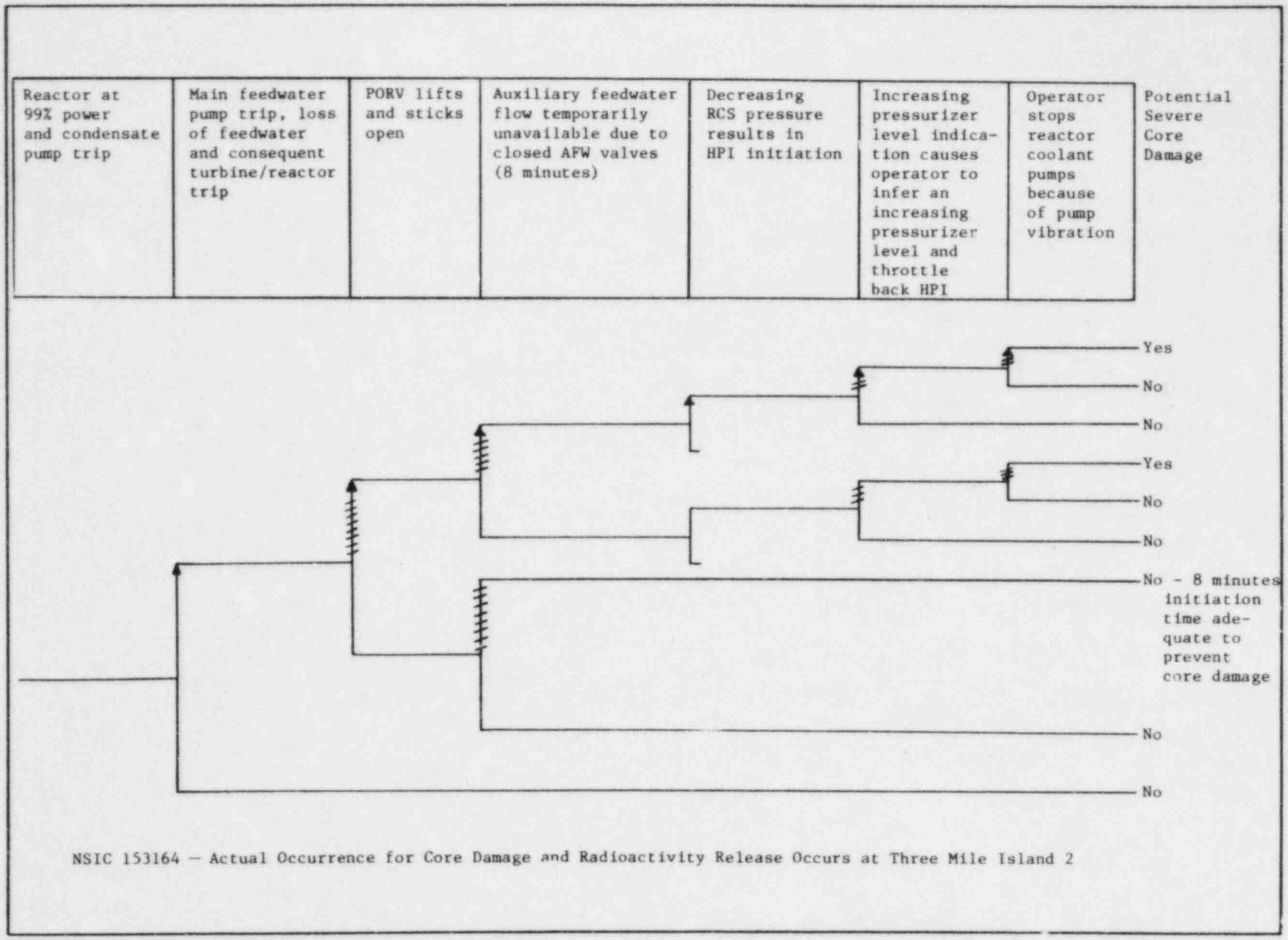
