UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 26, 1994

Docket Nos. 50-334
and 50-412

Mr. J. D. Sieber, Vice President
and Chief Nuclear Officer
Nuclear Power Division
Post Office Box 4
Shippingport, Pennsylvania 15077-0004

Dear Mr. Sieber:

SUBJECT: TRANSMITTAL OF NRC CONTRACTOR ANALYSIS OF BEAVER VALLEY OPERATIONAL EVENTS

The purpose of this letter is to transmit, for your review and comment, our contractor's (ORNL) analysis of certain operational events which occurred at Beaver Valley Units 1 and 2 in 1993. Our contractor's analysis is part of NRC's Accident Sequence Precursor (ASP) analysis program.

In recent years, licensees of U.S. nuclear power plants have added safety equipment, and have improved plant and emergency operating procedures. Some of these changes, particularly those involving use of alternate equipment or recovery actions in response to specific accident scenarios, are not currently incorporated in the basic ASP models. Consequently, the ASP estimates of core damage probabilities could be conservative for certain accident sequences. To address this issue, we are providing each preliminary ASP analysis to the pertinent plant licensee for Peer Review. The licensee is requested to review and comment on the technical adequacy of the analyses, including the depiction of their plant equipment and equipment capabilities. We will then evaluate the comments received during this Peer Review for reasonableness and pertinence to the ASP analysis in an attempt to use best estimate values. Upon completion of this evaluation, we will revise the conditional core damage probability calculations where necessary to consider information provided by the licensee during the review. The object of the Peer Review process is to provide as realistic an analysis of the significance of the event as possible. This year, we are sending the preliminary analyses out for Peer Review as they are completed, rather than in a batch mode, as was done with the 1992 events reviewed last year.

In order to maintain our schedule for issuance of the 1993 Precursor Report, we request you provide any comments within 30 days from receipt of this letter. In order to facilitate your review, we have provided two sets of 5 enclosures. One set is for the October 12, 1993, loss of offsite power event at Units 1 and 2, and the other set is for the November 4-6, 1993, emergency diesel generator load sequencer event at Unit 2. In each set Enclosure 1 is the preliminary precursor analysis. Enclosure 2: (1) contains specific guidance for the Peer Review, (2) identifies the criteria which we will apply to determine whether any credit should be given in the analysis for the use of licensee-identified additional equipment or specific actions in recovering from the event, and (3) describes the specific information that should be

DF01

provided by the licensee to support such a claim. Enclosure 3 is the licensee event report (LER) documenting the subject event. Enclosures 4 and 5 contain background information regarding the ASP methodology which may be useful to the licensee in reviewing the analysis. Enclosure 4, which is Section 2.0 from the 1992 ASP Annual Precursor Report, describes the precursor event identification and quantification process. Enclosure 5, which is Appendix A from the same report, describes the ASP models used in precursor analyses.

No new OMB clearance is needed for the ASP Peer Review process, since the process is already covered by the existing OMB clearance addressing staff followup review of events documented in LERs.

Sincerely,

Original signed by:
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

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No new OMB clearance is needed for the ASP Peer Review process, since the process is already covered by the existing OMB clearance addressing staff followup review of events documented in LERs.

Sincerely,

G E Edison

Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

Mr. J. D. Sieber
Duquesne Light Company

Beaver Valley Power Station
Units 1 & 2

cc:

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UNITS 1 AND 2

ENCLOSURES FOR

OCTOBER 12, 1993 EVENTS

ENCLOSURE 1

0.1 LER Number 334/93-013

Event Description: Dual Unit Loss of Offsite Power

Date of Event: October 12, 1993

Plant: Beaver Valley 1 and 2

0.1.1 Summary

On October 12, 1993, the Beaver Valley site experienced a dual unit loss of offsite power (LOOP). Unit 1 had been operating at 100% power and unit 2 was in refueling at the time of the event. The conditional core damage estimated for this event is 6.2×10^{-5}. The LOOP was not modeled for Unit 2 since it was in refueling shutdown at the time of the event.

0.1.2 Event Description

On October 12, 1993, Beaver Valley Unit 1 was operating at 100% power with normal station loads being supplied from the unit station service transformers (USST). Unit 2 was in a refueling outage with all fuel stored in the spent fuel pool. Unit 2 loads were being supplied from a backfeed through the main unit transformer.

At 1507 h, Unit 1 experienced a loss of the majority of its load when 10 offsite feeder breakers in the switchyard opened including the Unit 1 output breaker, PCB 341, and the Unit 2 output breaker, PCB 362. Loss of load in Unit 1 caused an overspeed trip of the turbine-generator. Generator speed peaked at 2051 rpm. This increase in generator speed caused a corresponding increase in reactor coolant pumps (RCPs) speed. The resulting flow transient caused a reactor trip on high flux rate.

Following the Unit 1 trip, all three auxiliary feedwater (AFW) pumps started and the three RCPs tripped on underfrequency. Thirty seconds after the turbine trip, the generator output breakers opened as designed. The Unit 1 main generator had been the only source of power to both units following the opening of the switchyard breakers. The trip of the Unit 1 generator caused a loss of offsite power to both units. The Unit 1 emergency diesel generators (EDGs) sequenced loads on their busses, and a natural circulation cooldown was established using AFW and the steam generator power operated relief valves (SG PORVs). At 1517 h, power was restored to the switchyard, and forced reactor coolant system (RCS) flow was reestablished. The safety-related busses were subsequently realigned to offsite power, and the EDGs were shut down.

Following the Unit 1 trip, the Unit 2 2-1 EDG sequenced all available train A safety-related loads including the low-pressure injection (LPI) pump. However, LPI did not inject any water into the RCS since the discharge valves were closed for refueling. The 2-2 EDG and associated safeguards bus had been removed from service for outage-related maintenance at the time of the event. Offsite power was restored to Unit 2 at 1522 h. The train A safety-related bus was repowered from offsite power at 1535 h, and the 2-1 EDG was shutdown.

Following the Unit 1 reactor trip, a small RCS leak was noted at the loop 1A cold leg vent valve, RC-27. Unit 1 then commenced a cooldown to cold shutdown. The leak was caused by a fillet weld failure.

The LOOP event was caused by an error during scheduled maintenance on the Unit 2 main output breaker. Continuity checks were being conducted on the auxiliary contacts for relays associated with the Unit 2 output breaker, PCB 352. During this process, underfrequency tripping relays were actuated when 125 Vdc from one set of contacts was inadvertently connected to another set of contacts in the underfrequency separation scheme via the multimeter. As a result, seven 345 kV breakers and three 138 kV breakers opened.

0.1.3 Additional Event-Related Information

Units 1 and 2 share a common 138 kV and 345 kV switchyard (see Fig. 1). The 138 kV and 345 kV switchyards are connected by two auto transformers. Numerous offsite lines originate in both sections of the switchyard. The output of both main generators can be aligned to feed both of the 345 kV switchyard busses. Each main generator also feeds two USSTs. There are two system station service transformers (SSST) for each unit. One of the SSSTs for each unit is fed from each of the 138 kV substation busses, busses 1 and 2. Each SSST and associated USST feed two nonsafety-related 4160 Vac busses, busses A, B, C, and D. During power operations, busses A through D are aligned to the USST. Upon a trip of the turbine-generator, the busses fast transfer to the SSSTs. Busses A and D each feed a safety-related 4160 Vac bus, busses AE and DF. Each safety-related bus has an associated emergency diesel generator that will load on a sustained loss of voltage.

The automatic loading capability of the EDGs on a safety injection (SI) signal was inoperable for both Unit 2 EDGs (see LER 412/93-012) at the time of this event. This failure would only occur when an SI signal is present coincident with a loss of the normal engineered safety feature (ESF) bus power supply. The failure mechanism had existed since November 1990. Operator actions would have been necessary to allow manual loading of equipment on the ESF busses. Since Unit 2 was in refueling at the time of the event, this does not impact the analysis.

0.1.4 Modeling Assumptions

This event was modeled as a plant-centered loss of offsite power to Unit 1. The short- and long-term nonrecovery values and seal LOCA probabilities were modified to reflect the plant-centered LOOP values (see ORNL/NRC/LTR-98-11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989).

The RCS leak associated with RC-27 was small enough that it was well within the capabilities of the charging system. This was considered to have no impact on the sequence of other events or on the viability of operator recovery of other systems.

It was assumed that the maintenance work conducted on the generator output breaker relays would only be done on a unit that was shutdown. In addition, the Unit 2 LOOP was of short duration, and all fuel had been moved to the spent fuel pool. As a result, the Unit 2 transient was not modeled. Since the Unit 2 transient was not modeled, the inoperability of the EDG load sequencer on a simultaneous LOOP and SI signal did not impact the analysis.

0.1.5 Analysis Results

The estimate of the conditional core damage probability for this event is 6.2×10^{-5}. The dominant core damage sequence, shown in Fig. 2, involves a LOOP, followed by successful trip, failure of emergency power, successful AFW actuation, a reactor coolant pump seal LOCA, and failure to recover offsite power long term.

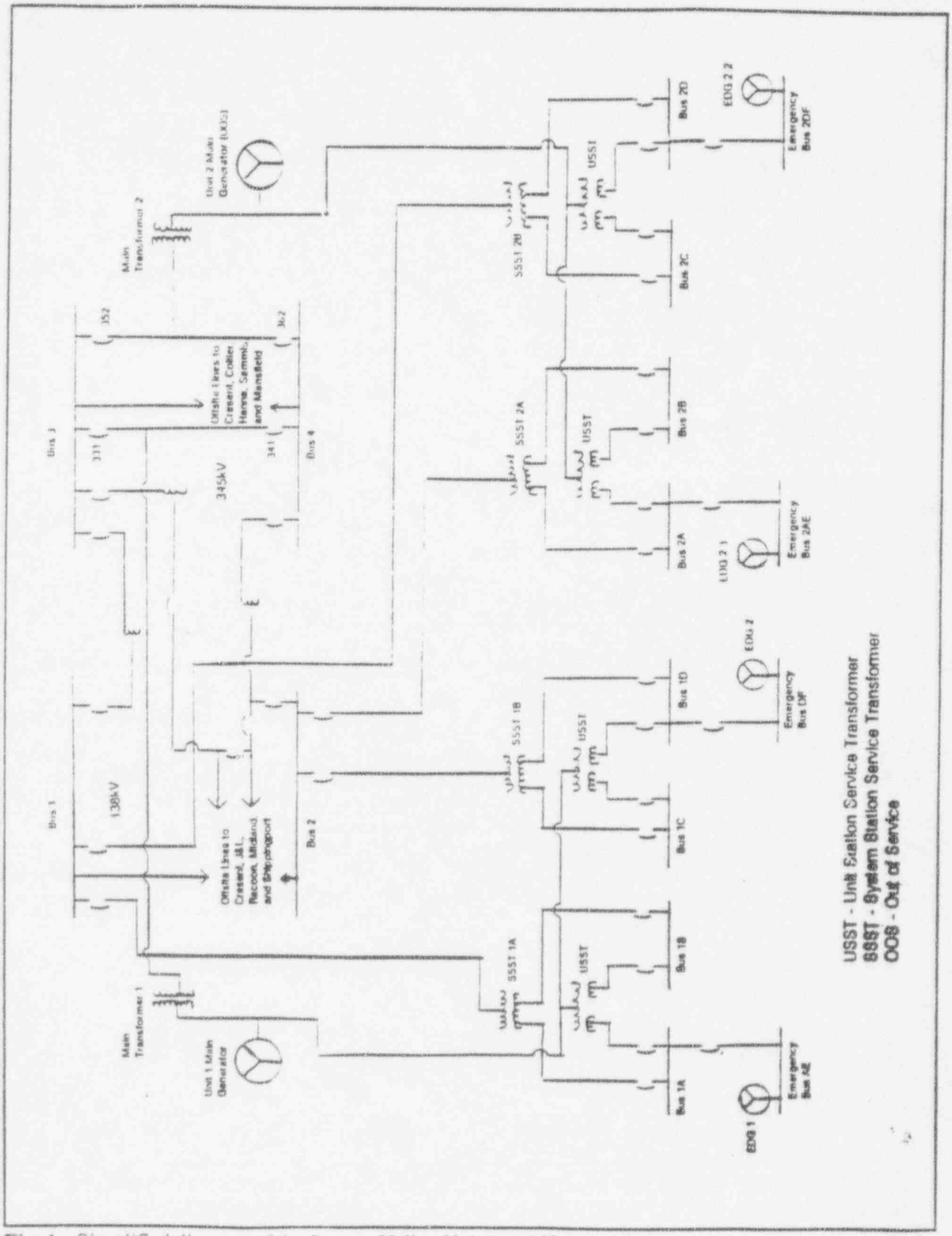

Fig. 1. Simplified diagram of the Beaver Valley Unit 1 and Unit 2 switchyard.

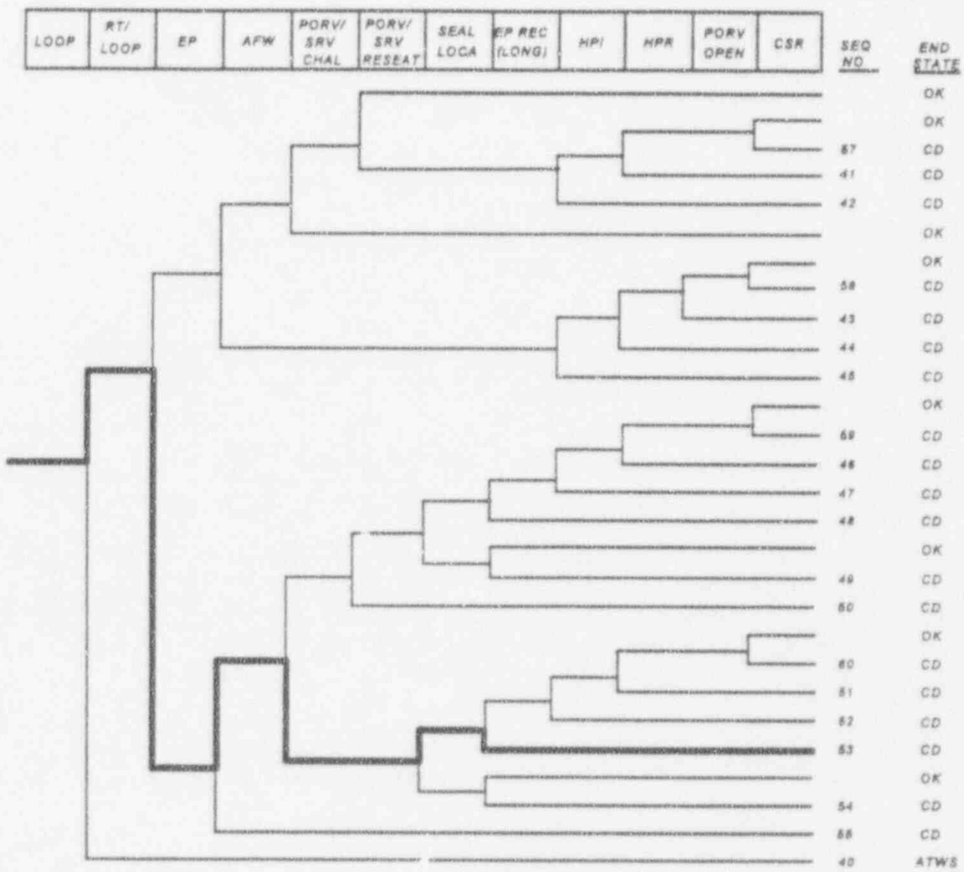

Fig. 2. Dominant core damage sequence for LER 334/93-013.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 334/93-013
Event Description: LOOP at Beaver Valley 1
Event Date: 10/12/93
Case:
Plant: Beaver Valley 1

INITIATING EVENT

NONRECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	3.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	6.2E-05
Total	6.2E-05
ATWS	
LOOP	0.0E+00
Total	0.0E+00

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
53	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall SEAL.LOCA EP.REC(SL)	CD	3.7E-05	2.4E-01
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05	8.2E-02
54	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall - SEAL.LOCA EP.REC	CD	9.3E-06	2.4E-01
48	LOOP -rt/loop emerg.power -afw/emerg.power porv.or.srv.chall - porv.or.srv.reseat/emerg.power SEAL.LOCA EP.REC(SL)	CD	1.5E-06	2.4E-01

** nonrecovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
48	LOOP -rt/loop emerg.power -afw/emerg.power porv.or.srv.chall - porv.or.srv.reseat/emerg.power SEAL.LOCA EP.REC(SL)	CD	1.5E-06	2.4E-01
53	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall SEAL.LOCA EP.REC(SL)	CD	3.7E-05	2.4E-01

Event Identifier: 334/93-013

54	LOOP -rt/loop emerg.power -afw/emerg.power -porv.or.srv.chall - SEAL.LOCA EP.REC	CD	9.3E-06	2.4E-01
55	LOOP -rt/loop emerg.power afw/emerg.power	CD	1.2E-05	8.2E-02

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwreseal.cmp
BRANCH MODEL: s:\asp\prog\models\beaver1.sl1
PROBABILITY FILE: s:\asp\prog\models\pwr_bsl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	3.3E-04	1.0E+00	
LOOP	1.6E-05 > 1.6E-05	3.6E-01 > 3.0E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05		
loca	2.4E-06	4.3E-01	
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	3.8E-04	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	2.0E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	3.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	3.0E-02	1.0E+00	
SEAL.LOCA	2.3E-01 > 1.5E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	2.3E-01 > 1.5E-01		
EP.REC(SL)	5.9E-01 > 3.8E-01	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	5.9E-01 > 3.8E-01		
EP.REC	6.1E-02 > 1.7E-02	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.1E-02 > 1.7E-02		
hpi	3.0E-04	8.4E-01	
hpi(f/b)	3.0E-04	8.4E-01	1.0E-02
porv.open	0.0E+00	1.0E+00	0.0E+00
hpr/-hpi	1.5E-04	1.0E+00	1.0E-03
csr	9.3E-05	1.0E+00	

* branch model file
** forced

Event Identifier: 334/93-013

ENCLOSURE 2

GUIDANCE FOR LICENSEE PEER REVIEW OF
PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event which occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include loss of off-site power (LOOP), Loss-of-Coolant Accident (LOCA), degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event. These sources are identified in the write-up documenting the analysis. The analysis methodology followed the process described in Section 2.1 and Appendix A of Volume 17 of NUREG/CR-4674, copies of which have been provided in this package for your use in this review.

Guidance for Peer Review and Criteria for Recovery Credit

The review of the preliminary analysis should use Section 2.1 and Appendix A of NUREG/CR-4674 for guidance. Comments regarding the analysis should address:

- Characterization of possible plant response,
- Representation of expected plant response used in the analytical models,
- Representation of plant safety equipment configuration and capabilities at the time of the event, and
- Assumptions regarding equipment recovery probabilities.

Any claims for credit for the use of additional systems, equipment, or specific actions in the recovery process must be supported by appropriate documentation in your response. The identified recovery measures must have existed at the time of the event, and should include:

- Normal or emergency operating procedures,
- Piping and instrumentation diagrams (P&IDs),
- Electrical one-line diagrams,
- Results of thermal-hydraulic analysis,
- Operator training (both procedures and simulator), etc.

Also, the documentation should address the impact of the use of the specific recovery measure on:

- The sequence of events,
- The timing of events,
- The probability of operator error in using the system or equipment, and
- Other systems/processes already modeled in the analysis.

For example, Plant A (a PWR) experiences a reactor trip and, during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW train modeling would be patterned after information gathered either from the plant PSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the standby feedwater system characteristics are documented in the FSAR, accounted for in the IPE, procedures for using the system during recovery existed at the time of the event, the plant operators had been trained in the use of the system prior to the event, a clear diagram (one-line diagram or better) of the system is available, previous analyses have indicated that there would be sufficient time available to implement the procedure successfully, and results of an assessment that evaluates the effect that use of the standby feedwater system has on already existing processes of procedures that would normally be used to deal with the event are available.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event:

- The specific licensee event report (LER), augmented inspection team AIT) report, or other pertinent reports as appropriate (separate enclosure).

- A calculation summary sheet indicating the dominant sequences and pertinent aspects of the modeling details (contained in the analysis writeup).

- An event tree with the dominant sequence(s) highlighted (contained in the analysis writeup).

- A copy of Section 2.1 and Appendix A of NUREG/CR-4674, Volume 17 (separate enclosures).

ENCLOSURE 3

Duquesne Light
Nuclear Group
P.O. Box 4
Shippingport, PA 15077-0004

Telephone (412) 393-6000

November 11, 1993
ND3MNO:3505

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, Licensee No. DPR-66
LER 93-013-00

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

In accordance with Appendix A, Beaver Valley Technical Specifications, the following Licensee Event Report is submitted:

LER 93-013-00, 10 CFR 50.73.a.2.i and 10CFR50.73.a.2.iv, "Unit 1 Reactor Trip and Required Shutdown, Dual Unit Loss of Offsite Power."

L. R. Freeland
General Manager
Nuclear Operations

JWM/ke

Attachment

150000

JE22 1/1

cc: Mr. T. T. Martin, Regional Administrator
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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Beaver Valley Power Station Unit 1	05000 3 3 4	1 OF 08

TITLE (4)
Unit 1 Reactor Trip and Required Shutdown, Dual Unit Loss of Offsite Power.

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Beaver Valley Unit 2	05000 412
10	12	9 3	9 3	- 0 1 3 -	0 0	11	11	93	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 1
POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
L. R. Freeland, General Manager Nuclear Operations	4 1 2 6 4 3 - 1 2 5 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	FK	XXX	XXX	N					
B	AB	XXX	XXX	N					

SUPPLEMENTAL REPORT EXPECTED (14)

				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 10/12/93 Unit 1 was operating at 100 percent power and Unit 2 was in a refueling outage with all fuel removed from the reactor vessel. At 1507 hours, Unit 1 experienced a large loss of offsite load when ten offsite feed breakers in the Beaver Valley switchyard opened as a result of an inadvertent underfrequency system separation actuation. The load reduction caused the Unit 1 turbine to trip on mechanical overspeed and resulted in a High Flux Rate Reactor Trip. The opening of the switchyard feed breakers and the resultant Unit 1 generator trip resulted in a loss of offsite power to Units 1 and 2. Both Unit 1 Emergency Diesel Generators (EDGs), and the required Unit 2 EDG, started and supplied their required loads. Unit 1 Auxiliary Feedwater actuated due to Low-Low Steam Generator Levels resulting from the Reactor Trip. Unit 1 was stabilized using the Emergency Operating Procedures. Following realignment of switchyard breakers, offsite power was restored to both units by 1522 hours. On 10/13/93, following a Unit 1 containment inspection, a Reactor Coolant System Pressure Boundary Leak was discovered on the Loop 1A cold leg vent valve RC-27. A Technical Specification Required cooldown was initiated and Mode 5 was entered at 0304 hours on 10/14/93.

DESCRIPTION OF EVENT

On October 12, 1993 Beaver Valley Unit 1 was operating at 100 percent power with normal station loads being supplied from the unit station service transformers. Unit 2 was in the Fourth Refueling Outage with all of the fuel removed from the reactor vessel and stored in the spent fuel pool. Required Unit 2 electrical loads were being supplied from offsite power via backfeed through the main unit transformer. Power was also available to Unit 2 via the 2A system station service transformer. The 2B system station service transformer was removed from service for maintenance.

At 1507 hours, Unit 1 experienced a loss of the majority of its electrical load when ten offsite feed breakers in the Beaver Valley switchyard opened unexpectedly. The loss of these offsite breakers, which included the in-service Beaver Valley Unit 2 main output breaker (PCB 362) and one Unit 1 output breaker (PCB 341), caused Unit 1 generator load to drop from approximately 810 net MWe to 85 net MWe. The loss of load caused the turbine speed to increase until the turbine tripped on mechanical overspeed (setpoint 1998 rpm). The Turbine Overspeed Protection (OPC) trip actuation operated but was not required since the turbine had already tripped on mechanical overspeed. Historical computer data from the event indicated turbine peak speed at 2051 rpm. The increased turbine speed caused an increase in generator output frequency forcing a corresponding increase in the Reactor Coolant Pump (RCP) speed. A transient Reactor Coolant System flow increase resulted from the RCP speed change. This flow transient translated into a positive reactivity change leading to a High Flux Rate Reactor Trip. All Control Rods inserted fully.

Following the Unit 1 Reactor Trip, the No. 1 Emergency Diesel Generator (EDG) auto-started, due to Train A Emergency 4KV bus (AE) undervoltage; however, the undervoltage condition was not sufficient to require the AE bus to shed its loads and cause EDG sequencing. All three Auxiliary Feedwater (AFW) Pumps (two motor driven and one steam driven) auto-started due to the shrink in steam generator levels. All three Reactor Coolant Pumps tripped on bus underfrequency as the Main Unit Generator speed reduced. Thirty seconds following the turbine trip, the generator output breakers opened as designed. The Unit 1 Main Unit Generator had been the only normal power source for Unit 1 and Unit 2 electrical loads since the underfrequency separation scheme actuated. When the Unit 1 generator tripped, Unit 1 and 2 both experienced a loss of offsite power.

Following the loss of offsite power, Unit 1 normal 4KV busses de-energized and shed their loads, and the Unit 1 No. 2 EDG started. Both Unit 1 EDGs then properly sequenced loads on their respective busses as designed, including charging, river water, component cooling, and AFW pumps. Unit 1 operators stabilized the plant using the Emergency Operating Procedures (EOPs). Initially, a natural circulation cooldown was established as no power was available for the Reactor Coolant Pumps. The Main Steamline Isolation Valves were closed manually, in accordance with Emergency Operating Procedure E-0, as there was no position indication available for the Reheater Steam Supply Isolation Valves during the loss of offsite power. Operators then utilized Steam Generator Atmospheric Steam Release Valves to remove decay heat and control the cooldown. At 1517 hours, the Duquesne Light Company System Operations Department restored offsite power by re-closing the switchyard breakers. The Unit 1 control room crew then established forced Reactor Coolant System cooling by starting Reactor Coolant Pump 1C. The AE and DF emergency busses were realigned to offsite power and the EDGs were secured.

At the initiation of the event at Unit 2 (prior to the loss of offsite power) the standby Primary Component Cooling Water Pump (2CCP-P21C) auto-started on low header pressure, the Unit 2, 2-1 Emergency Diesel Generator (EDG) started on degraded bus voltage, and the 2A and 2B normal 4KV busses transferred to offsite power. The dual unit Control Room Emergency Pressurization System actuated due to a loss of voltage to the Control Room Area Radiation Monitor 2RMC-RQ201. Following the Unit 1 main unit generator trip and the resultant loss of offsite power, the Unit 2, Train A emergency 4KV bus (2AE) shed its loads and the Unit 2, 2-1 EDG properly sequenced all available loads. Low Head Safety Injection Pump 2SIS-P21A auto-started via the EDG sequencer as designed, but no water was injected since the discharge valves were closed for refueling. The pump was secured eighty-four seconds after it started. The Unit 2 Train B emergency 4KV bus (2DF) and associated 2-2 EDG had been removed from service for outage related maintenance and were not required to be operable. Following restoration of offsite power at Unit 2 (1522 hours), the 4KV system was reenergized and the Train A normal to emergency 4KV tie breakers were closed. The Unit 2, 2-1 EDG was unloaded and output breaker opened at 1535 hours.

Following the Reactor Trip, Unit 1 was in Hot Standby, Mode 3. At 0345 hours, on October 13, 1993, a Unit 1 containment entry was made to perform routine, post trip, leak inspections. During this inspection, a leak was identified at the Loop 1A Cold Leg Vent Valve (RC-27). This valve is also used as a connection point for disc pressurization for isolating the 1A reactor coolant loop. A subsequent entry was made to perform more detailed inspections. A review of photographs and discussion by Mechanical Maintenance and Operations, led to the conclusion that potential Pressure Boundary Leakage existed.

Unit 1 then commenced a cooldown to Cold Shutdown per Technical Specification 3.4.6.2.a, and declared an Unusual Event per the Emergency Preparedness Plan. Unit 1 entered Mode 5 at 0304 hours on October 14, 1993 and the Unusual Event was terminated at that time. Upon inspection, RC-27 was found to have a through-wall crack at the fillet weld, verifying Pressure Boundary Leakage.

CAUSE OF EVENT

The cause of the loss of offsite power event was personnel error. A three man Electrical Maintenance crew, consisting of a Crew Leader, an Electrical Maintenance Technician, and a Senior Engineer, were performing scheduled outage maintenance on Unit 2 Main Output Breaker PCB 352. During the verification of auxiliary contact alignment of the PCB 352 breaker, an inadvertent application of 125 Volt DC actuated an underfrequency separation scheme in the Beaver Valley switchyard. This resulted in the opening of seven 345 KV feed breakers (including Unit 1 Main Unit Output Breaker PCB 341) and three 138 KV feed breakers, initiating the loss of electrical load at Unit 1.

A cracked mechanical linkage, for the center stack auxiliary contacts of breaker PCB 352, was replaced the morning of October 12, 1993. At 1400 hours, during timing tests of the breaker's mechanism, the Beaver Valley Relay Group Supervisor notified the maintenance crew that reset relays associated with PCB 352, located in the Unit 2 Relay Room, were overheating. It was determined that the auxiliary contacts, located in the center stack of a three stack assembly, were in the wrong position. This caused the operate and reset coils of the reset relays in the relay room to be energized simultaneously, resulting in overheating. The maintenance crew then visually checked the auxiliary contacts of PCB 352 on the stack where the cracked arm was replaced. They determined that the stack's shaft was rotated out of position. The problem was corrected and the auxiliary contact linkage reassembled. Using a multimeter on continuity scale and site electrical prints, the crew then started checking the three auxiliary contacts connected to this linkage for other possible misalignment problems. During this verification, underfrequency tripping relays were actuated when 125 Volt DC from one set of contacts was inadvertently connected to another set of contacts in the underfrequency separation scheme, via the multimeter.

The cause of the Unit 1 Pressure Boundary Leak was determined to be due to a fillet weld failure. Samples of pipe removed from RC-27 were sent to a laboratory for failure analysis. The results indicated that the weld failed due to the presence of an imbedded flaw that propagated inward and outward, causing a through-wall crack. RC-27 was inspected during the last refueling outage (9R) in response to a vendor recommendation concerning disc pressurization line socket weld cracking. A linear indication was found at that time and was believed to have been satisfactorily repaired. A minor design change was also implemented in 9R to reduce the pipe length, thereby reducing the probability of pipe failure due to cyclic loads.

CORRECTIVE ACTIONS

The following corrective actions have been initiated as a result of the event:

1. Detailed root cause analyses were performed to determine the cause of the switchyard transient and Reactor Coolant System leak.

2. Interim administrative controls over work performed in the Beaver Valley switchyard were issued that require Operations Department approval of all work activities in the switchyard.

3. Long term administrative controls governing work in the switchyard will be established by the managers responsible for switchyard activities.

4. The Underfrequency System Separation scheme in the Beaver Valley switchyard has been disabled. At the time the separation scheme was implemented, there was sufficient electrical load available in the local vicinity to maintain Beaver Valley Unit 1 on-line and separated from the rest of the system. As a result of load changes, this separation scheme is no longer valid.

5. Unit 1 Loop 1A Cold Leg Vent Valve (disc pressurization connection) RC-27 was removed, plugged, capped, and welded. All other disc pressurization taps penetrating loop stop valves were inspected at both Beaver Valley units and found to be satisfactory. Samples removed from RC-27 indicate that the failure was due to an imbedded flaw. Further evaluation will be performed to determine the need for additional corrective actions.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

REPORTABILITY

Beaver Valley Units 1 and 2 reported the Reactor Trip and Dual Unit Loss of Offsite Power to the Nuclear Regulatory Commission, via the Emergency Notification System, at 1843 hrs on October 12, 1993, and Unit 1 reported the Unusual Event at 0811 hours on October 13, 1993. The Unit 1 Reactor Trip and Dual Unit Loss of Offsite Power were reported in accordance with 10 CFR 50.72.b.2.ii. (Reactor Protection System and Engineered Safety Feature Actuations) and the Unit 1 Unusual Event was reported in accordance with the Emergency Preparedness Plan and 10 CFR 50.72.b.1.i.A. (Technical Specification Required Shutdown). This written report is being submitted in accordance with 10 CFR 50.73.a.2.iv. and 10 CFR 50.73.a.2.i.

SAFETY IMPLICATIONS

There were minimal safety implications at Units 1 or 2 as a result of this event. At Unit 1 the Reactor Protection System functioned as designed and actuated a reactor trip. The operating crew successfully stabilized the plant following the reactor trip using the Emergency Operating Procedures. Normal post-trip evaluations were performed and all ESF equipment was determined to have functioned as designed. The event is bounded by the following Updated Final Safety Analysis (UFSAR) Sections and plant response was deemed to be within the analysis results and conclusions: 14.1.7 (Loss of External Electrical Load and/or Turbine Trip), 14.1.8 (Loss of Normal Feedwater), 14.1.11 (Loss of Offsite Power to the Station Auxiliaries (Station Blackout)), 14.1.12 (Turbine - Generator Accidents), and 14.2.9 (Complete Loss of Forced Coolant Flow).

Unit 2 was in a Refueling Outage with all of the fuel removed from the reactor vessel and stored in the spent fuel pool. The 2-2 Emergency Diesel Generator (EDG) and the Train B emergency 4KV bus were on clearance. On the loss of off-site power all required Train A station loads were properly sequenced by the 2-1 EDG. At Unit 2 the event was bounded by UFSAR Section 15.2 6 (Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)).

There were also minimal safety implications to the public as a result of the Reactor Coolant Pressure Boundary leakage. All leakage was contained inside the Containment Building. Recent Reactor Coolant System Water Inventory Balance Tests, prior to the event, had shown unidentified leakage at less than 0.1 gpm. This event was bounded by Unit 1 UFSAR Section 14.3.1 (Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates Emergency Core Cooling System). Emergency Core Cooling was not actuated for this event.

DIESEL GENERATOR RELIABILITY

Both Unit 1 emergency diesel generators and the operable Unit 2 emergency diesel generator, properly started and sequenced all available loads at the proper times as designed for a loss of offsite power. The following is a summary of the past 20, 50 and 100 start and load demands for Unit 1 and 2 emergency diesel generators, trended in accordance with NUMARC 87-00 Rev. 1, Appendix D (Data as of September 30, 1993):

$$\text{Reliability} = 1 - \frac{\text{Number of Valid Failures}}{\text{Number of Valid Demands}}$$

Unit 1

Past 20 Start Demands: 1 = 1 - 0/20
Past 50 Start Demands: 1 = 1 - 0/50
Past 100 Start Demands: 1 = 1 - 0/100

Past 20 Load Demands: 1 = 1 - 0/20
Past 50 Load Demands: 1 = 1 - 0/50
Past 100 Load Demands: 0.99 = 1 - 1/100

Unit 2

Past 20 Start Demands: 1 = 1 - 0/20
Past 50 Start Demands: 1 = 1 - 0/50
Past 100 Start Demands: 1 = 1 - 0/100

Past 20 Load Demands: 1 = 1 - 0/20
Past 50 Load Demands: 1 = 1 - 0/50
Past 100 Load Demands: 1 = 1 - 0/100

Note: Subsequent to this summary, Unit 2 experienced relay failures on both diesel generators, which are not listed above, but would have prevented sequencer loading on a safety injection signal. These will be reported in a subsequent Unit 2 Licensee Event Report on the diesel generator failures.

PREVIOUS SIMILAR EVENTS

No similar events have previously occurred at Beaver Valley Units 1 and 2 involving a reactor trip and loss of offsite power.

Unit 1 has previously reported two events involving a required plant shutdown due to Reactor Coolant System (RCS) Pressure Boundary leakage:

1. LER 1-88-016 "Unit Shutdown Due to Pressure Boundary Leakage." This event involved a failed weld on the line near an RCS seal injection drain valve.

2. LER 1-91-002 "Reactor Coolant System Pressure Boundary Leakage Results in Plant Shutdown." This event involved the failure of a socket weld on the Loop 1B Cold Leg Vent Valve (disc pressurization connection).

ENCLOSURE 4

2.0 ACCIDENT SEQUENCE PRECURSOR IDENTIFICATION AND QUANTIFICATION

2.1 Accident Sequence Precursor Identification

The ASP Program is concerned with the identification and documentation of operational events that have involved portions of core damage sequences, and with the estimation of frequencies and probabilities associated with them.

Identification of precursors requires the review of operational events for instances in which plant functions that provide protection against core damage have been challenged or compromised. For core damage to occur, fuel temperature must increase. Such an increase requires the heat generation rate in the core to exceed the heat removal rate. This can result from either a loss of core cooling or excessive core power. The following functions are provided at all plants to protect against these two conditions:

- Reactor subcriticality. The reactor must be placed in a subcritical condition, normally by inserting control rods into the core to terminate the chain reaction.
- Reactor coolant inventory makeup. Sufficient water must be provided to the reactor coolant system (RCS) to prevent core uncovery.
- RCS integrity. Loss of RCS integrity requires the addition of a significant quantity of water to prevent core uncovery.
- Decay heat removal (DHR). Heat generated in the core by fission product decay must be removed.
- Containment integrity. Containment integrity (containment heat removal, isolation, and hydrogen control) is not addressed in the precursor analyses unless core DHR capability is impacted.

System-based event trees were developed to model potential sequences to core damage. The event trees are specific to eight plant classes so as to reflect differences in design among plants in the U.S. LWR population. Three initiators are addressed in the event trees: trip [which includes loss of main feedwater (LOFW) within its sequences], loss of offsite power (LOOP), and small-break loss-of-coolant accident (LOCA). These three initiators are primarily associated with loss of core cooling. [Excessive core power associated with anticipated transient without scram (ATWS) is represented by a failure-to-trip sequence but is not developed.] Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with these initiators. Detailed descriptions of the plant classification scheme and the event tree models are included in Appendix A. Operational events that cannot be associated with one of these initiators are accommodated by unique modeling.

Armed with a knowledge of the primary core damage initiator types plus the systems that provide protection against core damage (based on the event tree models), ASP Program staff members examine LERs to determine the impact of operational events on potential core damage sequences. While the sequences detailed on the event tree models do not describe all possible paths to core damage, they form a primary basis for selecting an operational event as a precursor. Operational events are also reviewed in a more general sense for their impact on the protective functions described above.

Identification of precursors within a set of LERs involved a two-step process. First, each LER was reviewed by two experienced engineers to determine if the reported event should be examined in detail. This initial review was a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that were clearly unimportant. This was done by eliminating events that satisfied pre-defined criteria for rejection and accepting all others as potentially significant and requiring analysis. In some cases, events are impractical to analyze due to lack of information or inability to reasonably model within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models. Events also were eliminated from further review if they had little impact on core damage sequences or provided little new information on the risk impacts of plant operation. Such events included single failures in redundant systems and uncomplicated reactor trips and LOFWs. Any event with an impact that can be mapped onto the ASP core damage models can, in principle, be assessed.

LERs were eliminated from further consideration as precursors if they involved at most one of the following:

- a component failure with no loss of redundancy,
- a loss of redundancy in only one system,
- a seismic design or qualification error,
- an environmental design or qualification error,
- a structural degradation,
- an event that occurred prior to initial criticality (since the core is not considered vulnerable to core damage at this time and since distinguishing initial testing failures from operational failures is difficult),
- a design error discovered by reanalysis,
- an event impact bounded by a reactor trip or LOFW,
- an event with no appreciable impact on safety systems, or
- an event involving only post-core damage impacts (selected containment-related events are documented).

Events identified for further consideration typically included

- unexpected core damage initiators (LOOP and small-break LOCA);
- all events in which reactor trip was demanded and a safety-related component failed;
- all support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- any event where two or more failures occurred;
- any event or operating condition that was not predicted or that proceeded differently from the plant design basis; and
- any event that, based on the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Operational events that were not eliminated in the first review received a more extensive analysis to identify those events considered to be precursors to potential severe core damage accidents either because of an initiating event or because of failures that could have affected the course of postulated off-normal events or accidents. These detailed reviews were not limited to the LERs; they also used final safety analysis reports (FSARs), their amendments, and other information available at the Nuclear Operations Analysis Center.

The detailed review of each event considered (l) the immediate impact of an initiating event or (2) the potential impact of the equipment failures or operator errors on readiness of systems in the plant for mitigation of off-normal and accident conditions.

In the review of each selected event, three general scenarios (involving both the actual event and postulated additional failures) were considered:

1. If the event or failure was immediately detectable and occurred while the plant was at power, then the event was evaluated according to the likelihood that it and the ensuing plant response could lead to severe core damage.
2. If the event or failure had no immediate effect on plant operation (i.e., if no initiating event occurred), then the review considered whether the plant would require the failed items for mitigation of potential severe core damage sequences should a postulated initiating event occur during the failure period.
3. If the event or failure occurred while the plant was not at power, then the event was first evaluated according to whether it could have occurred while at power or at hot shutdown immediately following power operation. If the event could only occur at cold shutdown, then its impact on continued DHR was assessed.

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER, the sequence of operation of various mitigating systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if they included one of the following attributes that impacted core damage sequences and if the conditional probability of subsequent core damage (described later) was at least 1.0×10^{-6}

- an unexpected core damage initiator (such as a LOOP, steam-line break (SLB), or small-break LOCA);
- a failure of a system (all trains of a multiple train system) required to mitigate the consequences of a core damage initiator,
- concurrent degradation in more than one system required to mitigate the consequences of a core damage initiator, or
- a transient or LOFW with a degraded mitigating system.

Events of low significance are thus excluded, allowing the reader to concentrate on the more important events. This approach is consistent with the approach used to define 1987-1991 precursors, but is different from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria, regardless of conditional core damage probability.

Events that occurred in 1992 were reviewed for precursors only if they satisfied an initial significance screening. This approach, which was similar to that used in the review of 1988-1991 events, eliminated many insignificant events from review and permitted some increase in the amount of documentation provided for precursors. Two approaches were used to select events to be reviewed for precursors.

First, events were reviewed for precursors if they were identified as significant by the Nuclear Regulatory Commission's (NRCs) Office for Analysis and Evaluation of Operational Data (AEOD). AEOD's screening process identifies operating occurrences involving, in part,

- violation of a safety limit;
- an alert or higher emergency classification;
- an on-demand failure of a safety system (except surveillance failures);
- events involving unexpected system or component performance with serious safety significance or generic implications;
- events where improper operation, maintenance, or design causes a common-mode/common-cause failure of a safety system or component, with safety significance or generic implications;
- safety-significant system interactions;
- events involving cognitive human errors with safety significance or generic implications;
- safety-significant events involving earthquakes, tornadoes, floods, and fires;
- a scram, transient, or engineered safety features (ESF) actuation with failure or inoperability of required equipment;
- on-site work-related or nuclear-incident-related death, serious injury, or exposure that exceeds administrative limits;
- unplanned or unmonitored releases of radioactivity, or planned releases that exceed Technical Specification limits; and
- infrequent or moderate frequency events.

AEOD-designated significant events also involve operating conditions, where a failure or accident has not occurred but where the potential for such an event is identified.

Second, LERs were also reviewed if they were identified through a computerized search using the sequence coding and search system (SCSS) data base of LERs. This computerized search identified LERs potentially involving (1) failures in plant systems that provided the protective functions described earlier and (2) initiating events addressed in the ASP models. Based on a review of the 1984-87 precursor evaluations, this computerized search successfully identifies almost all precursors within a subset of approximately one-third of all LERs.

While review of LERs identified by AEOD and through the use of SCSS is expected to identify almost all precursors, it is possible that a few precursors exist within the set of unreviewed LERs. Some potential precursors that would have been found if all 1992 LERs had been reviewed may not have been identified. Because of this (plus modeling changes that impact precursor probability somewhat), it should not be assumed that the set of 1988-92 precursors is consistent with precursors identified in 1984-87.

Following AEOD and SCSS computerized screening, 1022 LERs from 1992 were reviewed for precursors. Twenty-seven operational events with conditional probabilities of subsequent severe core damage greater than 1.0×10^{-6} were identified as accident sequence precursors.

Individual failures of boiling-water reactor (BWR) high-pressure coolant injection (HPCI), high-pressure core spray (HPCS), and reactor core isolation cooling (RCIC) systems (all single-train systems), and trips and LOFWs without additional mitigating system failures were not selected as precursors. The impact of such events was determined on a plant-class basis. The results of these evaluations are provided in Appendix A.

In addition to accident sequence precursors, events involving loss of containment functions — containment cooling, containment spray, containment isolation (direct paths to the environment only), and hydrogen control — were identified in the review of 1992 LERs. Other events that were not selected as precursors but that provided insight into unusual failure modes with the potential to compromise continued core cooling are also identified. Events identified as precursors are documented in Appendix B, the containment-related events are documented in Appendix C, events considered "interesting" are documented in Appendix D, and events that were determined to be impractical to analyze are documented in Appendix E.

2.2 Estimation of Precursor Significance

Quantification of ASP significance involves determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto the ASP event trees, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system failure given the operational event.

In the precursor quantification process, it is assumed that the failure probabilities for systems observed to have failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Failure probabilities for systems observed to have been degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. The failure probabilities associated with observed successes and with systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities, with consideration of the potential for recovery. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage that remains once the observed failures have occurred.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the LWR population, even though they are applied to sequences that are plant-class specific in nature. Because of this, the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred.

The evaluation of precursor events in this report consider and, where appropriate, give credit for additional equipment or recovery procedures the plants have recently added. Accordingly, the evaluations this year may not be directly comparable to the results of prior years. Examples of additional equipment and recovery procedures addressed in the 1992 analyses, when information was available, include use of supplemental diesel generators (DGs) for station blackout mitigation, alternate systems for steam generator (SG) and RCS makeup, and depressurization of the primary with low pressure injection (LPI) in lieu of high pressure injection (HPI).

The ASP calculational process is described in detail in Appendix A. This appendix documents the event trees used in the 1988-1992 precursor analyses, changes to these trees from prior years, the approach used to estimate event tree branch and sequence probabilities, and sample calculations; it also provides probability values used in the calculations. The overall precursor selection process is illustrated in Fig. 1.