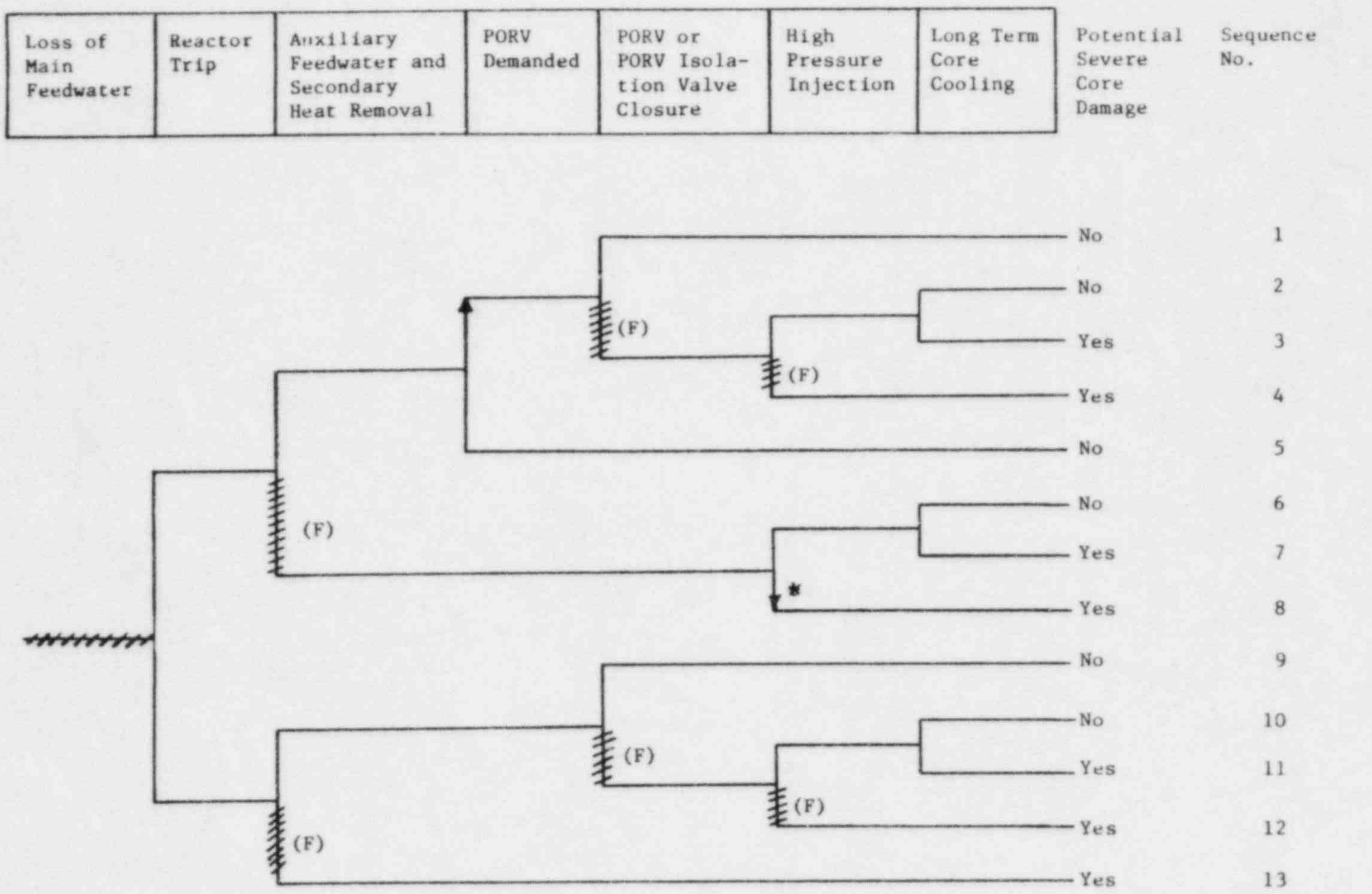