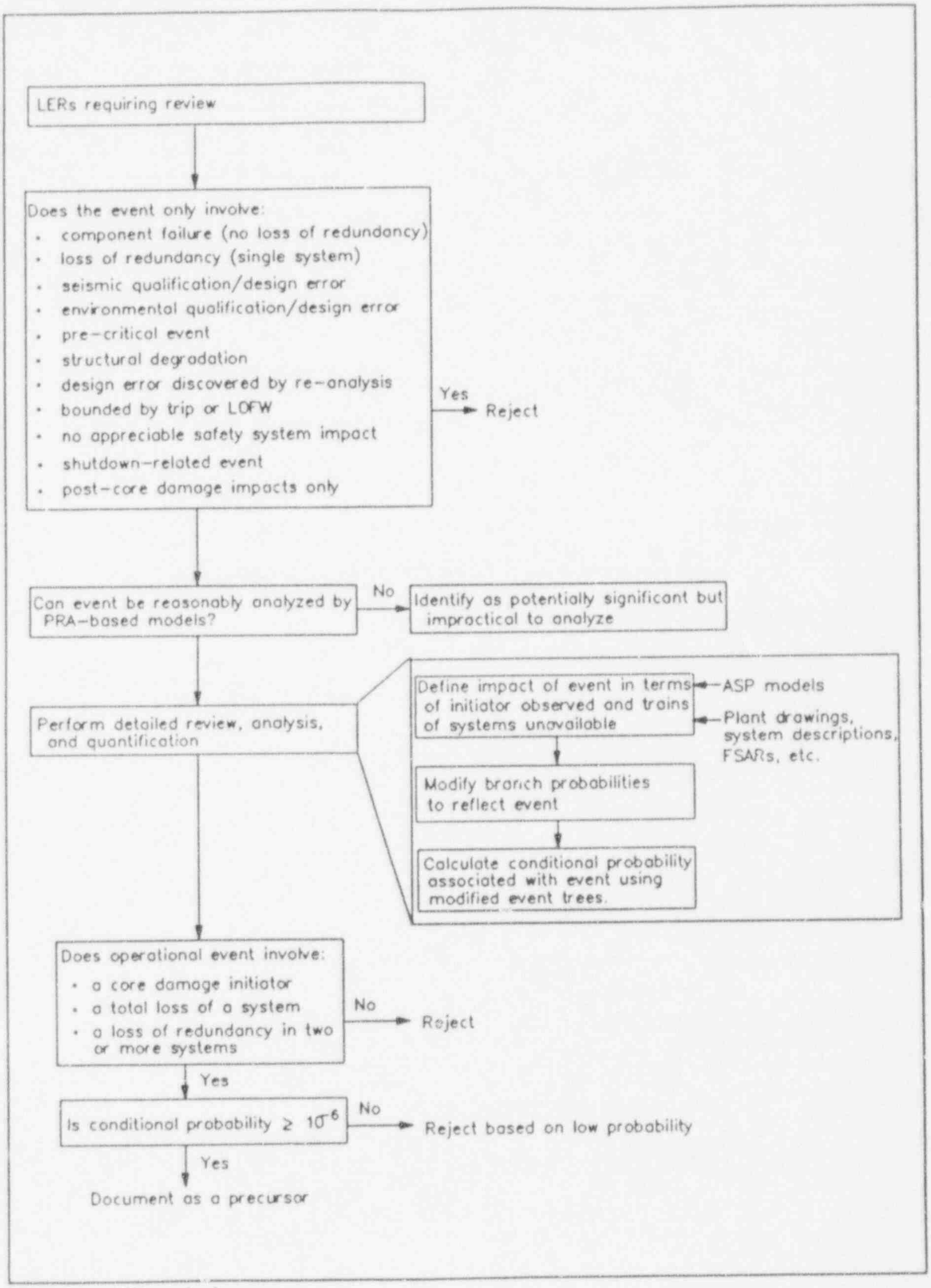


Fig. 1. ASP analysis process.

2.3 Documentation of Events Selected as Accident Sequence Precursors

Each 1992 precursor is documented in Appendix B. A description of the operational event is provided along with additional information relevant to the assessment of the event, the ASP modeling assumptions and approach used in the analysis, and analysis results. Two figures are also provided that (1) visually describe the dominant core damage sequence postulated for the event and (2) present a graph of the relative significance of the event compared with other potential events at the plant. The other potential events at the same plant are briefly described below:

<u>PWR & BWR</u>

Trip	• Trip with equipment operable.
LOOP	• Loss of offsite power. Includes plant-centered, grid-centered, severe weather and extreme severe weather-related initiators.
360h EP	• 360 h without emergency power sources (normally on-site emergency diesel generators).

<u>PWR</u>

LOFW + 1MTR AFW	• Transient with loss of main feedwater and one motor driven AFW (or EFW pump failed (turbine driven pump substituted if plant does not have any motor driven pumps).
360h w/o AFW	• 360 hours with all AFW (or EFW) pumps failed.

<u>BWR</u>

360 h w/o HPCI and RCIC	• 360 hours with HPCI and RCIC failed (not applicable for Type A BWRs).
LOFW and HPCI	• Transient with loss of main feedwater and HPCI (loss of main FW and loss of Isolation Condensor is run instead for Type A BWRs).

An additional item, the conditional core damage calculation, documents the calculations performed to estimate the conditional core damage probability associated with the precursor and includes probability summaries for end states, the conditional probability for the more important sequences, and the branch probabilities used. Copies of the LERs and AIT Reports relevant to the event are also provided in Appendix F, listed in docket number order.

Appendices C, D and E include similar documentation for other events selected in the ASP Program (containment-related, other, and impractical events). No probabilistic analysis was performed on these events.

2.4 Tabulation of Selected Events

The 1992 events selected as precursors are listed in Table 1. The precursors have been arranged in numerical order by event identifier and the following information is included.

1. docket/LER number associated with the event (Event Identifier);
2. name of plant where the event occurred (Plant);
3. a brief description of the event (Description);
4. date of the event (Event Date);
5. conditional probability of potential severe core damage associated with the event (C_D Probability);
6. initiator associated with the event or unavailability if no initiator was involved (TRANS).
7. abbreviations for the primary system and component involved in the event (System, Component);
8. plant operating status at the time of the event (O);
9. discovery method associated with the event (operational or testing) (D);
10. whether the event involved human error (E);
11. plant power rating, type, vendor, architect-engineer, and licensee (MWE, T, V, AE, Operator);

The information in Table 1 has been sorted in several ways to provide additional perspectives.

	Sorted by
Table 2	Plant name and LER number
Table 3	Event date
Table 4	Initiator or unavailability
Table 5	System
Table 6	Component
Table 7	Plant operating status
Table 8	Discovery method
Table 9	Conditional core damage probability
Table 10	Plant type and vendor

Abbreviations used in Tables 1—10 are defined in Tables 11a—11f.

2.5 Potentially Significant Events That Could Not Be Analyzed

A number of LERs identified as potentially significant were considered impractical to analyze. Examples of such events include component degradations where the extent of degradation could not be determined (for example, biological fouling of room coolers) or where a realistic estimate of plant response could not be made (for example, high energy line break concerns). Other events of this type include cable routing not in accordance with Appendix R requirements for fire protection, and inoperability of flood barriers. For both of these situations, detailed plant design information, and preferably an existing fire or flood PRA analysis, are required to reasonably estimate the significance of the event.

For many events classified as impractical to analyze, an assumption that the impacted component or function was unavailable over a 1-year period (as would be done using a bounding analysis) would result in a conclusion that a very significant condition existed. This conclusion was not supported by the specifics of the event as reported in the LER or by the limited engineering evaluation performed in the ASP Program. A reasonable estimate of significance for such events requires far more analysis resources than can be applied in the ASP Program.

Brief descriptions of events considered impractical to analyze are provided in Appendix E.

2.6 Potential Sources of Error

As with any analytic procedure, the availability of information and modeling assumptions can bias results. In this section, several of these potential sources of error are addressed.

1. *Evaluation of only a subset of 1992 LERs.* For 1969-81 and 1984-87, all LERs reported during the year were evaluated for precursors. For 1988-92, only a subset of LERs were evaluated in the ASP Program following a computerized search of the SCSS data base and screening by NRC personnel. While this subset is believed to include most serious operational events, it is possible that some events that would normally be selected as precursors were missed because they were not included in the subset that was screened.

2. *Inherent biases in the selection process.* Although the criteria for identification of an operational event as a precursor are fairly well defined, the selection of an LER for initial review can be somewhat judgmental. Events selected in the study were more serious than most, so the majority of the LERs selected for detailed review would probably have been selected by other reviewers with experience in LWR systems and their operation. However, some differences would be expected to exist; thus, the selected set of precursors should not be considered unique.

3. *Lack of appropriate information in the LER.* The accuracy and completeness of the LERs in reflecting pertinent operational information is questionable in some cases. Requirements associated with LER reporting (i.e., 10 CFR 50.73), plus the approach to event reporting practiced at particular plants, can result in variation in the extent of events reported and report details among plants. Although the LER rule of 1984 has reduced the variation in reported details, some variation still exists. In addition, only details of the sequence (or partial sequences for failures discovered during testing) that actually occurred are usually provided; details concerning potential alternate sequences of interest in this study must often be inferred.

4. *Accuracy of the ASP models and probability data.* The event trees used in the analysis are plant-class specific and reflect differences between plants in the eight plant classes that have been defined. While major differences between plants are represented in this way, the plant models utilized in the analysis may not adequately reflect all important differences. Known problems concern the representation of HPI for some pressurized-water reactors (PWRs), long-term DHR for BWRs, and ac power recovery following a LOOP and battery depletion (station blackout issues). Modeling improvements that address these problems are being pursued in the ASP Program.

   Because of the sparseness of system failure events, data from many plants must be combined to estimate the failure probability of a multitrain system or the frequency of low- and moderate-frequency events (such as LOOPs and small-break LOCAs). Because of this, the modeled response for each event will tend toward an average response for the plant class. If systems at the plant at which the event occurred are better or worse than average (this is difficult to ascertain without extensive operating experience), the actual conditional probability for an event could be higher or lower than that calculated in the analysis.

   Known plant-specific equipment and procedures that can provide additional protection against core damage beyond the plant-class features included in the ASP event tree models were addressed in the 1992 precursor analysis. This information was not uniformly available — much of it was provided in licensee comments on preliminary analyses and in Individual Plant Examination (IPE) documentation available at the time this report was prepared. As a result, consideration of additional features may not be consistent in precursor analyses of events at different plants. However, analyses of multiple events that occurred at an individual plant or at similar units at the same site were uniformly developed.

5. *Difficulty in determining the potential for recovery of failed equipment.* Assignment of recovery credit for an event can have a significant impact on the assessment of the event. The approach used to assign recovery credit is described in detail in Appendix A. The actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values currently used in the ASP analyses. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.

   Programmatic constraints have prevented substantial efforts in estimating actual recovery class distributions. The values currently used are based on a review of recovery actions during historic events and also include consideration of human error during recovery. These values have been reviewed both within and outside the ASP Program. While it is acknowledged that substantial uncertainty exists in them, they are believed adequate for ranking purposes, which is the primary goal of the current precursor calculations. This assessment is supported by the sensitivity and uncertainty calculations documented in the 1980-81 report.[1] These calculations demonstrated only a small impact on the relative ranking of events from changes in the numeric values used for each recovery class.

6. *Assumption of a 1-month test interval.* The core damage probability for precursors involving unavailabilities is calculated on the basis of the exposure time associated with the event. For failures discovered during testing, the time period is related to the test interval. A test interval of 1 month was assumed unless another interval was specified in the LER.

   If the test interval is longer than this, on the average, for a particular system, then the calculated probability will be lower than that calculated using the actual test interval. Examples of longer test intervals would be situations in which (1) system valves are operated monthly but a system pump is started only quarterly or (2) valves are partially stroked monthly but fully operated only during refueling. Conversely, more frequent testing will result in a higher calculated failure probability than that calculated using the actual, shorter test interval. Test interval assumptions can also impact system failure probabilities estimated from precursor events, as described in Ref. 1.

2.7 Reference

1. W. B. Cottrell, J. W. Minarick, P. N. Austin, E. W. Hagen, and J. D. Harris, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents: 1980-81, A Status Report,* USNRC Report NUREG/CR-3591, Vols. 1 and 2 (ORNL/NSIC-217/V1 and V2), July 1984.*

*Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

ENCLOSURE 5

APPENDIX A. ASP MODELS

A. ASP MODELS

This appendix provides information concerning the methods and models used to estimate event significance in the ASP Program. The basic models used in the analysis of 1992 precursors are the same as those used for 1989-91 precursors. However, the analysis of 1992 precursors considered the potential use of alternate equipment and procedures, beyond that addressed in the basic models, that recently have been added by the licensees to provide additional protection against core damage, if information regarding this equipment was available. This equipment is described in Sect. A.3.

A.1 Precursor Significance Estimation

Quantification of accident sequence precursor significance involves determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto event trees depicting potential paths to severe core damage and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. In the quantification processes, it is assumed that the event tree branch failure probabilities for systems observed failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Event tree branch failure probabilities for systems observed degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. Event tree branch failure probabilities used for systems observed to be successful and systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage remaining once the observed failures have occurred.

A.1.1 ASP Event Tree Models

Models used to rank precursors as to significance consist of plant-class specific event trees that are linked to simplified plant-specific system models. These models describe mitigation sequences for three initiating events: a nonspecific reactor trip [which includes LOFW within the model], LOOP, and small-break LOCA. The event tree models are system-based and include a model applicable to each of eight plant classes: three for BWRs and five for PWRs.

Plant classes are defined based on the use of similar systems in providing protective functions in response to transients, LOOPs, and small-break LOCAs. System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar in response. Plants where certain mitigating systems do not exist, but which are largely analogous in their initiator response, are grouped into the appropriate plant class. In modeling events at such plants, the event tree branch probabilities are modified to reflect the actual systems available at the plant. For operational events that cannot be described using the plant-class specific event trees, unique models are developed to describe the potential sequences to severe core damage.

Each event tree includes two undesired end states. The undesired end states are designated as (1) core damage (CD), in which inadequate core cooling is believed to exist; and (2) ATWS, for the failure-to-scram sequence. The end states are distinct; sequences associated with ATWS are not subsets of core damage sequences. The ATWS sequence, if fully developed, would consist of a number of sequences ending in either success or core damage. Successful operation is designated "OK" in the event trees included in this appendix.

A.1.2 Precursor Impact on Event Tree Branches

The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system failure given the operational event. This translation process is simplified in many cases through the use of train-based models that represent an event tree branch. If a train-based model exists, then the impact of the operational event need only be determined at the train level, and not at the system level.

Once the impact of an operational event on systems included in the ASP event tree models has been determined, branch probability values are modified to reflect the event, and the event trees are then used to estimate a conditional probability of subsequent core damage, given the precursor.

A.1.3 Estimation of Initiating Event Frequencies and Branch Failure Probabilities Used with the Event Tree Models

A set of initiating event frequencies and system failure probabilities was developed for use in the quantification of the event tree models associated with the precursors. The approach used to develop frequency and probability estimates employs failure or initiator data in the precursors themselves when sufficient data exists. When precursor data are available for a system, its failure probability is estimated by counting the effective number of nonrecoverable failures in the observation period, making appropriate demand assumptions, and then calculating the effective number of failures per demand. The number of demands is calculated based on the estimated number of tests per reactor year plus any additional demands to which a system would be expected to respond. This estimate is then multiplied by the number of applicable reactor years in the observation period to determine the total number of demands. A similar approach is employed to estimate initiator frequencies per reactor year from observed initiating events.

The potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes are currently used to describe the different types of recovery that could be involved:

Recovery class	Likelihood of nonrecovery	Recovery characteristic
R1	1.00	The failure did not appear to be recoverable in the required period, either from the control room or at the failed equipment.
R2	0.34	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the control room did not appear possible.
R3	0.12	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.04	The failure appeared recoverable in the required period from the control room and was considered routine and procedurally based.

The assignment of an event to a recovery class is based on engineering judgment, which considers the specifics of each operational event and the likelihood of not recovering from the observed failure in a moderate to high-stress situation following an initiating event. For analysis purposes, consistent probabilities of failing to recover an observed failure are assigned to each event in a particular recovery class. It must be noted that the actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values listed. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.*

The branch probability estimation process is illustrated in Table A.1. Table A.1 lists two operational events that occurred in 1984-86 involving failure of SG isolation. For each event, the likelihood of failing to recover from the failure is listed (Column 3). The effective number of nonrecoverable events (1.04 in this case) is then divided by an estimate of the total number of demands in the 1984-86 observation period (1968) to calculate a failure on demand probability of 5.3×10^{-4}.

The likelihood of system failure as a result of hardware faults is combined with the likelihood that the system could not be recovered, if failed, and with an estimate of the likelihood of the operator failing to initiate the system, if manual initiation were required, to estimate the overall failure probability for an event-tree branch. Calculated failure probabilities are then used to tailor the probabilities associated with train-based system models. Such an approach results in system failure probability estimates that reflect, to a certain extent, the degree of redundancy actually available and permits easy revision of these probabilities based on train failures and unavailabilities observed during an operational event.

*Programmatic constraints have prevented substantial efforts in estimating actual recovery class distributions. The values currently used were developed based on a review of events with the potential for short-term recovery, in addition to consideration of human error during recovery. These values have been reviewed both within and outside the ASP Program. While it is acknowledged that substantial uncertainty exists in them, they are believed adequate for ranking purposes, which is the primary goal of the current precursor calculations. This assessment is supported by the sensitivity and uncertainty calculations documented in the 1980-81 report. These calculations demonstrated little impact on the relative ranking of events from variance in recovery class values.

A.1.4 Conditional Probability Associated with Each Precursor

The calculation process for each precursor involves a determination of initiators that must be modeled and their probability, plus any modifications to system probabilities necessitated by failures observed in an operational event. Once the branch probabilities that reflect the conditions of the precursor are established, the sequences leading to the modeled end states (core damage and ATWS) are calculated and summed to produce an estimate of the conditional probability of each end state for the precursor. So that only the additional contribution to risk (incremental risk) associated with a precursor is calculated, conditional probabilities for precursors associated with equipment unavailabilities (during which no initiating event occurred) are calculated a second time using the same initiating event probability but with all branches assigned normal failure probabilities (no failed or degraded states) and subtracted from the initially calculated values. This eliminates the contribution for sequences unimpacted by the precursor, plus the normal risk contribution for impacted sequences during the unavailability. This calculational process is summarized in Table A.2.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the LWR population, even though they are applied to sequences that are plant-class specific in nature. Because of this, *the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred.* The probabilities calculated in the ASP study are homogenized probabilities considered representative of probabilities resulting from the occurrence of the selected events at plants representative of the plant class.

A.1.5 Sample Calculations

Three hypothetical events are used to illustrate the calculational process.

1. The first event assumes a trip and LOFW but no other observed failures during mitigation. An event tree for this event is shown in Fig. A.1. On the event tree, successful operation is indicated by the upper branch and failure by the lower branch. With the exception of relief valve lift, failure probabilities for branches are indicated. For HPI, the lowest branch includes operator action to initiate feed and bleed. Success probabilities are 1 - p(failure). The likelihood of not recovering the initiator (trip) is assumed to be 1.0, and the likelihood of not recovering MFW is assumed to be 0.34 in this example. Systems assumed available were assigned failure probabilities currently used in the ASP Program. The estimated conditional probabilities for undesirable end states associated with the event are then:

$$
\begin{aligned}
p(\text{cd}) \quad = \ & p[\text{seq. } 11] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times \\
& \qquad 3.3 \times 10^{-4} \times (1 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}] \\
+ \ & p[\text{seq. } 12] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times \\
& \qquad 3.3 \times 10^{-4} \times 8.4 \times 10^{-4}] \\
+ \ & p[\text{seq. } 13] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times 9.9 \times 10^{-5} \times (1 - 0.34) \times 4.0 \times \\
& \qquad 10^{-2} \times 3.3 \times 10^{-4} \times (1.0 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}]
\end{aligned}
$$

$$+ \ p[\text{seq. } 14] + p[\text{seq. } 15] + p[\text{seq. } 16] + p[\text{seq. } 17]$$

$$= 7.7 \times 10^{-7}$$

$$p(\text{ATWS}) = p[\text{seq } 18]$$

$$= 3.0 \times 10^{-5}$$

2. The second example event involves failures that would prevent HPI if required to mitigate a small-break LOCA or if required for feed and bleed. Assume such failures were discovered during testing. This event impacts mitigation of a small-break LOCA initiator and potentially impacts mitigation of a trip and LOOP, should a transient-induced LOCA occur or should feed and bleed be required upon loss of AFW and MFW. The event tree for a postulated small-break LOCA associated with this example precursor is shown in Fig. A.2. The failure probability associated with the precursor event (unavailability of HPI) is assigned based on the likelihood of not recovering from the failure in a 20-30 min time frame (assumed to be 1.0 in this case). No initiating event occurred with the example precursor; however, a failure duration of 360 h was estimated based on one-half of a monthly test interval. The estimated small-break LOCA frequency (assumed to be 1.0×10^{-6}/h in this example), combined with this failure duration, results in an estimated initiating event probability of 3.6×10^{-4} during the unavailability. The probabilities for small-LOCA sequences involving undesirable end states (employing the same calculational method as above and subtracting the nominal risk during the time interval) are 3.6×10^{-4} for core damage and 0.0 for ATWS. Note that the impact of the postulated failure on the ATWS sequence is zero because HPI success or failure does not impact that sequence as modeled.

   For most unavailabilities, similar calculations would be required using the trip and LOOP event trees, since these postulated initiators could also occur. In this example, neither of these two initiators contributes substantially to the core damage probability associated with the event.

3. The third example event involves a trip with unavailability of one of two trains of service water (SW). Assumed unavailability of the SW train results in unavailability of one train of HPI, high-pressure recirculation (HPR), and AFW, all because of unavailability of cooling to the respective pumps. In this example, SW cooling of two motor-driven AFW pumps is assumed. An additional turbine-driven pump is assumed to be self-cooled. Since SW is not explicitly addressed in the ASP event trees, the probabilities of front-line systems impacted by the loss of SW are instead modified.

   Figure A.3 shows a transient event tree with branch failure probabilities modified to reflect unavailability of one train of service water. The likelihoods of not recovering failed front line systems are assumed to be unchanged, since the failure mechanisms for (observed) non-faulted trains are expected to be consistent with historically observed failures. The conditional probability of core damage given the trip and one service water train unavailable is 1.1×10^{-6}. If the second train of service water were to fail, HPI and HPR (and hence feed and bleed) would be rendered unavailable; however, the turbine-driven AFW pump would still be operable. In this case, the likelihood of not recovering HPI and HPR is assumed to be 1.0 until service water is recovered. Sequences associated with loss of both service water trains increase the core damage probability associated with the event. The extent of this increase is dependent in PWRs on the likelihood of a reactor coolant pump seal failure following the loss of service water (since seal injection and seal cooling would be typically lost). Assuming that the conditional probability of loss of the second service water train is 0.01, that the likelihood of not recovering SW is 0.34, and that the failure probability of the turbine-driven AFW pump is 0.05, the increase in core damage probability is 1.7×10^{-4} if no RCP seal failure occurs, and 3.4×10^{-3} if the likelihood of seal failure is 1.0.

A.1.6 Event Tree Changes Made to 1988–1991 Event Models

Two changes were made to the event trees used in the 1988-91 precursor assessments: core vulnerability sequences on trees used for 1984-87 assessments were reassigned as success or core damage sequences, and the likelihood of PWR RCP seal LOCA following station blackout was explicitly modeled.