Figure B-2. Actual occurrence event tree.



NSIC 153164 — Sequence of Interest for Core Damage and Radioactivity Release Occurs at Three Mile Island 2

***Not included in mitigation procedures.**

Figure B-3. Sequence of interest event tree.

Table B-2. Summary: equipment and components identified as significant contributors based on ASP report

Plant Type: PWR

Category: Valves

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
59484	Point Beach-1	LOFW	28	1.58E-3	Sump isolation system	Valves failed shut
65757	Point Beach-1	LOFW	35	3.16E-4	Sump isolation system	Valves failed shut
120995	Rancho Seco	LOFW	35	3.16E-4	Sump isolation system	Valves failed shut
153164	TMI-2	LOFW	1	1	Reactor coolant, feedwater and auxiliary feedwater system	PORV stuck open after multiple trips
149250	Trojan	MSLB[a]	16	2.51E-2	Steam system	Main steam isolation valve failed to close; crud or hardened packing
97107	Zion-2	MSLB	36	2.51E-4	HPI; boron injection failure	B train valve failure
130788	Davis Besse-1	LOCA	29	1.29E-3	Feedwater and primary coolant system	Spurious signal caused trip and underpower; PORV stuck open
137918	TMI-2	MSLB	22	6.31E-3	Steam and instrument systems	Instrumentation spike caused reactor trip; steam relief stuck open
148764	Beaver Valley-1	MSLB	29	1.26E-2	Steam system	Air dampers improperly positioned; ice formation causes steam dump valve to freeze in open position

Table B-2. (Continued)

Plant Type: PWR

Category: Pumps

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
91676	Turkey Point-3	LOFW	16	2.51E-2	Feedwater system	Overtightened packings
133706	Davis Besse-1	LOFW	16	2.51E-2	Loss AFW pump control	Mechanical binding
133705	Farley	LOFW	19	1.26E-2	Feedwater system	Turbine pumps fail to start
154286	Salem-1	LOFW	35	3.16E-4	Safety injection system inoperable	Safety injection pumps failed
103077	Robinson-2	Small LOCA	26	2.51E-3	Primary coolant system	Reactor coolant pump shaft seal failure
89205	Surry 1	LRTR[b]	30	1.00E-3	Reactor coolant pump	Pump shaft broke

Table B-2. (Continued)

Plant Type: PWR

Category: Instruments

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
138830	Rancho Seco	LOFW	06	2.51E-1	Auxiliary feedwater system	Non-nuclear instruments cause insufficient flow

Table B-2. (Continued)

Plant Type: PWR

Category: Electrical: Controller, Fuses, Breakers, Relay and Cables

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
91676	Turkey Point-3	LOFW	16	2.51E-2	Feedwater system	Malfunctioning controller
133706	Davis Besse-1	LOFW	16	2.51E-2	Loss AFW feedpump control	Blown fuses: controller power
61565	Palisades	LOOP	22	6.31E-3	Site power	Failed relay cause tripped
61757	Point Beach-1	LOOP	28	1.58E-3	Site power	Breaker opening line
71694	Palisades	LOOP	32	6.31E-4	Site power	Spurious relay operation
82198	Turkey Point-3	LOOP	39	1.26E-4	Site power	Unconnected wire (cable)
85370	Ginna	LOOP	28	1.58E-3	Site power	Line trip due to high generated power
93702	Turkey Point-4	LOOP	32	6.31E-4	Site power	Sneak circuit and isolated start-up transfer
97578	Palisades	LOOP	33	5.01E-4	Site power	Spurious relay operation
125563	St. Lucie-1	LOOP	34	3.98E-4	Site power	Grid disturbances
128935	Ft. Calhoun	LOOP	32	6.31E-4	Site power	Defective relay reset mechanism
132943	Palisades	LOOP	32	6.31E-4	Site power	"R" bus de-energized
132958	Palisades	LOOP	32	6.31E-4	Site power	"R" bus de-energized
137543	Calvert Cliffs-1	LOOP	23	5.01E-3	Site power	Switch yard protective relays
140335	Beaver Valley-1	LOOP	23	5.01E-3	Site power	Main transformer faults and improper relay operation
152187	St. Lucie-1	LOOP	25	3.16E-3	Site power	Switch yard failure
152951	Davis Besse-1	LOOP	36	2.51E-4	Site power	Switch yard failure
103207	Turkey Point-4	LOOP	39	1.26E-4	Emergency site power	Diesel generators failed to load; breaker and relay fail
150882	Crystal River	LOOP	30	1.00E-3	Emergency site power	Diesel trip during test; reactive load imbalance
141097	TMI-2	Small LOCA	30	1.00E-3	Primary coolant system	Inverter failure; open PORV
97107	Zion-2	MSLB	36	2.51E-4	HPI; boron injection failure	A train power supply failure
					pressure relief systems	signal caused plant shutdown, primary relief stuck open during plant depressurization
161906	Quad Cities-2	LOFW	--	--	RCIC and HPCI systems inoperable	Faulty torque switch on RCIC discharge isolation valve. Oil leak in HPCI turbine stop valve
166082	Brunswick-2	LOFW	--	--	HPCI system valve RCIC system speed control	HPCI injection valve failed in open position; failed windings resistor failed in RCIC speed control governor