In the prior models, the core vulnerability end state was assigned to sequences in which core protection was expected to be provided but for which no specific analytic basis was generally available or which involved non-proceduralized operator actions. Core vulnerability sequences were assigned to either success or core damage end states in the current models, as follows:

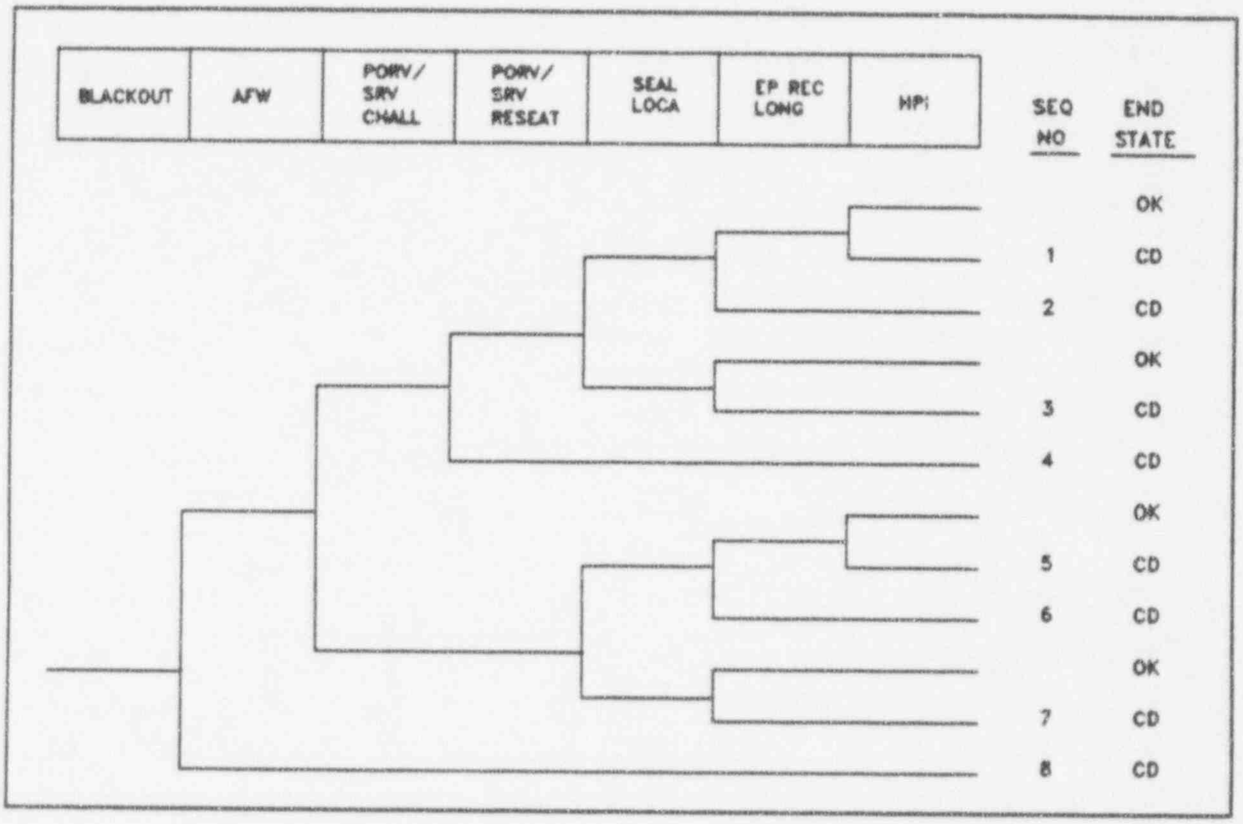
Core vulnerability sequence type	Revised end state
Stuck-open secondary-side relief valve with a failure of HPI in a PWR	Success
Steam generator (SG) depressurization and use of condensate system following failure of AFW, MFW, and feed and bleed in a PWR	Core damage (except for PWR Class H)
Use of containment venting as an alternate core cooling method in a BWR	Core damage

The net effect of this change is a significant reduction in the complexity of the event trees, with little impact on the relative significance estimated for each precursor. The impact of this modeling change on conditional probability estimates for 1987 precursors is described in Sect. 3.6 of Ref. 1. (Alternate calculations using models with the above changes were performed on 1987 events.) As illustrated in Ref. 1, modest differences existed between the core damage, core damage plus core vulnerability, and revised core damage model conditional probability estimates for most of the more significant events. Where differences did exist, the sum of probabilities of core damage and core vulnerability (all non-ATWS undesirable end states in the earlier models) was closer to the core damage probability estimated with the revised models.

Three 1987 events had substantially higher "sum" probabilities—these events involved trips with single safety-related train unavailabilities, for which the dominant core vulnerability sequence was a stuck-open secondary-side relief valve with HPI failure (assigned to success in the revised models).

The second modeling change was the inclusion of PWR RCP seal LOCA in blackout sequences. The impact of such a seal LOCA on the core damage probability estimated for an event had previously been bounded by the use of a conservative value for failure to recover ac power prior to battery depletion following a LOOP and loss of emergency power.

The PWR event trees have been revised to address potential seal LOCA during station blackout through the use of seal LOCA and electric power recovery branches, as shown below:



Two time periods are represented in the sequences in the above figure. Auxiliary feedwater, power-operated relief valve/safety relief valve (PORV/SRV) challenge, and PORV/SRV reseat are shot-term responses following loss of the diesel generators. If turbine-driven AFW is unavailable, or if an open PORV/SRV fails to close, then core damage is assumed to occur, since no high-pressure injection is available as an alternate means of core cooling or for RCS makeup. SEAL LOCA, EP REC LONG, and HPI are branches applicable in the long term. SEAL LOCA represents the likelihood of a seal LOCA prior to restoration of ac power. EP REC LONG represents the likelihood of not restoring ac power prior to core uncovery (if a seal LOCA exists) or prior to battery depletion (in the case of no seal LOCA). Once the batteries are depleted, core damage is assumed to occur, since control of turbine-driven pumps and the ability to monitor core and RCS conditions are lost. HPI represents the likelihood of failing to provide HPI following a seal LOCA to prevent core damage. The ASP models have been simplified somewhat by assuming that HPI is always adequate to make up for flow from a failed seal or seals.

The three seal LOCA-related sequences are illustrated in sequences 1, 2, and 3. In sequence 1, a seal LOCA occurs prior to restoration of ac power, ac power is successfully restored prior to core uncovery, but HPI fails to provide makeup flow. In sequence 2, a seal LOCA also occurs, and ac power is not restored prior to core uncovery. In sequence 3, no seal LOCA occurs, but ac power is not recovered prior to battery depletion. The likelihood of seal LOCA prior to ac power restoration and the likelihood of ac power recovery are time-dependent, and this time-dependency is accounted for in the analysis. A more detailed description of the changes associated with explicitly modeling RCP seal LOCA is included in Ref. 2.

In addition to elimination of core vulnerability sequences, two other changes were made to simplify the previously complex BWR event trees:

- Failure to trip with soluble boron injection success was previously developed in detail and involved a large number of low probability sequences. All failure to trip sequences are now assigned to the ATWS end state.

- The condensate system was previously modeled as an alternate source of low-pressure injection water. This use of the condensate system is now considered a recovery action. this reduces the number of sequences on the event trees without substantially impacting the core damage probability estimates developed using the trees. Systems addressed on the event trees for low-pressure injection include LPCS, LPCI, and RHRSW.

A.2 Plant Categorization

Both the 1969–79 and 1980–81 precursor reports (Refs. 1 and 2) used simplified, functionally based event trees to model potential event sequences. One set of event trees was used to model for PWR initiating events: LOFW, LOOP, small-break LOCA, and steam line break. A separate set of event trees was used to model BWR response to the same initiators. Operational events that could not be modeled using these "standardized" event trees were addressed using models specifically developed for the event.

It was recognized during the review of the 1969-79 precursor report that plant designs were sufficiently different that multiple models would be required to more correctly describe the impact of an operational event in different plants. In 1985, substantial effort was expended to develop a categorization scheme for all U.S. LWRs that would permit grouping of plants with similar response to a transient or accident at the system or functional level, and to subsequently develop eight sets of plant-class specific event tree models. Much of the categorization and early event sequence work was done at the University of Maryland (Refs. 3 and 4). The ASP Program has generally employed these categorizations; however, some modifications have been required to reflect more closely the specific needs of the precursor evaluations.

In developing the plant categorizations, each reactor plant was examined to determine the systems used to perform the following plant functions required in response to reactor trip, LOOP, and small-break LOCA initiators to prevent core damage: reactor subcriticality, RCS integrity, reactor coolant inventory, short-term core heat removal, and long-term core heat removal.

Functions related to containment integrity (containment overpressure protection and containment heat removal) and post-accident reactivity removal are not included on the present ASP event trees (which only concern core damage sequences) and are not addressed in the categorization scheme.

For each plant, systems utilized to perform each function were identified. Plants were grouped based on the use of nominally identical systems to perform each function; that is, systems of the same type and function without accounting for the differences in the design of those systems.

Three BWR plant classes were defined. BWR Class A consists of the older plants, which are characterized by isolation condensers (ICs) and feedwater coolant injection (FWCI) systems that employ the MFW pumps. BWR Class B consists of plants that have ICs but a separate HPCI system instead of FWCI. BWR Class C includes the modern plants that have neither ICs nor FWCI. However, they have a RCIC system that Classes A and B lack. The Class C plants could be separated into two subgroups, those plants with turbine-driven HPCI systems and those with motor-driven HPCS systems. This difference is addressed instead in the probabilities assigned to branches impacted by the use of these different system designs.

PWRs are separated into five classes. One class represents most Babcock & Wilcox Company plants (Class D). These plants have the capability of performing feed and bleed without the need to open the PORV. Combustion Engineering plants are separated into two classes, those that provide feed and bleed capability (Class G) and those that provide for secondary-side depressurization and the use of the condensate system as an alternate core cooling method, and for which no feed and bleed is available (Class H).[**]

The remaining two classes address Westinghouse plants – Class A is associated with plants that require the use of spray systems for core heat removal following a LOCA, and Class B is associated with plants that can utilize low-to-high pressure recirculation for core heat removal.

Plants in which initiator response cannot be described using plant-class models are addressed using unique models, for example, the now deactivated LaCrosse BWR.

Table A.17 lists the class associated with each plant.

A.3 Event Tree Models

The plant class event trees describe core damage sequences for three initiating events: a nonspecific reactor trip, a LOOP, and a small-break LOCA. The event trees constructed are system-based and include an event tree applicable to each plant class defined.

System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar. Plants where certain mitigating systems do not exist, but which are largely analogous in their transient response, were grouped into the plant classes accordingly. In modeling events at such plants, the event tree branch probabilities were modified to reflect the systems available at the plant. Certain events (such as a postulated steam line break) could not be described using the plant-class event trees presented in this appendix. In these cases, unique event trees were developed to describe the sequences of interest.

[**]Maine Yankee Atomic Power Plant was built by Combustion Engineering but has a response to initiating events more akin to the Westinghouse Electric Corporation design, so it is grouped in a class with other Westinghouse plants. Davis-Besse Nuclear Power Station was also placed in a Westinghouse plant class because its HPI system design requires the operator to open the PORV for feed and bleed, as in most Westinghouse plants. The requirement to open the PORV for feed and bleed is a primary difference between event trees for Westinghouse and Babcock and Wilcox plants. Plant response differences resulting from the use of different SG designs are not addressed in the models.

This section (1) describes the potential plant response to the three initiating events described above, (2) identifies the combinations of systems required for the successful mitigation of each initiator, and (3) briefly describes the criteria for success of each system-based function. The sequences are considered first for PWRs and then separately for BWRs. PWR Class B event trees are described first, along with those for Class D, which are similar. (The major difference between Class B and Class D plants is that PORV operability is not required for feed and bleed on Class D plants.) The event trees for the combined group apply to the greatest number of operating PWRs. Therefore, these are discussed first, followed by those for PWR Classes G, H, and then A. For the BWR event trees, the plant Class C models are described first, because these are applicable to the majority of the BWRs, followed by discussions for the A and B BWR classes, respectively. The event trees are constructed with branch (event or system) success as the upper branch and failure as the lower branch. Each sequence path is read from left to right, beginning with the initiator followed by subsequent systems required to preclude or mitigate core damage.

The event trees can be found following the discussion sections and are grouped according to plant classes, beginning with the PWR classes and followed by the BWR classes. The abbreviations used in the event tree models are defined in Table A.16 preceding the event trees. Sequence numbers are provided on the event trees for undesirable end states (core damage and ATWS). Because of the similarities among PWR sequences for different plant classes, common sequence numbers have been assigned when possible. PWR Class B sequences were used as a basis for this. Sequence numbers beyond those for Class B are used for uncommon sequences on other plant classes. This approach facilitates comparison of sequences among plant classes. This approach could not be used for BWRs because of the significant difference in systems used on plants in the three plant classes. For BWRs, sequences are numbered in increasing order moving down each event tree. The following sequence number groups are employed for all event trees: transient with reactor trip success, 11-39; LOOP with reactor trip success, 40-69; small-break LOCA with reactor trip success, 71-79; ATWS sequences, 91-99.

The trees are presented in the following order:

Figure No.	Event tree
A.4	PWR Class A nonspecific reactor trip
A.5	PWR Class A loss of offsite power
A.6	PWR Class A small-break loss-of-coolant accident
A.7	PWR Classes B and D nonspecific reactor trip
A.8	PWR Classes B and D loss of offsite power
A.9	PWR Classes B and D small-break loss-of-coolant accident
A.10	PWR Class G nonspecific reactor trip
A.11	PWR Class G loss of offsite power
A.12	PWR Class G small-break loss-of-coolant accident
A.13	PWR Class H nonspecific reactor trip
A.14	PWR Class H loss of offsite power
A.15	PWR Class H small-break loss-of-coolant accident
A.16	BWR Class A nonspecific reactor trip
A.17	BWR Class A loss of offsite power
A.18	BWR Class A small-break loss-of-coolant accident
A.19	BWR Class B nonspecific reactor trip
A.20	BWR Class B loss of offsite power
A.21	BWR Class B small-break loss-of-coolant accident
A.22	BWR Class C nonspecific reactor trip

A.3.1 PWR Event Sequence Models

The PWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following three initiating events: reactor trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to an initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor trip
Reactor coolant system integrity:	Addressed in small-break LOCA models plus trip and LOOP sequences involving failure of primary relief valves to close
Reactor coolant inventory:	High-pressure injection (assumed required only following a LOCA)
Short-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure injection and PORV (feed and bleed, PWR Classes A, B, D, and G)
	Secondary-side depressurization and use of condensate system (PWR Class H)
Long-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure recirculation (PWR Classes B and D) (also required to support RCS inventory for all classes)
	Secondary-side depressurization and use of condensate system (PWR Class H)
	Containment spray recirculation (PWR Classes A and G)

PWR Nonspecific Reactor Trip

The PWR nonspecific reactor trip event tree constructed for plant Classes B and D is shown in Fig. A.7. The event-tree branches and the sequences leading to severe core damage and ATWS follow.

1. Initiating event (transient). The initiating event for the tree is a transient or upset event that requires or is followed by a rapid shutdown of the plant. LOOP and small-break LOCA initiators are modeled in separate event trees. Large-break LOCA or large SLB initiators are not addressed in the models described here.

2. Reactor trip. To achieve reactor subcriticality and thus halt the fission process, the reactor protection system (RPS) is required to insert control rods into the core. If the automatically initiated RPS fails, a reactor trip may be initiated manually. Failure to trip was considered to lead to the end state ATWS and was not developed further.

3. Auxiliary feedwater. AFW must be provided following trip to remove the decay heat still being generated in the reactor core via the SGs. Successful AFW operation requires flow from one or more AFW pumps to one or more SGs over a period of time ranging from 12 to 24 h (typically, one pump to one SG is adequate).

4. Main feedwater. In lieu of AFW, MFW can be utilized to remove the post shutdown decay heat. Depending on the individual plant design, either main or AFW may be used as the primary source of secondary-side heat removal.

5. PORV or SRV challenged. For sequences in which both reactor trip and steam generator feedwater flow (MFW or AFW) have been successful, the pressurizer PORV may or may not lift, depending on the peak pressurizer pressure following the transient. (In most transients, these valves do not lift.) The upper branch indicates that the valve or valves were challenged and opened. Because of the multiplicity of relief and safety valves, it was assumed that a sufficient number would open if the demand from a pressure transient exists.

   The lower branch indicates that the pressurizer pressure was not sufficiently high to cause opening of a relief valve. For the sequence in which both AFW and MFW fail following a reactor trip, at least one PORV or SRV was assumed to open for overpressure protection.

6. PORV or SRV reseats. Success for this branch requires the closure of any open relief valve once pressurizer pressure has decreased below the relief valve set point. If a PORV sticks open, most plants are equipped with an isolation valve that allows for manual termination of the blowdown. Failure of a primary-side relief valve to close results in a transient-induced LOCA that is modeled as part of this event tree.

7. High-pressure injection. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered. Success for this branch requires introduction of sufficient borated water to keep the core covered, considering core decay heat. (Typically, one HPI train is sufficient for this purpose.)

8. HPI and PORV open. If normal methods of achieving decay heat removal via the SGs (MFW and AFW) are unavailable, core cooling can be accomplished on most plants by establishing a feed and bleed operation. This operation (1) allows heat removal via discharge of reactor coolant to the containment through the PORVs and (2) RCS makeup via injection of borated water from the HPI system. Except at Class D plants, successful feed and bleed requires the operator to open the PORV manually. At Class D plants, the HPI discharge pressure is high enough to lift the primary-side safety valves, and feed and bleed can be accomplished without the operator manually opening the PORVs. HPI success is dependent on plant design but requires the introduction of sufficient

amounts of borated water into the RCS to remove decay heat and provide sufficient reactor coolant makeup to prevent core damage.

9. High-pressure recirculation. Following a transient-induced LOCA (a PORV or SRV fails to reseat), or failure of secondary-side cooling (AFW and MFW) and initiation of feed and bleed, continued core cooling and makeup are required. This requirement can be satisfied by using HPI in the recirculation mode. In this mode the HPI pumps recirculate reactor coolant collected in the containment sump and pass it through heat exchangers for heat removal. When MFW or AFW is available, heat removal is only required for HPI pump cooling; if AFW or MFW is not available, HPR is required to remove decay heat as well. Typically, at Class B and D plants, the LPI pumps are utilized in the HPR mode, taking suction from the containment sump, passing the pumped water through heat exchangers, and providing net positive suction head to the HPI pumps.

The event tree applicable to a PWR Class G nonspecific reactor trip is shown in Fig. A.10. Many of the event tree branches and the sequences leading to successful transient mitigation and core damage are similar to those following a nonspecific reactor trip transient for plant Class B. At Class G plants, however, the HPR system performs both the high- and low-pressure recirculation (LPR), function, taking suction directly from the containment sump without the aid of the low-pressure pumps. DHR is accomplished during recirculation by the containment spray recirculation (CSR) system. The event-tree branches and sequences are discussed further.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for PWR Classes B and D. The following branches have functions and success requirements similar to those following a transient at PWR Class B.

2. Reactor trip.

3. Auxiliary feedwater or main feedwater.

4. PORV or SRV challenged reseats.

5. High-pressure injection.

6. HPI and PORV open (feed and bleed). Success requirements for feed and bleed are similar to those following the plant Class B transient. Feed and bleed with operator opening of the PORV is required in the event that both AFW and MFW are unavailable for secondary-side cooling. In addition, DHR was assumed required to prevent potential core damage. This is provided by the CSR system.

7. High-pressure recirculation. In the event of a transient-induced LOCA, continued HPI via sump recirculation is needed to provide makeup to the break to prevent potential core damage. In addition, HPR is required when both AFW and MFW are unavailable following a transient, to recirculate coolant during the feed and bleed procedure. If HPR fails and normal secondary-side cooling is also failed, core damage will occur. In Class G plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

8. Containment spray recirculation. When feed and bleed (HPI, HPR, and PORV open) is required, the CSR system operates to remove decay heat from the reactor coolant being recirculated. Without the CSR system, the feed and bleed operation could not remove decay heat. Successful operation of feed and bleed and CSR was assumed to result in successful mitigation of core damage.

The event tree for PWR Class H non-specific reactor trip is shown in Fig. A.13. This class of plants is different than other PWR classes in that PORVs are not included in the plant design and feed and bleed cannot be used to remove decay heat in the event of main and AFW unavailability. If main or AFW cannot be recovered, the atmospheric dump valves can be used to depressurize the SGs to below the shutoff head of the condensate pumps, and these can be used, if available, for RCS cooling. Because of the need for secondary-side cooling for all success sequences, a requirement for CC to prevent core damage has not been modeled.

1. Initiating event (transient). The initiating event is a non-specific reactor trip, similar to that described for the previous PWR classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with previously described PWR classes.

2. Reactor trip.

3. Auxiliary feedwater.

4. Main feedwater.

5. SRV challenged. The upper branch indicates that at least one safety valve has lifted as a result of the transient. In most transients in which reactor trip has been successful and main or AFW is available, these valves do not lift. In the case where both main and AFW are unavailable, at least one SRV is assumed to lift. The lower branch indicates that the pressurizer pressure was not sufficiently high to cause the opening of a relief valve.

6. SRV reseat. Success for this branch requires the closure of any open safety valve once pressurizer pressure has been reduced below the safety valve set point.

7. High-pressure injection. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered.

8. High-pressure recirculation. The requirement for continued core cooling during mitigation of a transient-induced LOCA and following depletion of the refueling water tank can be satisfied by using HPI in the recirculation mode. In Class H plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

9. Steam generator depressurization. In the event that main and AFW are unavailable, the atmospheric dump valves (or turbine bypass valves if the main steam isolation valves are open) may be used on Class H plants to depressurize the SGs to the point that the condensate pumps can be used for SG cooling. In the event of main and AFW unavailability, failure to depressurize one SG to the operating pressure of the condensate system is assumed to result in core damage.

10. Condensate pumps. As described above, use of the condensate pumps on Class H plants along with secondary-side depressurization can provide adequate core cooling. Flow from one condensate

pump to one SG is assumed adequate. Unavailability of the condensate pumps in the event of failure to recover main and AFW is assumed to result in core damage.

The event tree applicable to PWR plant Class A nonspecific reactor trip is shown in Fig. A.4. Many of the event-tree branches and the sequences leading to successful transient mitigation and severe core damage are similar to those following a nonspecific reactor trip transient for plant Classes B and G.

Like the Class G plants, the Class A plants have a CSR system that provides DHR during HPR. Use of CSR for DHR was assumed to be required if AFW and MFW were unavailable. LPI pumps are required to provide suction to the HPI pumps during recirculation. The event-tree branches and sequences are discussed further below.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for the other PWR plant classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with plant Classes B, D, and G.

2. Reactor trip.

3. Auxiliary feedwater

4. Main feedwater.

5. PORV or SRV challenged.

6. PORV/SRV reseats.

7. High-pressure injection.

8. High-pressure recirculation. In the event of a transient-induced LOCA, HPR can provide sufficient makeup to the break to terminate the transient. The LPI pumps provide suction to the high-pressure pumps in the recirculation mode. In the event that feed and bleed is required (following a transient in which both AFW and MFW are unavailable), HPR success is required.

9. Containment spray recirculation. The CSR system provides DHR during HPR when AFW and MFW are not available. In transient-induced LOCA sequences, HPI and HPR success is required to mitigate the event. In the event that secondary-side cooling via AFW or MFW is unavailable, feed and bleed with CSR, for DHR is considered sufficient to prevent core damage.

10. PORV open. The PORV must be opened by the operator below its set point to establish feed and bleed operation in the event that secondary-side cooling via AFW or MFW is unavailable.

Sequences resulting in core damage or ATWS following a PWR transient, shown on event trees applicable to each plant class, are described in Table A.4.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the use of SG depressurization and condensate pumps for RCS cooling in lieu of feed and bleed on Class H. Because of this similarity, consistent sequence numbers have been used for like sequences in different PWR plant classes. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.5.

PWR Loss of Offsite Power

The event trees constructed define representative plant responses to a LOOP. A LOOP (without turbine runback on plants with this feature) will result in reactor trip due to unavailability of power to the control rod drive (CRD) mechanisms and a loss of MFW because of the unavailability of power to components in the condensate and condenser cooling systems.

The PWR LOOP tree constructed for plant Classes B and D is shown in Fig. A.8. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (LOOP). The initiating event for the tree is a grid or switchyard disturbance to the extent that the generator must be separated from the grid and all offsite power sources are unavailable to plant equipment. The capability of a runback of the unit generator from full power to supply house loads exists at some plants but is not considered in the event tree. Only LOOPs that challenge the emergency power system (EPS) are addressed in the ASP Program.