Table B-2. (Continued)

Plant Type: PWR

Category: Maintenance and Test

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
90421	Point Beach-1	LOFW	16	2.51E-2	Feedwater systems	Clogged strainers
108078	Kewaunee	LOFW	16	2.51E-2	Feedwater systems	Clogged strainers
64600	Indian Point-1	LOOP	32	6.31E-4	System substation transformer	Single transformer tripped during inspection of second transformers; procedures changed
103111	Oconee-3	LOOP	40	1.00E-4	Condenser system	Discharge select valves improperly positioned; inadequate procedures
120293	Hatch-1	LOOP	30	1.00E-3	Plant service water	Plugged strainers (strainer drive motors also failed)
153333	Cook-1	LOFW	40	1.00E-4	Auxiliary feedwater	Motor control center was removed for service inspection while AFW pump was out of service for repair

Table B-2. (Continued)

Plant Type: PWR

Category: Systems and Miscellaneous Hardware

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
152563	Praire Island-1	SGTR	27	2.00E-3	RC[c] and steam supply system	Steam generator tube break

Table B-2. (Continued)

Plant Type: PWR

Category: Weather

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
78418	Maine Yankee	LOFW	34	3.98E-4	Charging system	Ice formation causes charging pump valves to fail open
117944	Millstone-2	LOOP	32	6.31E-4	Site power	Hurrican and salt spray
130111	Cook-1	LOOP	39	1.26E-4	Site power	Multiple lightening strikes
130119	Palisades	LOOP	32	6.31E-4	Site power	Electrical storm
135006	Farley	LOOP	32	6.31E-4	Site power	Lightening strike caused relay operation
143219	Indian Point-3	LOOP	33	5.01E-4	Site power	Severe electrical storms

Table B-2. (Continued)

Plant Type: PWR

Category: Human Error

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
137305	Farley	LOFW	19	1.26E-2	Feedwater system	Open bypass valve; turbine pump plumbing
81523	Turkey Point-4	LOFW	29	1.26E-3	Feedwater system	Fuses not installed
153164	TMI-2	LOFW	00	1	Auxiliary feedwater system	Erroneously closed valves
153164	TMI-2	LOFW	00	1	High pressure injection system	Flow incorrectly reduced
132927	Davis Besse-1	LOOP	29	1.26E-3	Site power	Premature manual transfer
139565	St. Lucie-1	LOOP	23	5.01E-3	Site power	Improper switching and connections
63144	Robinson-2	LOOP	29	1.26E-3	Emergency power	Diesel generator didn't run due to instrument re-routing
116212	Millstone-2	LOOP	20	1.00E-2	Emergency power	Safety busses failed to load due to incorrect undervoltage set points
145209	Fort Calhoun	Small LOCA	37	2.00E-4	Primary coolent	Technician removed fuses during trouble-shooting; causes open PORV
36147	Hadden Neck	LOOP	28	1.58E-3	Reactor system; trip	Procedural error caused trip
115875	Connetticut Yankee	LOOP	29	1.26E-3	Site electrical	Maintenance switching caused spurious signal
123118	Rancho Seco	LOFW	36	2.51E-4	Auxiliary feedwater	Failure to properly reset AFW pump breaker
127284	Zion-2	LOCA	29	1.26E-3	Reactor protection system	System logic test dummy loads improper; procedures changed
146744	Davis Besse-1	LOFW	30	1.00E-3	Hydrogen analyzer system	Accidental grounding of electrical system causes loss of vital bus and trip
148764	Beaver Valley-1	MSLB	29	1.26E-3	Steam dump and heater drain	Ice formation causes steam dump valve to freeze open; air dampers improperly positioned

Table B-2. (Continued)

Plant Type: BWR

Category: Pumps

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
105540	Quad Cities-2	LOCA	38	1.60E-4	Feedwater system pump and piping	Extreme feedwater pump vibration caused suction line pipe fitting breakage
106333	Quad Cities-2	LOFW	31	7.90E-4	Reactor core isolation cooling and high pressure coolant injection systems	Failed auxiliary lubrication oil pump line

Table B-2. (Continued)

Plant Type: BWR

Category: Valves

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure: SC	Precursor Probability Measure: Probability	Affected System or Component	Event Cause
66990	Millstone-1	LOCA	27	2.00E-3	Reactor coolant and turbine systems	Turbine control system malfunctioned: primary relief valve stuck open
77916	Oyster Creek	LOCA	[illegible]7	2.00E-4	Reactor coolant and generator monitoring protective system	Human error caused reactor scram and relief valve to stick open
85738	Browns Ferry-1	LOFW	26	3.20E-4	Reactor core isolation cooling and high pressure coolant injection systems	RCIC steam supply valve failure to open during startup test
106616	Pilgrim-1	LOOP	30	1.00E-3	Reactor coolant and steam supply systems	Turbine trip on reactor shut-down; resultant reactor trip and relief valve stuck open
115870	Vermont Yankee	LOCA	28	1.60E-3	Automatic depressurization system failure	Failure of air operators on relief valves
120439	Quad Cities-1	LOFW	33	5.00E-4	Automatic depressurization system failure	Electromatic relief valves fail to open
120443	Quad Cities-2	LOFW	28	1.60E-3	Automatic depressurization system failure	Electromatic relief valves fail to open
124222	Duane Arnold	LOFW	28	1.60E-3	None: bench tests	6 each main steam relief valves failed to open during bench tests
128569	Brunswick-2	LOCA	27	2.00E-3	Turbine, reactor, and reactor coolant systems	Safety relief valve failed to close during pressure control
149961	Hatch-1	LOFW	19	1.20E-2	RCIC and HPCI systems	Failed turbine stop valve
158229	Dresden-3	LOFW	--	--	Scram discharge system	Level detector switch failure and vent check valve failure
160497	Pilgrim-1	MSLB	--	--	Reactor coolant system	High pressure setting on, and leakage thru solenoid valve caused primary relief valve to open and leakage from RCS
160559	Pilgrim-1	MSLB	--	--	Reactor coolant system	Nitrogen pressure regulator frozen in open position causes RCS relief valve to open and leakage from RCS
160926	Pilgrim-1	MSLB	--	--	Reactor coolant and	Main steam line high radiation
123150	Crystal River-3	MSLB	29	1.29E-3	Steam system	Inverter B output divide failed caused steam dump
142462	Salem-1	LOCA	23	5.0[illegible]E-3	Electrical, RCS, and AFW	Failure of output transformers and resistors
152183	Hadden Neck	LOCA	36	2.51E-4	RC system and pressure-izer	Pressurizer pressure controller failed: pressurized PORV openeu

Table B-2. (Continued)