2. Reactor trip given LOOP. Unavailability of power to the CRD mechanisms is expected to result in a reactor trip and rapid shutdown of the plant. If the reactor trip does not occur, the transient was considered to proceed to ATWS and was not developed further.

3. Emergency power. Given a LOOP and a reactor trip, electric power would be lost to all loads not backed by battery power. When power is lost, DGs are automatically started to provide power to the plant safety-related loads. Emergency power success requires the starting and loading of a sufficient number of DGs to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

4. Auxiliary feedwater. The AFW system functions to remove decay heat via the SG secondary side. Success requirements for this branch are equivalent to those following a nonspecific reactor trip and unavailability of MFW. Both MFW and condensate pumps would be unavailable following a LOOP. Therefore, with emergency power and AFW failed, no core cooling would be available, and core damage would be expected to occur. Because, specific AFW systems may contain different combinations of turbine-driven and motor-driven AFW pumps, the capability of the system to meet its success requirements will depend on the state of the EPS and the number of turbine-driven AFW pumps that are available.

5. PORV or SRV challenged. The upper and lower states for this branch are similar to those following a nonspecific reactor trip. The PORV or SRV may or may not lift, depending on the peak pressure following the transient.

6. PORV or SRV reseats. The success requirements for this branch are similar to those following a nonspecific reactor trip. However, for the sequence in which emergency power is failed and the PORV fails to reseat, the HPI/HPR system would be without power to mitigate potential core damage.

7. Seal LOCA. In the event of a loss of emergency power following LOOP, both SW and component cooling water (CCW) are faulted. This results in unavailability of RCP seal cooling and seal injection (since the charging pumps are also without power and cooling water). Unavailability of seal cooling and injection may result in seal failure after a period of time, depending on the seal

design (for some seal designs, seal failure can be prevented by isolating the seal return isolation valve).

The upper event tree branch represents the situation in which seal failure occurs prior to restoration of ac power. The lower branch represents the situation in which a seal LOCA does not occur.

8. Electric power recovered (long term). For sequences in which a seal LOCA has occurred, success requirements are the restoration of ac power [either through recovery of offsite power or recovery of a DG] prior to core uncovery. For sequences in which a seal LOCA does not occur, success requires the recovery of ac power prior to battery depletion, typically 2 to 4 h.

9. High-pressure injection and recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. Because all HPI/HPR systems use motor-driven pumps, the capability of the HPI or HPR system to meet its success requirements depends on the success of the EPS.

10. PORV open (for feed and bleed). The success requirements for this branch are similar to those following a nonspecific reactor trip. The PORV is opened in conjunction with feed and bleed operations when secondary-side heat removal is unavailable. For Class D plants, the PORV does not have to be manually opened to establish feed and bleed because the HPI pump discharge pressure is high enough to lift the PORV or primary relief valve.

The event tree constructed for the PWR Class G LOOP is shown in Fig. A.11. Most of the event-tree branches and the sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, at Class G plants, DHR during recirculation is provided by the CSR system, not the HPR system. The event-tree branches and sequences are discussed further below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for PWR plant Classes B and D. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

5. PORV or SRV challenged.

6. PORV/SRV valve reseats.

7. Seal LOCA.

8. Electric power recovered (long term).

9. High-pressure injection and recirculation.

10. PORV open (for feed and bleed).

11. Containment spray recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. The CSR system provides DHR for sequences in which secondary-side cooling is unavailable.

The event tree constructed for a PWR Class H LOOP is shown in Fig. A.14. Many of the event tree branches and sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, Class H plants do not have feed and bleed capability and rely instead on secondary-side depressurization and the condensate system as an alternate DHR method. The condensate system is assumed unavailable following a LOOP, which limits the diversity of DHR methods on this plant class following this initiator. The event branches and sequences are discussed further below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for BWR Classes B and D. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

5. SRV challenged. The function of this branch is similar to that described under the PWR Class H transient.

6. SRV reseat. Success requirements for this branch are similar to those described under the PWR Class H transient.

7. Seal LOCA.

8. Electric power recovered (long-term).

9. High pressure injection and recirculation.

The event tree constructed for the plant Class A LOOP is shown in Fig. A.5. All of the event-tree branches and the sequences leading to successful transient mitigation, potential core vulnerability, and severe core damage are analogous to those following a LOOP at Class B plants with the addition of the CSR branch, which is required for successful feed and bleed. At Class A plants, DHR during HPR is accomplished by the CSR system; whereas at Class B and D plants, DHR is an integral part of the HPR system. Additional information on the use of the CSR system is provided in the discussion of the PWR Class A nonspecific reactor trip event tree.

Sequences resulting in core damage and ATWS following a PWR LOOP, shown on event trees applicable to each plant class, are described in Table A.6.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the unavailability of feed and bleed on Class H. As with the PWR transient sequences, this similarity permits consistent numbering of a large number of sequences. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.7.

PWR Small-Break Loss-of-Coolant Accident

Event trees were constructed to define the responses of PWRs to a small-break LOCA. The LOCA chosen for consideration is one that would require a reactor trip and continued HPI for core protection. Because of the limited amount of borated water available, the mitigation sequence also includes the requirement to recirculate borated water from the containment sump.

The LOCA event tree constructed for PWR plant Classes B and D is shown in Fig. A.9. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (small-break LOCA). The initiating event for the tree is a small-break LOCA that requires reactor trip and continued HPI for core protection.

2. Reactor trip. Reactor trip success is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition. Failure to trip was considered to lead to the end state ATWS.

3. Auxiliary feedwater or main feedwater. Use of AFW or MFW was assumed necessary for some small breaks to reduce RCS pressure to the point where HPI is effective. At Class D plants, the HPI pumps operate at a much higher discharge pressure and hence can function without secondary-side cooling from the AFW or MFW systems.

4. High-pressure injection. Adequate injection of borated water from the HPI system is required to prevent excessive core temperatures and consequent core damage.

5. High-pressure recirculation. Following a small-break LOCA, continued high pressure injection is required. This is typically accomplished with the residual heat removal (RHR) system, which takes suction from the containment sump and returns the lost reactor coolant to the core via the HPI pumps. The RHR system includes heat exchangers that remove decay heat prior to recirculating the sump water to the RCS.

6. PORV open. In the event AFW and MFW are unavailable following a small break LOCA, opening the PORV can result in core cooling using the feed and bleed mode. Depending on the size of the small break, opening the PORV may not be required for success. PORV open is not required for success for Class D.

The event tree constructed for a small-break LOCA at Class G plants is shown in Fig. A.12. The LOCA event tree for Class G plants is similar to that for Class B and D plants except that long-term cooling is provided by the CSR system rather than by the HPR system. The event-tree branches and sequences are discussed further below.

1. Initiating event (small-break LOCA). The initiating event is a LOCA similar to that described for PWR plant Classes B and D. The following branches have functions and success requirements similar to those following a small-break LOCA at PWRs associated with all of the plant classes defined.

2. Reactor trip.

3. Auxiliary feedwater and main feedwater

4. High-pressure injection.

5. High-pressure recirculation.

6. PORV open.

7. Containment spray recirculation. In the event that normal secondary-side cooling (AFW or MFW) is unavailable following a small LOCA, cooling via the CSR system during HPR is required to mitigate the transient.

The event tree constructed for a small-break LOCA at PWR Class H plants is shown in Fig. A.15. The event tree has been developed assuming that SG depressurization and condensate pumps can provide adequate RCS pressure reduction in the event of an unavailability of AFW and MFW to permit HPI and HPR to function in these plants. The event tree branches and sequences are discussed further below.

1. Initiating event (small-break LOCA). The initiating event is similar to that described above for PWR Classes B, D, and G. The following branches have functions and success requirements similar to those discussed previously.

2. Reactor trip.

3. Auxiliary and main feedwater.

4. High-pressure injection.

5. High-pressure recirculation.

6. SG depressurization. In the event that AFW and MFW are unavailable following a small-break LOCA, SG depressurization combined with the use of the condensate pumps can provide for RCS depressurization such that adequate HPI and HPR can be achieved. Success requirements are the same as those following a transient with unavailability of AFW and MFW.

7. Condensate pumps. Use of one condensate pump provided flow to at least one SG as required in conjunction with SG depressurization to provide for RCS depressurization and cooling.

The event tree constructed for a small LOCA at Class A plants is shown in Fig. A.6. The LOCA event tree for Class A plants is similar to that for Classes B and D except that the CSR system is required in conjunction with HPR in some sequences where secondary cooling is not provided. The sequences that follow combined AFW and MFW failure with HPR and CSR success are identical to those that follow HPR success at Class B and D plants; and sequences that follow HPR or CSR failure at Class A plants are identical to those that follow HPR failure.

Sequences resulting in core damage or ATWS following a PWR small-break LOCA, shown on event trees applicable to each plant class, are described in Table A.8.

As with the PWR transient and LOOP sequences, differences between plant classes are driven by the use of CSR on plant classes A and G, and by the use of secondary-side depressurization and condensate pumps in lieu of feed and bleed on PWR Class H. All small-break LOCA sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.9.

Alternate Recovery Actions

The PWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection from core damage are unavailable. In the event AFW and MFW are unavailable and cannot be recovered in the short term, the use of feed and bleed cooling is modeled on all plants except for Class H, where SG depressurization and use of the condensate pumps is modeled instead. In addition, the potential for short-term recovery of a faulted system is also included in appropriate branch models (AFW, MFW, and HPI, for example).

Alternate equipment and procedures, beyond the systems and functions included in the event trees, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental DGs, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.

- Depressurization following a small-break LOCA to the initiation pressure of the LPI systems to provide RCS makeup in the event that HPI fails. Procedures to support this action are known to exist on some plants.

- Depressurization following a small-break LOCA to the initiation pressure of the DHR system, and then proceeding to cold shutdown. While plant procedures specify the use of sump recirculation following a small LOCA or feed and bleed, sufficient RWST inventory exists to delay this action until many hours into the event, during which recovery of faulted systems may be affected. It is likely that operators will delay sump recirculation as long as possible while trying to place the plant in a stable condition through recovery of secondary-side cooling and the use of RHR.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.3.2 BWR Event Sequence Models

The BWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following the same three initiating events addressed for PWRs: trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to any initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor scram
Reactor coolant system integrity:	Addressed in small-break LOCA models and in trip and LOOP sequences involving failure of primary relief valves to reseat
Reactor coolant inventory:	High-pressure injection systems [HPCI or HPCS, RCIC (non-LOCA situations), CRD (non-LOCA situations), FWCI]
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS, RHRSW or equivalent]
Short-term core heat removal:	Power conversion system
	High-pressure injection systems [HPCI, RCIC, CRD, FWCI (BWR Class A)]
	Isolation condenser (BWR Classes A and B)
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS]
	Note: Short-term core heat removal to the suppression pool (all cases where power conversion system is faulted) requires use of the RHR system for containment heat removal in the long term.
Long-term core heat removal:	Power conversion system
	Isolation condenser (BWR Class A)
	Residual heat removal [shutdown cooling or suppression pool cooling modes (BWR Class C)]
	Shutdown cooling (BWR Classes A and B)
	Containment cooling (BWR Class A)
	Low-pressure coolant injection [CC mode (BWR Class B)]

BWR Nonspecific Reactor Trip

The nonspecific reactor trip event tree constructed for BWR plant Class C is shown in Fig. A.22. The event tree branches and the sequences leading to potential severe core damage follow. The Class C plants are discussed first because all but a few of the BWRs fit into the Class C category.

1. Initiating event (transient). The initiating event is a transient or upset event that results in a rapid shutdown of the plant. Transients that are initiated by a LOOP or a small-break LOCA are modeled

in separate event trees. Transients initiated by a large-break LOCA or large SLB are not addressed in the event trees described here; trees applicable to such initiators are developed separately if required.

2. Reactor shutdown. To achieve reactor subcriticality and thus halt the fission process, the RPS commands rapid insertion of the control rods into the core. Successful scram requires rapid insertion of control rods with no more than two adjacent control rods failing to insert.

3. Power conversion system (PCS). Upon successful reactor scram, continued operation of the PCS would allow continued heat removal via the main condenser. This is considered successful mitigation of the transient. Continued operation of the PCS requires the MSIVs to remain open and the operation of the condenser, the turbine bypass system (TBS), the condensate pumps, the condensate booster pumps, and the feedwater pumps.

4. SRV challenged. Depending on the transient, one or more SRVs may open. The upper branch on the event tree indicates that the valves were challenged and opened. If the transient is followed by continued PCS operation and successful scram, the SRVs are not expected to be challenged. If the PCS is unavailable, at least some of the SRVs are assumed to be challenged and to open.

5. SRV close. Success for this branch requires the reseating of any open relief valves once the reactor pressure vessel (RPV) pressure decreases below the relief valve set point. If an SRV sticks open, a transient-induced LOCA is initiated.

6. Feedwater. Given unavailability of the PCS, continued delivery of feedwater to the RPV will keep the core from becoming uncovered. This, in combination with successful long-term DHR, will mitigate the transient, preventing core damage. For plants with turbine-driven feed pumps, the PCS failure with subsequent feedwater success cannot involve MSIV closure, or loss of condenser vacuum, because this would disable the feed pumps.

7. HPCI or HPCS. The primary function of the HPCI or HPCS system is to provide makeup following small-break LOCAs while the reactor is at high-pressure (not depressurized). The system is also used for DHR following transients involving a loss of feedwater. Some later Class C plants are equipped with HPCS systems, but the majority are equipped with HPCI systems. HPCI or HPCS can provide the required makeup and short-term DHR when DHR is unavailable from the condenser and the feedwater system cannot provide makeup.

8. RCIC. The RCIC system is designed to provide high-pressure coolant makeup for transients that result in LOFW. Both RCIC and HPCI (or HPCS) initiate when the reactor coolant inventory drops to the low-low level set point, taking suction from the condensate storage tank or the suppression pool. HPCI is normally secured after HPCI/RCIC initiation when pressure and water level are restored, to prevent tripping of HPCI and RCIC pumps on high water level. RCIC must then be operated until the RHR system can be placed in service. Following a transient, scram, and unavailability of the PCS, reactor pressure may increase, causing the relief valves to open and close periodically to maintain reactor pressure control.

9. CRD pumps. In transient-induced sequences where heat removal and minimal core makeup are required (i.e., not transient-induced LOCA sequences), the CRD pumps can deliver high-pressure coolant to the RPV.

10. Depressurization via SRV or the automatic depressurization system (ADS). In the event that short-term DHR and core makeup are required and high-pressure systems have failed to provide adequate flow, the RPV can be depressurized to allow use of the low-pressure, high-capacity injection systems. If depressurization fails in this event, core damage is expected to occur. The ADS will automatically initiate on high drywell pressure and low-low reactor water level, and the availability of one train of the LPCI or LPCS systems, following a time delay. The SRVs can be opened by the operators to speed the depressurization process or to initiate it if ADS fails and if additional, operable valves are available.

11. LPCS. LPI can be provided by the LPCS system if required. The LPCS system performs the same functions as the LPCI system (described below) except that the coolant, which is drawn from the SP or the condensate storage tank (CST), is sprayed over the core.

12. LPCI. The LPCI system can provide short-term heat removal and cooling water makeup if the reactor has been depressurized to the operating range of the low-head RHR pumps. At Class C plants, LPCI is a mode of the RHR system; thus, the RHR pumps operate during LPCI. LPCI takes suction from the suppression pool (SP) or the CST and discharges into the recirculation loops or directly into the reactor vessel. If LPCI is successful in delivering sufficient flow to the reactor, long-term heat removal success is still required to mitigate core damage.

13. Residual heat removal shutdown cooling (SDC) mode. In this mode, the RHR system provides normal long-term DHR. Coolant is circulated from the reactor by the RHR pumps through the RHR heat exchangers and back to the reactor vessel. Long-term core cooling success requires that heat transfer to the environment commence within 24 h of the transient. RHR SDC success following successful reactor scram and high- or low-pressure injection of water to the RPV will prevent core damage.

14. RHR SP cooling mode. If RHR SDC is unavailable, the RHR pumps and heat exchangers can be aligned to take water from the SP, cool it via the RHR heat exchangers, and return it to the SP. This alignment can provide long-term cooling for transient mitigation.

15. RHR service water or other. This is a backup measure for providing water to the reactor to reflood the core and maintain core cooling if LPCI and LPCS are unavailable. Typically, the high-pressure SW pumps are aligned to the shell side of the RHR heat exchangers for delivery of water to one of the recirculation loops.

The event tree constructed for a BWR plant Class A nonspecific reactor trip is shown in Fig. A.16. The event tree is similar to that constructed for BWR Class C plants with the following exceptions: Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. The isolation condensers can provide long-term core cooling. Class A plants do not have LPCI systems, although they are equipped with LPCS; SP cooling is provided by a system independent of the SDC system. The event tree branches and sequences are discussed further below.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip similar to that described for BWR Class C plants. The following branches have functions and success requirements similar to those following a transient at BWRs associated with Class C.

2. Reactor shutdown.

3. Power conversion system.

4. SRV challenged and closed.

5. Isolation condensers and isolation condenser makeup. If PCS is not available and significant inventory has not been lost via the SRVs, then the IC system can provide for DHR and mitigate the transient. The IC system is an essentially passive system that condenses steam produced by the core, rejecting the heat to cooling water and returning the condensate to the reactor. Makeup is provided to the cooling water as needed. The system does not provide makeup to the reactor vessel.

6. FW or FWCI. Either FW or FWCI can provide short-term transient mitigation. When feedwater or FWCI is required and is successful, long-term DHR is required for complete transient mitigation. (PCS unavailability is assumed prior to feedwater or FWCI demand.) FWCI or feedwater is required for makeup in transient-induced LOCA sequences and for heat removal in sequences when the IC system would have mitigated the transient but was not available. FWCI is initiated automatically on low reactor level and uses the normal feedwater trains to deliver water to the reactor vessel.

7. CRD pumps.

8. Depressurization via SRV or ADS.

9. LPCS.

10. Fire water or other. Fire water or other raw water systems can provide a capability similar to that provided by the SW/RHR connection on Class C BWRs. As a backup source, if all normal core cooling is unavailable, fire water can be aligned to the LPCS injection line to provide water to the reactor vessel.

11. SDC. Like the RHR system at Class C plants, the SDC system is a closed-loop system that performs the long-term DHR function by circulating primary coolant from the reactor through the system's heat exchangers and back to the reactor vessel. Success requires the operation of at least one SDC loop. Long-term DHR is required to terminate transients in which high- or low-pressure injection is required to mitigate the transient.

12. Containment cooling. If the SDC system fails to provide long-term DHR, the CC system can remove decay heat. The system utilizes dedicated CC pumps, drawing suction from the SP, passing it through heat exchangers where heat is rejected to the SW system and then either returning it directly to the SP or spraying it into the dry well.

The event tree constructed for a BWR plant Class B nonspecific reactor trip is shown in Fig. A.19. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same except that Class B plants are equipped with HPCI systems instead of FWCI systems, and they are equipped with a LPCI system that represents an additional capability for providing LPCI. Also, at Class B BWRs, the CC system considered in the event tree utilizes the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR transient, shown on event trees applicable to each plant class, are described in Table A.10. Because of differences in the mitigation systems used in the three BWR classes, it is not possible to associate most sequences among different plant classes. Because of this, similar sequence numbers used for sequences in different plant classes do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss of Offsite Power

The event cores constructed define responses of BWRs to a LOOP in terms of sequences representing success and failure of plant systems. A LOOP condition will result in a generator load rejection that would trip the turbine control valves and initiate a reactor scram.

The event tree constructed for a LOOP at BWR Class C plants is shown in Fig. A.23. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (LOOP). The initiating event for a LOOP corresponds to any situation in which power from both the auxiliary and startup transformers is lost. This situation could result from grid disturbances or onsite faults.

2. Emergency power. Emergency power is provided by DGs at almost all plants. The DGs receive an initiation signal when an undervoltage condition is detected. Emergency power success requires the starting and loading of a sufficient number of DGs to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

3. Reactor shutdown. Given a load rejection, a scram signal is generated. Successful scram is the same as for the transient trees: a rapid insertion of control rods with no more than two adjacent control rods failing to insert. The scram can be automatically or manually initiated.

4. LOOP recovery (long-term). Success for this branch requires recovery of offsite power or diesel-backed ac power before the station batteries are depleted, typically 2 to 4 h.

5. SRV challenged and closed. If one or more SRV is challenged and fails to close, a transient-induced LOCA is initiated.

6. HPCI (or HPCS) or RCIC. Success requirements for these branches are identical to those following a transient at Class C BWRs. Either RCIC or HPCI (or HPCS) can provide the makeup and short-term core cooling required following most transients, including failure of the EPS. HPCI and RCIC only require dc power and sufficient steam to operate the pump turbines. HPCS systems utilize a motor-driven pump but are diesel-backed and utilize dedicated SW cooling.

7. CRD pumps. Given emergency power success, CRD pump success requirements following a LOOP are identical to those following a transient. The CRD pumps can provide sufficient makeup to remove decay heat but not enough makeup to mitigate a transient-induced LOCA. Manual restart of the CRD pumps is required following the LOOP.

8. Depressurization via SRV or the ADS.

9. LPCS, LPCI, or RHR service water.

10. RHR SDC mode or RHR SP cooling mode. For emergency power success sequences, the success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class C BWRs. Success for any one of these three branches can provide the long-term DHR required for transient mitigation. If emergency power fails, it must be recovered to power long-term DHR equipment. However, long-term DHR is not required until several hours (up to 24 h) into the transient.

The event tree constructed for a LOOP at BWR Class A plants is shown in Fig. A.17. The event tree is similar to that constructed for BWR Class C plants with the major exception that Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. However, given a LOOP, FWCI would be unavailable, because it is not backed by emergency power. Also, additional long-term core cooling is not required with IC success, as long as no transient-induced LOCA is initiated. In the emergency power failure sequences, the IC system is the only system that can provide core cooling because FWCI would be without power. The event-tree branches and sequences are further discussed below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for Class C BWRs. The following branches have functions and success requirements similar to those following a LOOP at BWRs associated with previously described BWR classes.

2. Emergency power.

3. Reactor shutdown.

4. LOOP recovery (long-term).

5. SRV challenged and closed.

6. IC. Following successful reactor scram, the IC system can provide enough DHR, in both the short and long term, to mitigate the transient if a transient-induced LOCA has not been initiated. The IC system cannot provide coolant makeup, which would be required in a transient-induced LOCA. The IC system is an essentially passive system that does not require ac power for success.

7. FWCI. The FWCI system can provide short-term core cooling and makeup for transient mitigation. However, FWCI success requires normal power supplies and cannot be powered by emergency power following a LOOP.

8. CRD pumps.

9. Depressurization via SRV or ADS.

10. LPCS, fire water, or other water source. Success requirements for these branches are similar to those following a nonspecific reactor trip at Class A BWRs. With interim high-pressure cooling unavailable, either LPCS or, as a last resort, fire water or another water source can be used to provide low-pressure water for core makeup and cooling.

11. SDC and containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs.

The event tree constructed for a BWR plant Class B LOOP is shown in Fig. A.20. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that Class B plants are equipped with HPCI systems instead of FWCI systems and are equipped with a LPCI system, which represents an additional capability for providing LPCI. At Class B BWRs the CC system utilizes the LPCI pumps rather than having its own dedicated pumps. In emergency power failure sequences, either the IC or HPCI system can provide the required core cooling for short-term transient mitigation. However, if an SRV sticks open (transient-induced LOCA), the ICs cannot provide the makeup needed, and HPCI is required. The ICs can also provide long-term cooling, but when only HPCI is operable, recovery of emergency power is necessary to power SDC-related loads.