Plant Type: BWR

Category: Instruments

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
150499	Hatch-2	LOFW	36	2.50E-4	Steam supply, HPCI and RCIC systems	During test HPCI and RCIC were isolated because ΔP switch was set too high
158229	Dresden-3	LOFW	--	--	Scram system	Vent check valve and level switch mechanical failures
163478	Hatch-1	LOFW	--	--	HPCI speed controller and RCIC trip switch and relay	Erroneous high water level signal for reactor tripped feedwater pumps subsequently resulting in scram

Table B-2. (Continued)

Plant Type: BWR

Category: Electrical: Fuses, Connectors

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
128906	Cooper	LOFW	19	1.20E-2	Electrical power, feedwater, HPCI, and RCIC systems	Blown fuses on "no break" power panel; HPCI governor not functioning
153810	Brunswick-1	LOFW	25	3.20E-4	Reactor, RCIC, HPCI, and turbine	Cracked electrical connector on turbine tachometer

Table B-2. (Continued)

Plant Type: BWR

Category: Maintenance and Test

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure: SC	Precursor Probability Measure: Probability	Affected System or Component	Event Cause
163405	Browns Ferry-3	LOFW	--	--	Reactivity control; scram discharge volume	Approximately 76 control rods failed to insert fully on scram due to poorly designed scram discharge volume level switch system
163478	Hatch-1	LOFW	--	--	HPCI speed controller RCIC trip switch and relay	HPCI: turbine speed controller out of calibration; caused high steam ΔP; RCIC limit switch and/or relay faulty
166072	Brunswick-1	LOCA	--	--	Service water system	Sea life (oyster shells) clogged water intake duct
85566	Browns Ferry-1	LOOP	25	3.20E-4	RCIC and HPCI systems	AC power loss during shutdown due to poor electrical power tripping logic. Poor design

Table B-2. (Continued)

Plant Type: BWR

Category: Systems and Miscellaneous Equipment

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Precursor Probability Measure Probability	Affected System or Component	Event Cause

(No items in this category identified for BWRs)

Table B-2. (Continued)

Plant Type: BWR

Category: Human Error

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure SC	Probability	Affected System or Component	Event Cause
116780	Millstone-1	LOOP	28	1.60E-3	Electrical power system	LOOP due to salt buildup on lines and insulators. Subsequent trip due to auxiliary generators out of phase
160497	Pilgrim-1	MSLB	--	--	Reactor coolant system	Pressure setting on solenoid valve to RCS relief valve too high; relief opened causing RCS leakage
86990	Nine Mile Point-1	LOOP	39	1.30E-4	Electrical power system	Electrician bumped relay in auxiliary control room causing it to open and a LOOP for 10 second duration
102146	Brunswick-2	LOOP	39	1.30E-4	Electrical power system	230 KV busses tied incorrectly causing LOOP: tie corrected
101444	Browns Ferry-1	LOFW	04	4.0E-1	Electrical distribution	Cable tray fire causes anomalous control and instrument behavior. Operator scrams reactor. Fire due to inspection of cabling.
83833	Oyster Creek	LOOP	37	2.0E-4	Electrical power system	Differential relay on 4160 VAC bus improperly set during maintenance

Table B-2. (Continued)

Plant Type: BWR

Category: Weather

Accession Number (NSIC)	Plant	Initiator Event	Precursor Probability Measure		Affected System or Component	Event Cause
			SC	Probability		
116780	Millstone-2	LOOP	28	1.60E-3	Electrical power system	Salt buildup on lines and insulators caused LOOP

a. Main steam line break (MSLB).

b. Locked rotor (LRTR).

c. Recirculation (RC).

NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-3762
EGG-2311

2. (Leave blank)

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4. TITLE AND SUBTITLE
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The Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, sponsored the Equipment Qualification Research Program which performed a survey of applicable severe accident study reports to aid in focusing the program efforts. The objective of the survey was to identify, where possible, equipment and components that have been predicted to be significant contributors to high probability accident sequence resulting in severe core damage. A survey of the results of the survey is presented in Tables 1 and 2 of this report. Future updates of this report are anticipated as applicable risk study reports become available.

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