Sequences resulting in core damage following a BWR LOOP, as shown on each plant-class event tree, are described in Table A.11. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss-of-Coolant Accident

The event trees constructed define the response of BWRs to a small LOCA in terms of sequences representing success and failure of plant systems. The LOCA chosen for consideration is a small LOCA, one that would require a reactor scram and continued operation of HPI systems. A large LOCA would require operation of the high-volume/low-pressure systems and is not addressed in the models.

The LOCA event tree constructed for BWR Class C plants is shown in Fig. A.24. The event-tree branches and sequences leading to core damage and core vulnerability follow.

1. Initiating event (small LOCA). Any breach in the RCS on the reactor side of the MSIVs that results in coolant loss in excess of the capacity of the CRD pumps is considered a LOCA. A small LOCA is considered to be one in which losses are not great enough to reduce the system pressure to the operating range of the LPI systems.

2. Reactor shutdown. Successful scram is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition.

3. HPCI or HPCS. HPCI (or HPCS, depending on the plant) can provide the required inventory makeup.

4. Depressurization via SRV or ADS. The success requirements for this branch are similar to those following a nonspecific reactor trip transient. SRV/ADS success allows the use of low-pressure systems to provide short-term core cooling and makeup.

5. LPCS, LPCI, or RHR service water. The success requirements for these branches are similar to those following a nonspecific reactor trip transient. Any one of these branches can provide short-term core cooling and makeup if SRV/ADS is successful.

6. RHR (SDC mode) or RHR (SP cooling mode). Success requirements for these branches are similar to those following a nonspecific reactor trip transient, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the break size. These methods each have the capability of providing long-term DHR. Long-term DHR is required in all sequences for LOCA mitigation.

The LOCA event tree constructed for BWR Class A plants is shown in Fig. A.18. The event tree is similar to the LOCA tree constructed for BWR Class C plants except that Class A plants have FWCI instead of HPCI or HPCS systems and are, in general, not equipped with LPCI systems (only LPCS systems). In addition, SP and CC systems are independent of the SDC system. The event tree branches and sequences leading to core damage follow.

1. Initiating event (small LOCA). The initiating event is a small LOCA similar to that described for BWR Class C plants. The following branches have functions and success requirements similar to those following a small LOCA at BWRs associated with the previously described BWR classes.

2. Reactor shutdown.

3. FWCI. The FWCI system has the capability to keep the core covered and provide interim core cooling. FWCI initiates automatically on low reactor water level.

4. Depressurization via SRV or ADS.

5. LPCS or fire water (or other water source). The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs. Either of these systems (branches) can provide LPI for makeup and short-term core cooling if high-pressure systems are unavailable.

6. SDC or containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the size of the break. Either of these methods can provide the long-term DHR required to mitigate a small LOCA.

The LOCA event tree constructed for BWR Class B plants is shown in Fig. A.21. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that some Class B plants are equipped with HPCI systems instead of FWCI systems and Class B BWRs have a LPCI system, which provides an additional capability for LPCI. At Class B BWRs the CC system uses the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR small-break LOCA, as shown on each plant-class event tree, are described in Table A.12. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

Alternate Recovery Actions

The BWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection against core damage are unavailable. If feedwater, HPCI, and RCIC are unavailable (FWCI and ICs on BWR Classes A and B) and cannot be recovered in the short term, the use of the CRD pumps (provided no LOCA exists) and the use of ADS (to depressurize below the operating pressure of low-pressure systems) are modeled. In addition, the potential for short-term recovery of a faulted system is also included in the appropriate branch model.

Alternate equipment and procedures, beyond the systems and functions included in the event tree, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental diesel generators, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.

- The use of RCIC to provide RPV makeup for a single stuck-open relief valve. Thermal-hydraulic analyses performed to support a number of BWR probabilistic risk assessments have demonstrated the viability of RCIC for this purpose.

- The use of the condensate system for LPI. This recovery action requires that the condensate system be available (even though PCS and feedwater are unavailable) and that the plant has been depressurized.

- The use of containment venting for long-term DHR, provided an injection source is available. This core cooling method has been addressed in some PRAs.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.4 Branch Probability Estimates

Branch probability estimates used in the 1988-1992 precursor calculations were developed using information in the 1984-86 precursors when possible. Probability values developed from precursor information are shown in Table A.13. The process used to estimate branch probability values used in the precursor calculations is described in detail in Appendix C to Ref. 5 and in Ref. 6.

In addition to system failures caused by equipment failures, the likelihood of failing to actuate manually actuated systems was also included in the models. Examples of such systems are the DHR system in BWRs and feed and bleed in PWRs. For actions in the control room, revised failure to initiate probabilities consistent with those utilized for 1987 precursor calculations were also used for 1988-1992 calculations. These revised values typically assume a failure probability of 0.001 for an unburdened action and 0.01 for a burdened action. The failure probability for subsequent actions is assumed to be higher. Operator action failure probabilities used in the 1988-1992 calculations are shown in Table A.14.

A.5 Reference Event Calculations

Conditional core damage probability estimates were also calculated for nonspecific reactor trip, LOFW, and unavailabilities in certain single-train BWR systems (HPCI, HPCS, RCIC, and CRD cooling). These calculations indicate the relative importance of these events, which are too numerous to warrant individual calculation. The results of these calculations, performed without consideration of alternate recovery actions that were addressed in certain 1992 precursor assessments, are listed in Table A.15.

Table A.15 shows that nonspecific reactor trips without additional observed failures have conditional core damage probabilities below 5×10^{-6} per trip, depending on plant class. The likelihood of LOFW in conjunction with a trip is included in these calculations. LOFW conditional core damage probabilities are less than 4×10^{-5} per LOFW event, again depending on plant class, except for BWR Class A plants (1.7×10^{-4}). The conditional core damage probabilities associated with unavailabilities of HPCI and HPCS (single-train BWR systems) are also above 10^{-5}, assuming a one-half month unavailability.

A.6 References

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5. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents; 1986, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 5 and 6), May 1988.
6. J. W. Minarick, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp., *Revised LOOP Recovery and PWR Seal LOCA Models*, Technical Letter Report ORNL/NRC/LTR-89/11, August 1989.*

*Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

Table A.1 Branch probability estimation process

Branch failure	Observed operational event	Non-recovery likelihood for event	Effective number of non-recoverable events	Observation period	Probability estimate
Steam generator isolation	Steam line pressure transmitters (9 of 12) were found in faulty alignment, which would have prevented automatic steam line isolation on demand at Maine Yankee (LER 309/85-009, 8/7/85)	0.04	1.04	12 demands per reactor year due to testing in 164 PWR reactor years (1984 – 86 observation period) results in 1968 demands	5.3×10^{-4}
	All MSIVs failed to close prior to entering refueling at Point Beach 2 (LER 301/86-004, 9/28/86)	1.0			

Table A.2 Rules for calculating precursor significance

1. Event sequences requiring calculation.

   If an initiating event occurs as part of a precursor (i.e., the precursor consists of an initiating event plus possible additional failures), then use the event tree associated with that initiator; otherwise, use all event trees impacted by the observed unavailability.

2. Initiating event probability.

   If an initiating event occurs as part of a precursor, then the initiator probability used in the calculation is the probability of failing to recover from the observed initiating event (i.e., the numeric value of the recovery class for the event).

   If an initiating event does not occur as part of a precursor, then the probability used for the initiating event is developed using the initiating event frequency and event duration. Event durations (the period of time during which the failure existed) are based on information included in the event report, if provided. If the event is discovered during testing, then one-half of the test period (15 days for a typical 30-day test interval) is assumed, unless a specific failure duration is identified.

3. Branch probability estimation.

   For event tree branches for which no failed or degraded condition is observed, a probability equal to the estimated branch failure probability is assigned.

   For event tree branches associated with a failed system, a probability equal to the numeric value associated with the recovery class is assigned.

   For event tree branches that include a degraded system (i.e., a system that still meets minimum operability requirements but with reduced or no redundancy), the estimated failure probability is modified to reflect the loss of redundancy.

4. Support system unavailabilities.

   Systems or trains rendered unavailable as a result of support system failures are modeled recognizing that, as long as the affected support system remains failed, all impacted systems (or trains) are unavailable; but if the support system is recovered, all the affected systems are recovered. This can be modeled through multiple calculations that address support system failure and success. Calculated core damage probabilities for each case are normalized based on the likelihood of recovering the support system. (Support systems, except emergency power, are not directly modeled in the current ASP models.)

Table A.3 ASP reactor plant classes

Plant name	Plant class	Plant name	Plant class
ANO-Unit1	PWR Class D	Millstone 3	PWR Class A
ANO-Unit	PWR Class G	Monticello	BWR Class C
Beaver Valley 1	PWR Class A	Nine Mile Point 1	BWR Class A
Beaver Valley 2	PWR Class A	Nine Mile Point 2	BWR Class C
Big Rock Point	BWR Class A	North Anna 1	PWR Class A
Browns Ferry 1	BWR Class C	North Anna 2	PWR Class A
Browns Ferry 2	BWR Class C	Oconee 1	PWR Class D
Browns Ferry 3	BWR Class C	Oconee 2	PWR Class D
Braidwood 1	PWR Class B	Oconee 3	PWR Class D
Braidwood 2	PWR Blass B	Oyster Creek	BWR Class A
Brunswick 1	BWR Class C	Palisades	PWR Class G
Brunswick 2	BWR Class C	Palo Verde 1	PWR Class H
Byron 1	PWR Class B	Palo Verde 2	PWR Class H
Byron 2	PWR Class B	Palo Verde 3	PWR Class H
Callaway 1	PWR Class B	Peach Bottom 2	BWR Class C
Calvert Cliffs 1	PWR Class G	Peach Bottom 3	BWR Class C
Calvert Cliffs 2	PWR Class G	Perry 1	BWR Class C
Catawba 1	PWR Class B	Pilgrim 1	BWR Class C
Catawba 2	PWR Class B	Point Beach 1	PWR Class B
Clinton 1	BWR Class C	Point Beach 2	PWR Class B
Comanche Peak 1	PWR Class B	Prairie Island 1	PWR Class B
Comanche Peak 2	PWR Class B	Prairie Island 2	PWR Class B
Cook 1	PWR Class B	Quad Cities 1	BWR Class C
Cook 2	PWR Class B	Quad Cities 2	BWR Class C
Cooper Station	BWR Class C	Rancho Seco	PWR Class D
Crystal River 3	PWR Class D	River Bend 1	BWR Class C
Davis-Besse	PWR Class B	Robinson 2	PWR Class B
Diablo Canyon 1	PWR Class B	Salem 1	PWR Class B
Diablo Canyon 2	PWR Class B	Salem 2	PWR Class B
Dresden 2	BWR Class B	San Onofre 1	Unique
Dresden 3	BWR Class B	San Onofre 2	PWR Class H
Duane Arnold	BWR Class C	San Onofre 3	PWR Class H
Farley 1	PWR Class B	Seabrook 1	PWR Class B
Farley 2	PWR Class B	Sequoyah 1	PWR Class B
Fermi 2	BWR Class C	Sequoyah 2	PWR Class B
Fitzpatrick	BWR Class C	South Texas 1	PWR Class B
Fort Calhoun	PWR Class G	South Texas 2	PWR Class B
Ginna	PWR Class B	St. Lucie 1	PWR Class G
Grand Gulf 1	BWR Class C	St. Lucie 2	PWR Class G
Haddam Neck	PWR Class B	Summer 1	PWR Class B
Harris 1	PWR Class B	Surry 1	PWR Class A
Hatch 1	BWR Class C	Surry 2	PWR Class A
Hatch 2	BWR Class C	Susquehanna 1	BWR Class C
Hope Creek 1	BWR Class C	Susquehanna 2	BWR Class C
Indian Point 2	PWR Class B	Three Mile Island 1	PWR Class D
Indian Point 3	PWR Class B	Trojan	PWR Class B
Kewaunee	PWR Class B	Turkey Point 3	PWR Class B
LaCrosse	Unique	Turkey Point 4	PWR Class B
LaSalle 1	BWR Class C	Vermont Yankee	BWR Class C
LaSalle 2	BWR Class C	Vogtle 1	PWR Class B
Limerick 1	BWR Class C	Vogtle 2	PWR Class B
Limerick 2	BWR Class C	WNPSS 2	BWR Class C
Maine Yankee	PWR Class B	Waterford 3	PWR Class H
McGuire 1	PWR Class B	Wolf Creek 1	PWR Class B
McGuire 2	PWR Class B	Yankee Rowe	PWR Class B
Millstone 1	BWR Class A	Zion 1	PWR Class B
Millstone 2	PWR Class G	Zion 2	PWR Class B

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
11	Core damage	Unavailability of HPR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI. (PWR Classes A, B, D, G, and H)
12	Core damage	Unavailability of HPI following successful trip and AFW initiation, primary relief valve lift, and primary relief valve failure to reseat. (PWR Classes A, B, D, G, and H)
13	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
14	Core damage	Similar to sequence 12, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
15	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but the PORV fails to open. (PWR Classes A, B, and G)
16	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but fails in the recirculation phase. (PWR Classes A, B, D, and G)
17	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed fails in the injection phase. (PWR Classes A, B, D, and G)
18	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models. (PWR Classes A, B, D, G, and H)
19	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful but CSR is unavailable. (PWR Class G)
20	Core damage	Unavailability of CSR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI and HPR. (PWR Class A)
21	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Class A)
22	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful, but CSR is unavailable for containment heat removal. This sequence is distinguished from sequence 19 because of differences in the function of CSR on Class A and G plants. (PWR Class A)

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
23	Core damage	Unavailability of AFW and MFW following successful trip. The SGs are successfully depressurized, but the condensate pumps fail to provide SG cooling. (PWR Class H)
24	Core damage	Unavailability of AFW and MFW following successful trip, plus failure to depressurize the SGs to allow for the use of the condensate pumps for SG cooling. (PWR Class H)
25	Core damage	Unavailability of AFW and MFW following successful trip. At least one open SRV fails to reseat, but HPI and HPR are successful. SG depressurization is successful, but the condensate pumps fail to provide SG cooling. (PWR Class H)
26	Core damage	Similar to sequence 25 except that SG depressurization fails. (PWR Class H)
27	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat. HPI is initiated but HPR fails. (PWR Class H)
28	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat and HPI fails. (PWR Class H)

Table A.5 PWR transient sequences summary

Seq. No.	End State	RT	AFW	MFW	RV Chall	RV Reseat	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	B	D	G	H
11	CD	S	S		S*	F	S	F					x	x	x	x	x
12	CD	S	S		S*	F	F						x	x	x	x	x
13	CD	S	F	S	S*	F	S	F					x	x	x	x	x
14	CD	S	F	S	S*	F	F						x	x	x	x	x
15	CD	S	F	F			S	S	F				x	x		x	
16	CD	S	F	F			S	F					x	x	x	x	
17	CD	S	F	F			F						x	x	x	x	
18	ATWS	F											x	x	x	x	x
19	CD	S	F	F			S	S	S	F						x	
20	CD	S	S		S*	F	S	S		F			x				
21	CD	S	F	S	S*	F	S	S		F			x				
22	CD	S	F	F			S	S	S	F			x				
23	CD	S	F	F		S					S	F					x
24	CD	S	F	F		S					F						x
25	CD	S	F	F		F	S	S			S	F					x
26	CD	S	F	F		F	S	S			F						x
27	CD	S	F	F		F	S	F									x
28	CD	S	F	F		F	F										x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
40	ATWS	Failure to trip following a LOOP. (PWR Classes A, B, D, G, and H)
41	Core damage	Unavailability of HPR following a LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat; and successful HPI. (PWR Classes A, B, D, G, and H)
42	Core damage	Unavailability of HPI following LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat. (PWR Classes A, B, D, G, and H)
43	Core damage	Failure of the PORV to open for feed and bleed cooling following successful trip and emergency power, and AFW failure. (PWR Classes A, B, and G)
44	Core damage	Failure of HPR for recirculation cooling following feed and bleed initiation. Trip and emergency power are successful, but AFW fails. (PWR Classes A, B, D, and G)
45	Core damage	Unavailability of HPI for feed and bleed cooling following successful trip and emergency power and AFW failure. (PWR Classes A, B, D, and G)
46	Core damage	Unavailability of HPR following HPI success for RCP seal LOCA mitigation. AC power is recovered following successful trip, emergency power failure, turbine-driven AFW train(s) success, primary relief valve lift and reseat, and a subsequent seal LOCA. (PWR Classes A, B, D, G, and H)
47	Core damage	This sequence is similar to sequence 46 except that HPI fails for RCP seal LOCA mitigation. (PWR Classes A, B, D, G, and H)
48	Core damage	Failure to recover AC power following an RCP seal LOCA. The seal LOCA occurs following successful trip, failure of emergency power, turbine-driven AFW train(s) success, and primary relief valve lift and closure. (PWR Classes A, B, D, G, and H)
49	Core damage	Failure to recover AC power following successful trip and emergency power system failure, AFW turbine train(s) success, and primary relief valve lift and reseat. No RCP seal LOCA occurs in the sequence. (PWR Classes A, B, D, G, and H)
50	Core damage	Failure of a primary relief valve to reseat following lift subsequent to a successful trip, emergency power system failure, and AFW turbine trains(s) success. (PWR Classes A, B, D, G, and H)

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
51	Core damage	This sequence is similar to sequence 46 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
52	Core damage	This sequence is similar to sequence 47 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
53	Core damage	This sequence is similar to sequence 48 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
54	Core damage	This sequence is similar to sequence 49 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
55	Core damage	Failure of AFW following successful trip and emergency power system failure (PWR Classes A, B, D, G, and H)
56	Core damage	Failure of CSR in conjunction with successful feed and bleed following trip, emergency power system success, and AFW failure (PWR Class G)
57	Core damage	Failure of CSR following LOOP with successful trip, emergency power and AFW, primary relief valve challenge and failure to reseat, and successful HPI and HPR. (PWR Class A)
58	Core damage	Failure of CSR in conjunction with successful feed and bleed following LOOP with successful trip and emergency power initiation, and AFW failure. (PWR Class A)
59	Core damage	Failure of CSR following successful HPI and HPR required to mitigate a seal LOCA. This sequence involves a LOOP with successful trip, emergency power system failure, primary relief valve challenge and reseat, and a subsequent seal LOCA with AC power recovery prior to core uncovery. (PWR Class A)
60	Core damage	This sequence is similar to sequence 59 except that the primary relief valves are not challenged. (PWR Class A)
61	Core damage	Failure of AFW following a LOOP with successful trip and emergency power. (PWR Class H)

Table A.7 PWR LOOP sequences summary

Seq. No.	End State	RT/ LOOP	EP	AFW	RV Chall	RV Reseat	Seal LOCA	EP Recov	HPI	HPR	PORV Open	CSR	PWR Class A	PWR Class B	PWR Class D	PWR Class G	PWR Class H
40	ATWS	F											x	x	x	x	x
41	CD	S	S	S	S^*	F			S	F			x	x	x	x	x
42	CD	S	S	S	S^*	F			F				x	x	x	x	x
43	CD	S	S	F					S	S	F		x	x		x	
44	CD	S	S	F					S	F			x	x	x	x	
45	CD	S	S	F					F				x	x	x	x	
46	CD	S	F	S	S^*	S	S^*	S	S	F			x	x	x	x	x
47	CD	S	F	S	S^*	S	S^*	S	F				x	x	x	x	x
48	CD	S	F	S	S^*	S	S^*	F					x	x	x	x	x
49	CD	S	F	S	S^*	S		F					x	x	x	x	x
50	CD	S	F	S	S^*	F							x	x	x	x	x
51	CD	S	F	S			S^*	S	S	F			x	x	x	x	x
52	CD	S	F	S			S^*	S	F				x	x	x	x	x
53	CD	S	F	S			S^*	F					x	x	x	x	x
54	CD	S	F	S				F					x	x	x	x	x
55	CD	S	F	F									x	x	x	x	x
56	CD	S	S	F					S	S	S	F				x	
57	CD	S	S	S	S^*	F			S	S		F	x				
58	CD	S	S	F					S	S	S	F	x				
59	CD	S	F	S	S^*	S	S^*	S	S	S		F	x				
60	CD	S	F	S			S^*	S	S	S		F	x				
61	CD	S	S	F													x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S^* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.8 PWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
71	Core damage	Unavailability of HPR following a small-break LOCA with trip, AFW and HPI success. (PWR Classes A, B, D, G, and H)
72	Core damage	Unavailability of HPI following a small-break LOCA with trip and AFW success. (PWR Classes A, B, D, G, and H)
73	Core damage	This sequence is similar to sequence 71 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
74	Core damage	This sequence is similar to sequence 72 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
75	Core damage	Unavailability of AFW and MFW following a small-break LOCA and successful trip. The PORV is unavailable to depressurize the RCS to the HPI pump discharge pressure. (PWR Classes A, B, and G)
76	Core damage	Unavailability of AFW and MFW following a small-break LOCA with trip success. HPI is successful but HPR fails. (PWR Classes A, B, D, G, and H)
77	Core damage	Unavailability of AFW and MFW following trip success. HPI fails to provide RCS makeup. (PWR Classes A, B, D, G, and H)
78	ATWS	Failure of reactor trip following a small-break LOCA. (PWR Classes A, B, D, G, and H)
79	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success, AFW and MFW failure, and feed and bleed success. (PWR Class G)
80	Core damage	Unavailability of CSR following a small-break LOCA with trip, AFW, HPI and HPR success. (PWR Class A)
81	Core damage	This sequence is similar to sequence 80 except that MFW is used for SG cooling in the event AFW is unavailable. (PWR Class A)
82	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success, AFW and MFW unavailability, and feed and bleed success. (PWR Class A)
83	Core damage	Unavailability of the condensate pumps for SG cooling following a small-break LOCA with trip success, unavailability of AFW and MFW, and successful SG depressurization. (PWR Class H)
84	Core damage	This sequence is similar to sequence 83 except that SG depressurization is unavailable. (PWR Class H)

Table A.9 PWR small-break LOCA sequences summary

Seq. No.	End State	RT	AFW	MFW	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	PWR Class B	PWR Class D	PWR Class G	PWR Class H
71	CD	S	S		S	F					x	x	x	x	x
72	CD	S	S		F						x	x	x	x	x
73	CD	S	F	S	S	F					x	x	x	x	x
74	CD	S	F	S	F						x	x	x	x	x
75	CD	S	F	F	S	S	F				x	x		x	
76	CD	S	F	F	S	F					x	x	x	x	x
77	CD	S	F	F	F						x	x	x	x	x
78	ATWS	F									x	x	x	x	x
79	CD	S	F	F	S	S	S	F						x	
80	CD	S	S		S	S		F			x				
81	CD	S	F	S	S	S		F			x				
82	CD	S	F	F	S	S	S	F			x				
83	CD	S	F	F	S	S			S	F					x
84	CD	S	F	F	S	S			F						x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of isolation condenser, and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater and successful feedwater coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and feedwater coolant injection, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, feedwater coolant injection and control rod drive cooling; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and success of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
16	Core damage	Similar to Sequence 15 except the shutdown cooling system fails followed by successful containment cooling.
17	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and failure of low-pressure core spray.
18	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
19	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
20	Core damage	Similar to Sequence 19 except unsuccessful main feedwater followed by successful feedwater coolant injection.
21	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief challenge and unsuccessful reseat, unsuccessful main feedwater and followed by successful vessel depressurization and low-pressure core spray.
22	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray.
23	Core damage	Similar to Sequence 22 except failure of the shutdown cooling system and successful containment spray.
24	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, unsuccessful main feedwater and feedwater coolant injection, successful vessel depressurization, and unsuccessful low-pressure core spray.
25	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and feedwater coolant injection.
26	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
27	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
28	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
30	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and failure of isolation condenser and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater followed by successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization, and failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
16	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; and failure of isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.
17	Core damage	Similar to Sequence 16 except the shutdown cooling system fails followed by successful containment cooling mode of the low-pressure coolant injection system.
18	Core damage	Similar to Sequence 15 except low-pressure coolant injection system fails.
19	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling.
20	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
21	Core damage	Similar to Sequence 20 except unsuccessful main feedwater followed by successful high-pressure coolant injection.
22	Core damage	Similar to Sequence 20 except unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray.
23	Core damage	Similar to Sequence 20 except failure of main feedwater and high-pressure coolant injection, followed by successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.
24	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
25	Core damage	Similar to Sequence 24 except failure of the shutdown cooling system and successful containment spray mode of low-pressure core injection.
26	Core damage	Similar to Sequence 23 except unsuccessful low-pressure coolant injection.
27	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection.
28	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
11	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and successful main feedwater.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
12	Core damage	Similar to Sequence 11 except failure of main feedwater with successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, with successful reactor core isolation cooling.
14	Core damage	Similar to Sequence 11 except failure of main feedwater, high-pressure coolant injection, and reactor core isolation cooling, with successful control rod drive cooling.
15	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
16	Core damage	Similar to Sequence 15 except failure of low-pressure core spray and successful low-pressure coolant injection.
17	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
18	Core damage	Similar to Sequence 17 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
19	Core damage	Similar to Sequence 16 except failure of low-pressure coolant injection.
20	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat. Failure of the main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
21	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, and successful main feedwater.
22	Core damage	Similar to Sequence 21 except unsuccessful main feedwater with successful high-pressure coolant injection.
23	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray
24	Core damage	Similar to Sequence 23 except failure of low-pressure core spray and successful low-pressure coolant injection.
25	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
26	Core damage	Similar to Sequence 25 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
27	Core damage	Similar to Sequence 24 except failure of low-pressure coolant injection.
28	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection systems.
29	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
32	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
37	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
38	Core damage	Similar to Sequence 20 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power reactor scram, safety relief valve challenge and reseat. Failure of isolation condenser and successful feedwater coolant injection.
42	Core damage	Similar to Sequence 41 except failure of the feedwater coolant injection and successful control rod drive cooling.
43	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the feedwater coolant injection and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
45	Core damage	Similar to Sequence 44 except failure of the shutdown cooling system and successful containment spray.
46	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of feedwater coolant injection and control rod drive cooling, with successful vessel depressurization and failure of the low-pressure core spray.
47	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, feedwater coolant injection, and control rod drive cooling.
48	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful feedwater coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
49	Core damage	Similar to Sequence 48 except failure of feedwater coolant injection followed by successful vessel depressurization and low-pressure core spray.
50	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of feedwater coolant injection. Successful vessel depressurization, failure of low-pressure core spray, and successful shutdown cooling system.
51	Core damage	Similar to Sequence 50 except failure of shutdown cooling system and successful containment cooling.
52	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of feedwater coolant injection, successful vessel depressurization, and failure of low-pressure core spray.
53	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the feedwater coolant injection system.
54	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
55	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
56	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged..
60	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged
61	Core damage	Unavailability of the isolation condenser following a loss of offsite power, failure of emergency power, successful scram, and safety relief valve challenge and successful reseat.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
62	Core damage	Failure of an SRV to reseat following challenge after a loss of offsite power with failure of emergency power and successful reactor scram.
63	Core damage	Similar to Sequence 61 except the safety relief valves are not challenged.
64	Core damage	Failure of recovery of electric power in the long-term following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser and successful high-pressure coolant injection.
42	Core damage	Similar to Sequence 41 except failure of high-pressure coolant injection and successful control rod drive cooling.
43	Core damage	Similar to Sequence 41 except failure of the high-pressure coolant injection and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the high-pressure coolant injection and control rod drive cooling systems, with successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
45	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, high-pressure coolant injection, and control rod drive cooling. Successful vessel depressurization, failure of low-pressure core spray, and low-pressure coolant injection with successful shutdown cooling.
46	Core damage	Similar to Sequence 45 except failure of the shutdown cooling system and successful containment spray mode low-pressure coolant injection.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram, challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, high-pressure coolant injection, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray.
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.
52	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure core injection, and successful shutdown cooling system.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
53	Core damage	Similar to Sequence 52 except failure of shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
54	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of high-pressure coolant injection, successful vessel depressurization and failure of low-pressure core spray and low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.
64	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, failed isolation condenser, and successful high-pressure coolant injection.
65	Core damage	Unavailability of high-pressure core injection following a loss of offsite power, failure of emergency power, successful reactor scram, safety relief valve challenge and reseat, and failed isolation condenser and high-pressure coolant injection systems.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
66	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and successful high-pressure coolant injection.
67	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and failure of high-pressure coolant injection.
68	Core damage	Similar to Sequence 64 except the safety relief valves are not challenged.
69	Core damage	Similar to Sequence 65 except the safety relief valves are not challenged.
84	Core damage	Failure of long-term recovery of electric power following a loss of offsite power, with failure of emergency power and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
40	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
41	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection system and successful reactor core isolation cooling.
42	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection and reactor core isolation cooling systems with successful control rod drive cooling

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
43	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat; failure of the high-pressure coolant injection, reactor core isolation cooling and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Similar to Sequence 43 except failure of low-pressure core spray and successful low-pressure coolant injection.
45	Core damage	Unavailability of fire water or other equivalent water source for reactor makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, and failure of low-pressure core spray and low-pressure coolant injection with successful residual heat removal in shutdown cooling mode.
46	Core damage	Similar to Sequence 45 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
52	Core damage	Unavailability of fire water or other equivalent water source following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
53	Core damage	Similar to Sequence 52 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
54	Core damage	Similar to Sequence 51 except failure of low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 40 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
64	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
65	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
66	Core damage	Similar to Sequence 65 except high-pressure coolant injection fails with successful reactor core isolation cooling.
67	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, with failures of high-pressure coolant injection and reactor core isolation cooling.
68	Core damage	Similar to Sequence 65 except the safety relief valves fail to reseat.
69	Core damage	Failure of high-pressure coolant injection following a loss of offsite power, with emergency power failure, successful reactor scram, safety relief valve challenge, and unsuccessful reseat.
80	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, and long-term recovery of electric power. The safety relief valves are not challenged, and high-pressure coolant injection is successful.
81	Core damage	Similar to Sequence 66 except the safety relief valves are not challenged.
82	Core damage	Similar to Sequence 67 except the safety relief valves are not challenged.
83	Core damage	Unable to recover long-term electric power following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, and successful feedwater coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, failure of feedwater coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray, and successful shutdown cooling system.
74	Core damage	Similar to Sequence 73 except failure of the shutdown cooling system and successful containment cooling.
75	Core damage	Similar to Sequence 72 except failure of the low-pressure core spray.
76	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the feedwater coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, failure of high-pressure coolant injection, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
75	Core damage	Similar to Sequence 74 except failure of the shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
71	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, failure of the high-pressure coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray, and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
75	Core damage	Similar to Sequence 74 except failure of the residual heat removal system in the shutdown cooling mode and success in the suppression pool cooling mode.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.

Table A.13 Average initiating event frequency and branch failure probability estimates developed from 1984-1986 precursors.

Initiator/branch	Initial estimate (no recovery attempted)	Nonrecovery estimate	Total
	PWRs		
LOOP	4.1×10^{-2}/year	0.39	1.6×10^{-2}/year[a]
Small-break LOCA	1.5×10^{-2}/year	0.43	6.4×10^{-3}/year
Auxiliary feedwater	3.8×10^{-4}	0.26	9.9×10^{-5}
High-pressure injection	6.1×10^{-4}	0.84	5.1×10^{-4}
Long-term core cooling (high-pressure recirculation)	1.5×10^{-4}	1.00	1.5×10^{-4}
Emergency power	6.4×10^{-4}	0.78	5.0×10^{-4}
SG isolation (MSIVs)	8.3×10^{-4}	0.64	5.3×10^{-4}
	BWRs		
LOOP	1.0×10^{-1}/year	0.32	3.3×10^{-2}/year[a]
Small-break LOCA	2.0×10^{-2}/year	0.50	1.0×10^{-2}/year
HPCI/RCIC	1.7×10^{-3}	0.49	8.4×10^{-4}
RV isolation	1.7×10^{-3}	1.00	1.7×10^{-3}
LPCI	1.0×10^{-3}	0.71	7.4×10^{-4}
Emergency power	1.0×10^{-4}	0.85	8.9×10^{-5}
Automatic depressurization	3.7×10^{-3}	0.71	2.6×10^{-3}

[a]Precursor calculations utilize plant-specific LOOP frequency estimates developed from information in P.W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

Table A.14 Operator action failure probabilities

Operation action	Failure probability
BWRs	
Condensate/feedwater recovery	0.001
Containment venting	0.01
Control rod drive water use	0.01
Initiation of RHR service water, fire water	0.01
Shutdown cooling	0.001
Standby liquid control initiation	0.01
PWRs	
Condensate/MFW recovery	0.01
Containment spray recirculation	0.001
Emergency core cooling recirculation	0.001
Fail to block stuck-open PORVs	0.001
Open PORVs for feed and bleed	0.0004
SG depressurization	0.001
Use feed and bleed to cool core	0.01

Table A.15 Reference event conditional probability values

Postulated operational event	Conditional core damage probability
BWR Class A nonspecific reactor trip	2.8×10^{-6}
BWR Class A LOFW	1.7×10^{-4}
BWR Class B nonspecific reactor trip	7.7×10^{-8}
BWR Class B LOFW	4.3×10^{-6}
BWR Class C (turbine-driven feed pumps) nonspecific reactor trip	1.2×10^{-6}
BWR Class C (turbine-driven feed pumps) LOFW	1.5×10^{-5}
PWR Class A nonspecific reactor trip	1.8×10^{-7}
PWR Class A LOFW	2.4×10^{-6}
PWR Class B nonspecific reactor trip	1.8×10^{-7}
PWR Class B LOFW	2.2×10^{-6}
PWR Class D nonspecific reactor trip	4.7×10^{-7}
PWR Class D LOFW	6.8×10^{-6}
PWR Class G nonspecific reactor trip	1.8×10^{-7}
PWR Class G LOFW	2.4×10^{-6}
PWR Class H nonspecific reactor trip	4.9×10^{-6}
PWR Class H LOFW	3.9×10^{-5}
BWR Class C HPCI unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.0×10^{-5}
BWR Class C HPCS unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.4×10^{-5}
BWR Class C RCIC unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	3.8×10^{-8}
BWR Class C CRD cooling unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	6.2×10^{-8}

[a]The probability of a transient, LOOP, or small-break LOCA during the 360-h unavailability was estimated as described in Sect. A.1.

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	PWR event trees
AFW	auxiliary feedwater fails
ATWS	anticipated transient without scram end state
COND	condensate system fails
CD	core damage end state
CSR	containment spray recirculation fails
EP	emergency power fails
EP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
HPI	high-pressure injection fails
HPR	high-pressure recirculation fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
MFW	main feedwater fails
PORV OPEN	power-operated relief valve fails to open for feed and bleed cooling
PORV/SRV CHALL	power-operated relief valve or safety relief valves challenged (challenge rate)
PORV/SRV RESEAT	power-operated relief valve and/or safety relief valve fails to reseat
RT	reactor trip fails
RT/LOOP	reactor trip fails given a loss of offsite power
SEAL LOCA	RCP seal LOCA occurs
SEC SIDE DEP	secondary-side depressurization fails
SEQ NO	sequence number
SRV CHALL	safety relief valves challenged
SRV RESEAT	safety relief valve fails to reseat
TRANS	nonspecific reactor-trip transient

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	BWR Event Trees
CC	containment cooling fails
CRD	control-rod-drive cooling fails
EP	emergency power fails
FIREWTR or OTHER	fire water or other equivalent water source fails
FW	unavailabilty of main feedwater
FWCI	failure of feedwater coolant injection system
HPCI OR HPCS	high-pressure coolant injection or high-pressure core spray fails
IC/IP MUP	isolation condenser or isolation condenser makeup fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
LOOP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
LPCI	low-pressure coolant injection fails
LPCI (CC MODE)	containment cooling mode of low-pressure coolant injection system fails
LPCI (RHR)	residual heat removal mode of low-pressure coolant injection core spray fails
LPCS	low-pressure core spray fails
PCS	failure of continued power conversion system operation
RCIC	reactor core isolation cooling fails
RHR (SDC MODE)	residual-heat-removal shutdown cooling mode fails
RHR (SP COOLING MODE)	residual-heat-removal suppression pool cooling mode fails
RHR SW or OTHER	residual-heat-removal service water or other water source fails
RX SHUTDOWN	reactor fails to scram
SDC	shutdown cooling system fails
SRVs/ADS	safety relief valve(s) fail to open for depressurization or automatic depressurization system fails
SRV CHAL	safety relief valve(s) challenged (challenge rate)
SRV-C	safety relief valve fails to close
TRANSIENT	nonspecific reactor-trip transient

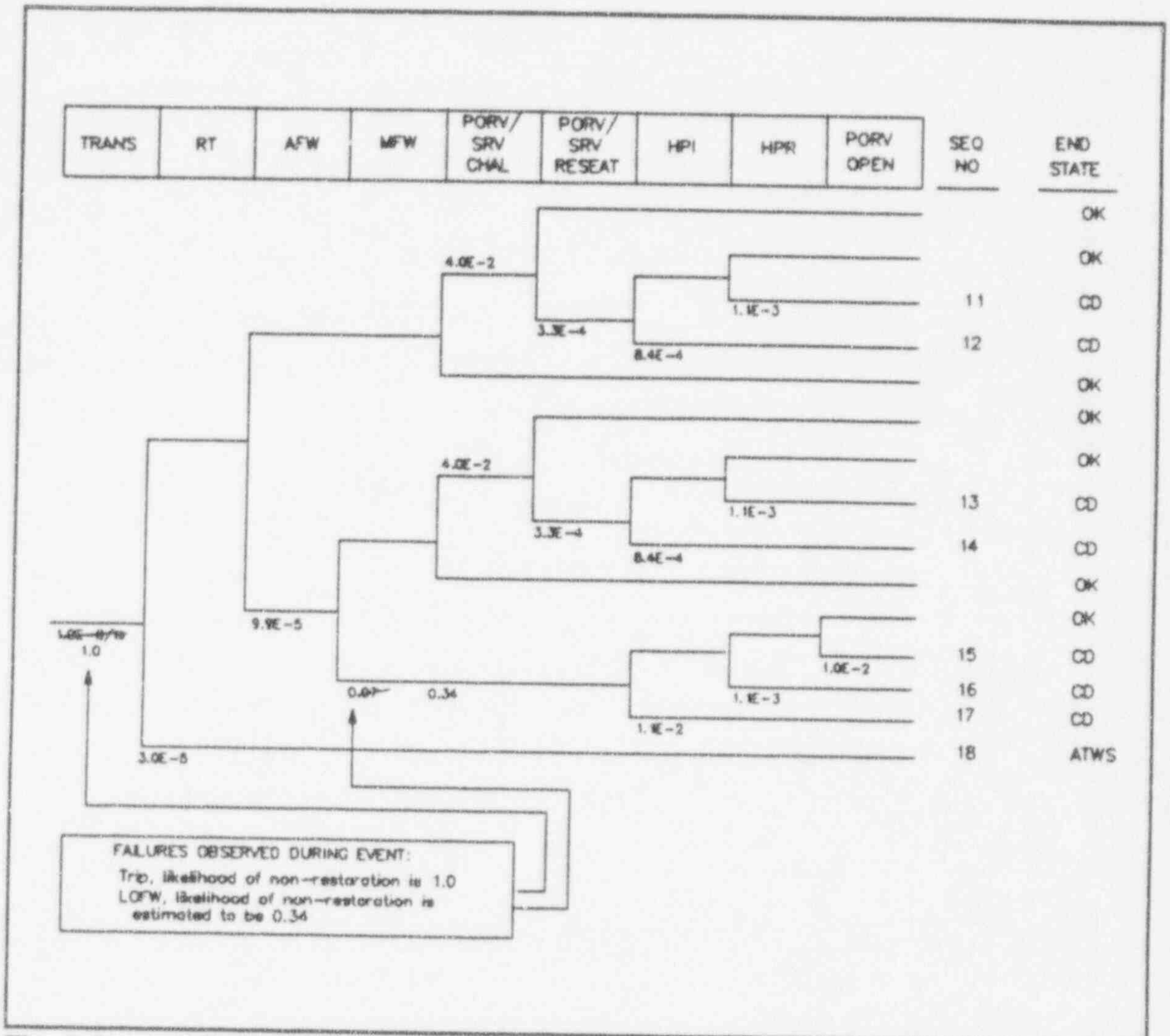

Fig. A.1. Example initiator calculation.

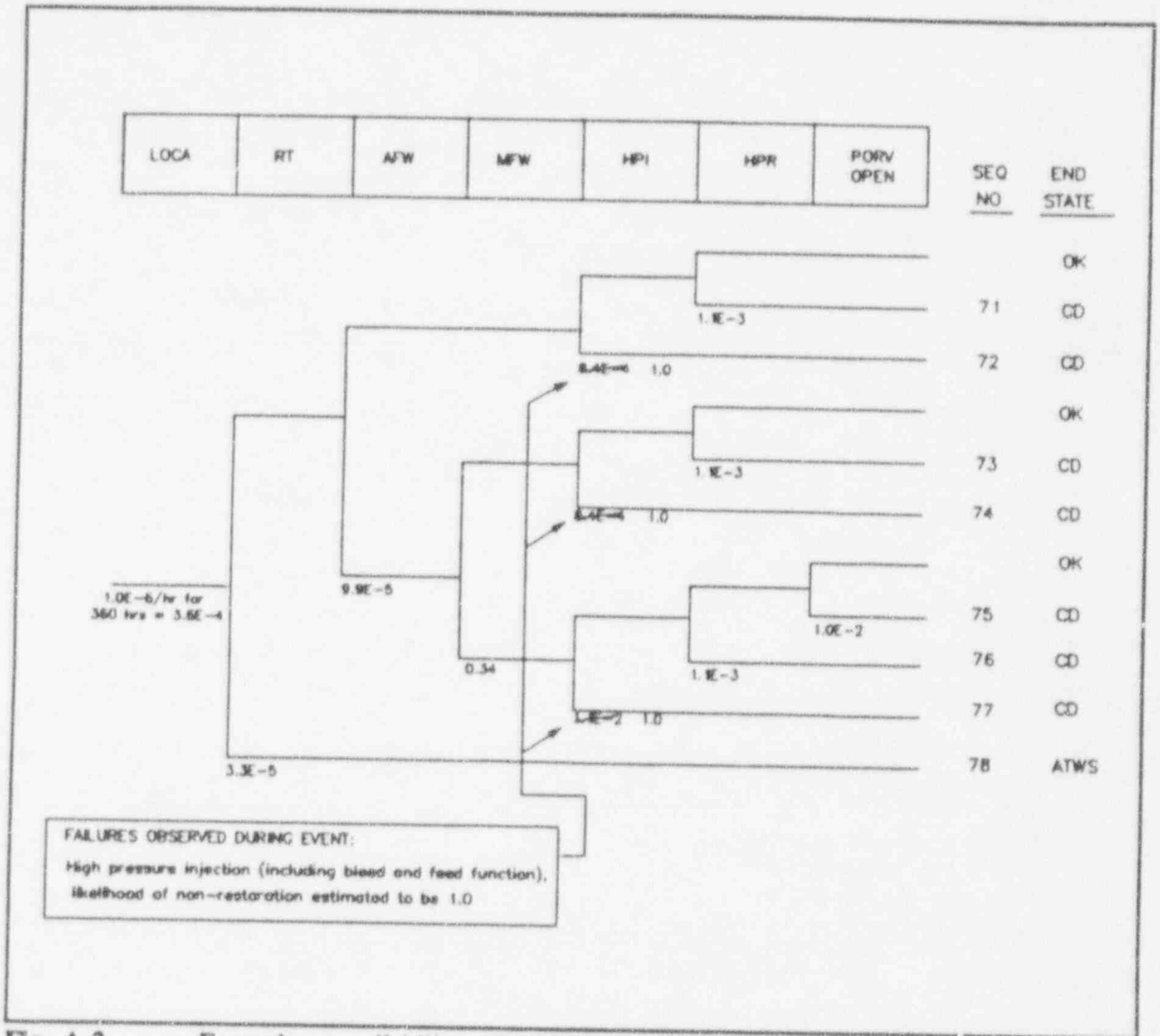

Fig. A.2. Example unavailability calculation

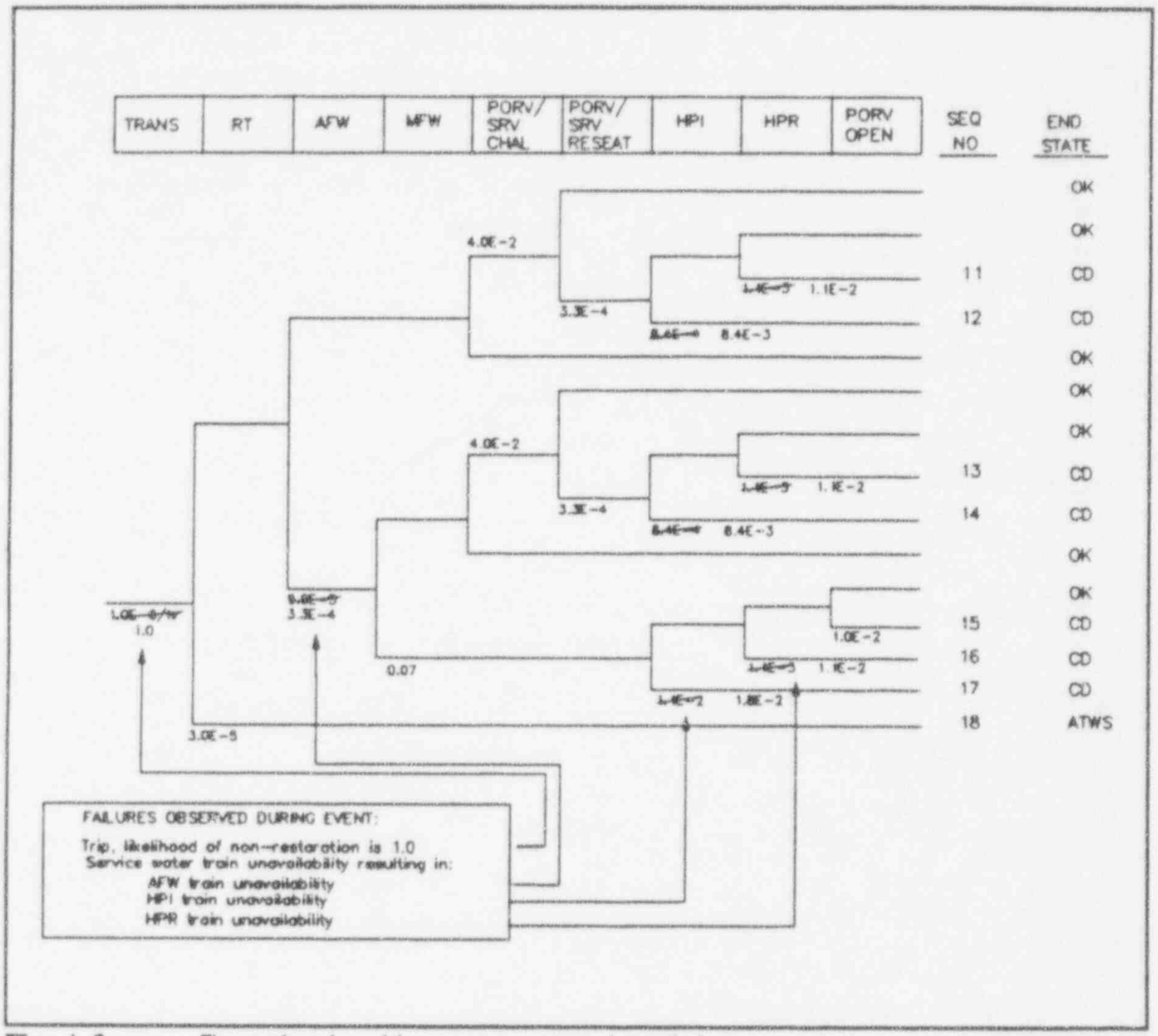

Fig. A.3. Example trip with support system degraded

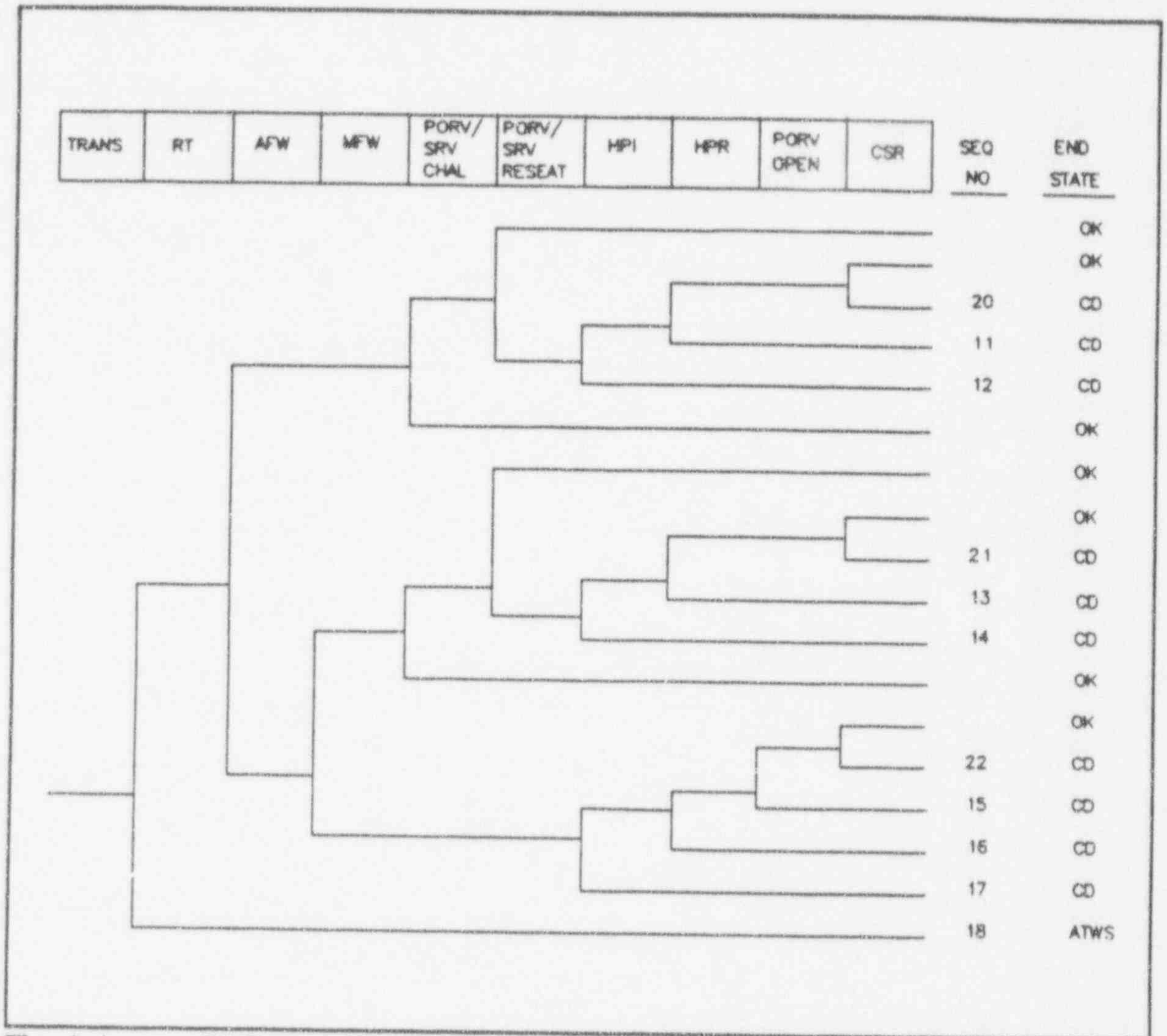

Fig. A.4. PWR class A nonspecific reactor trip

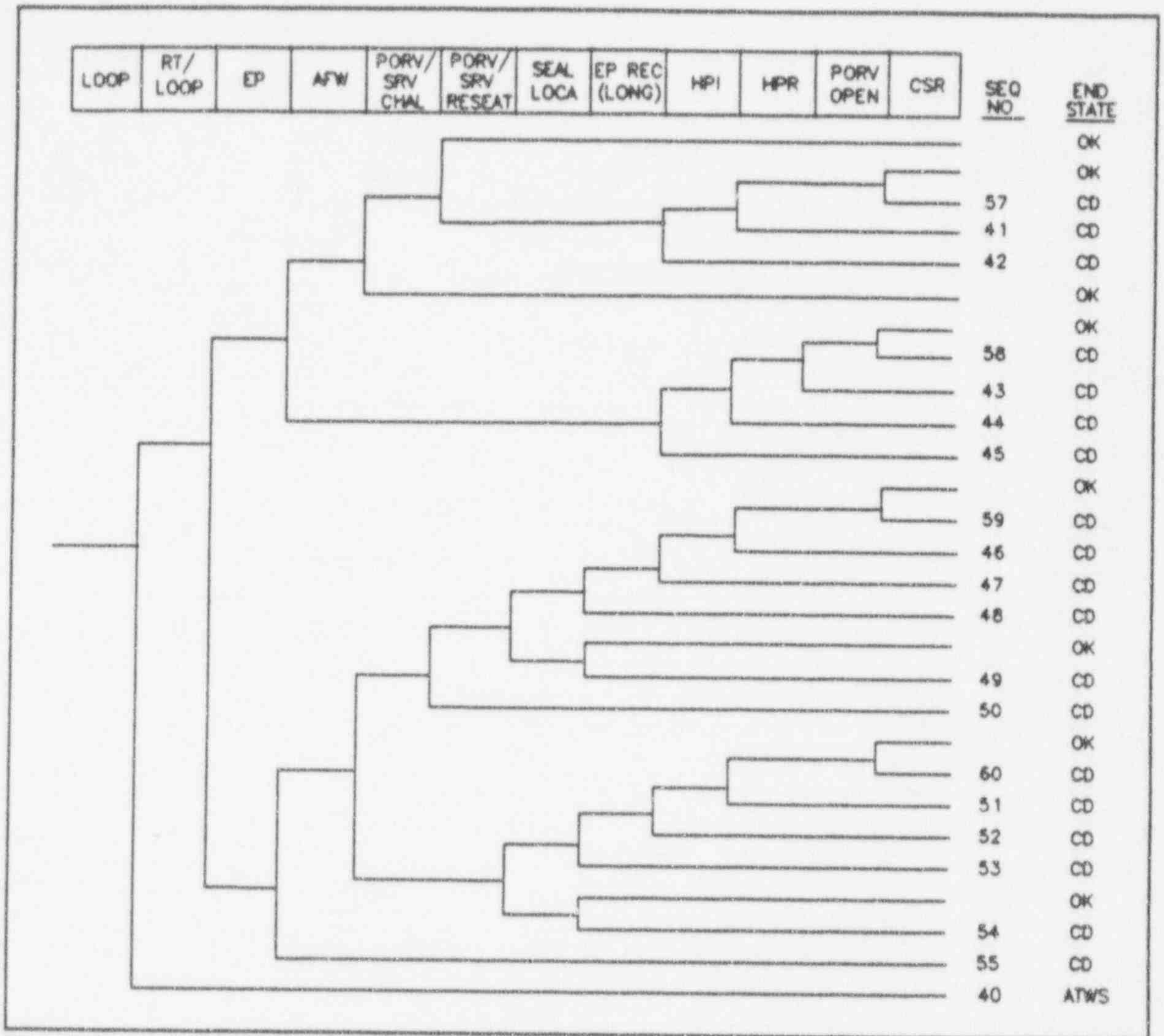

Fig. A.5. PWR class A loss of offsite power

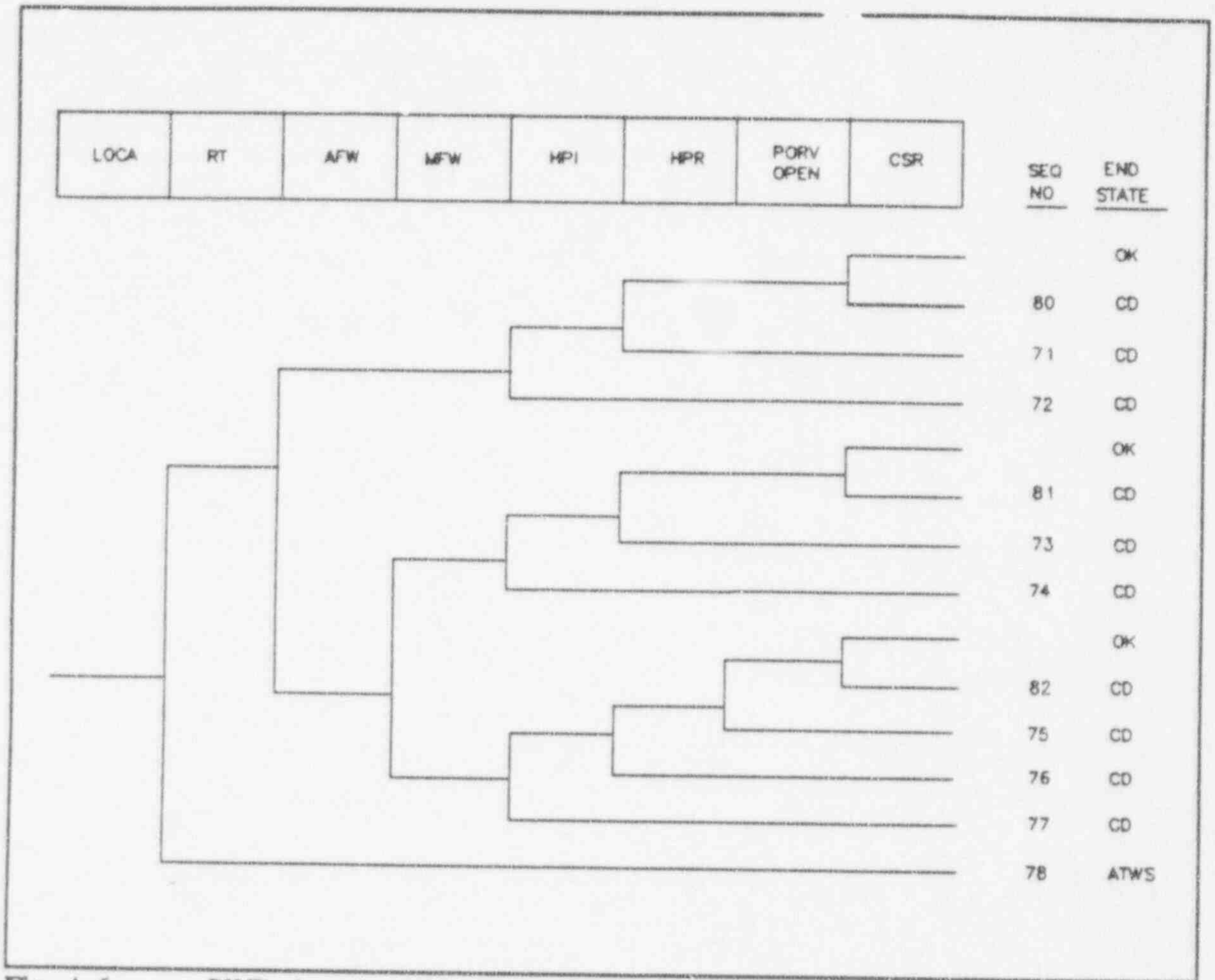

Fig. A.6. PWR class A small-break loss-of-coolant accident

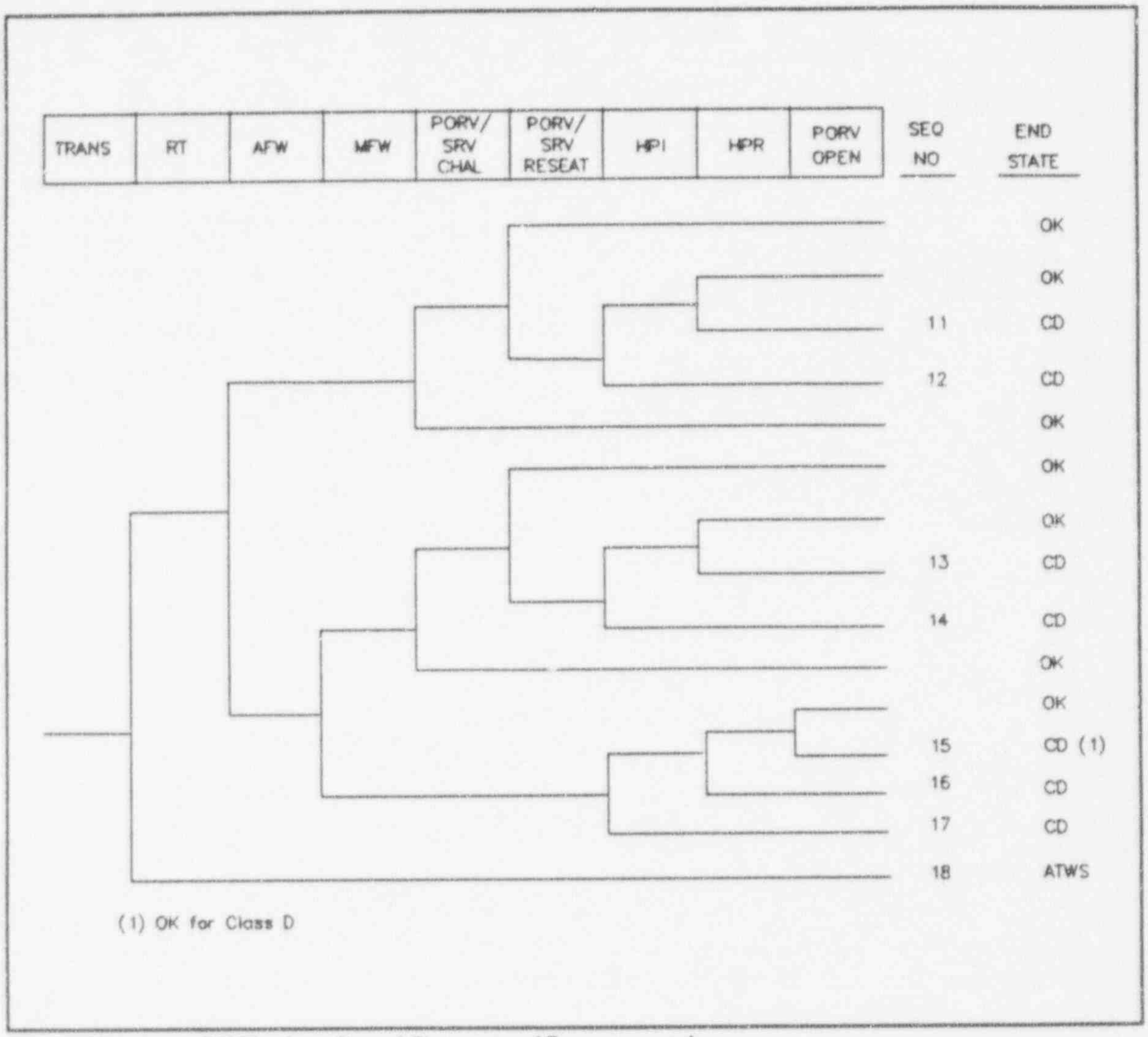

Fig. A.7. PWR class B and D nonspecific reactor trip

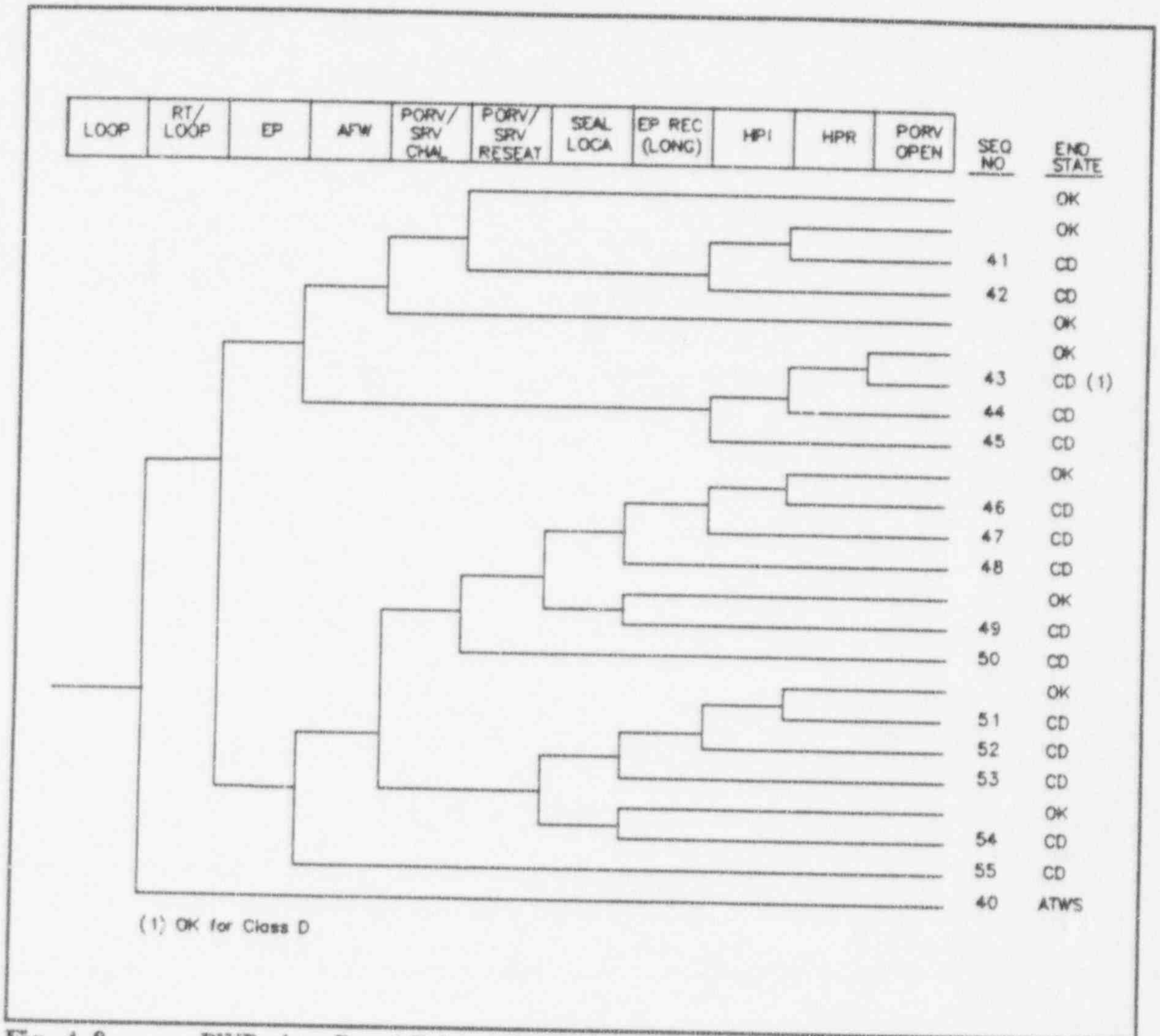

Fig. A.8. PWR class B and D loss of offsite power

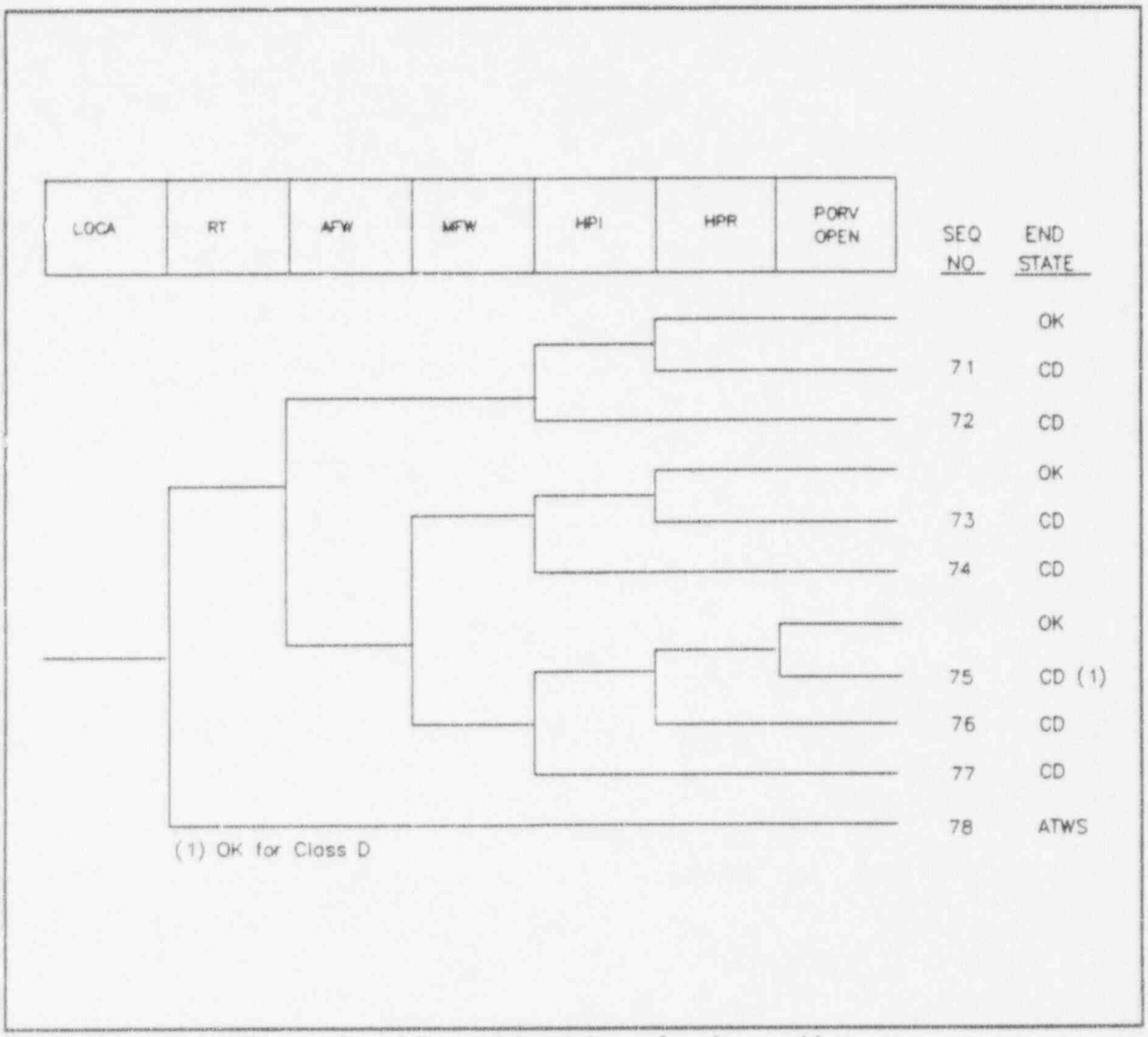

Fig. A.9. PWR class B and D small-break loss-of-coolant accident

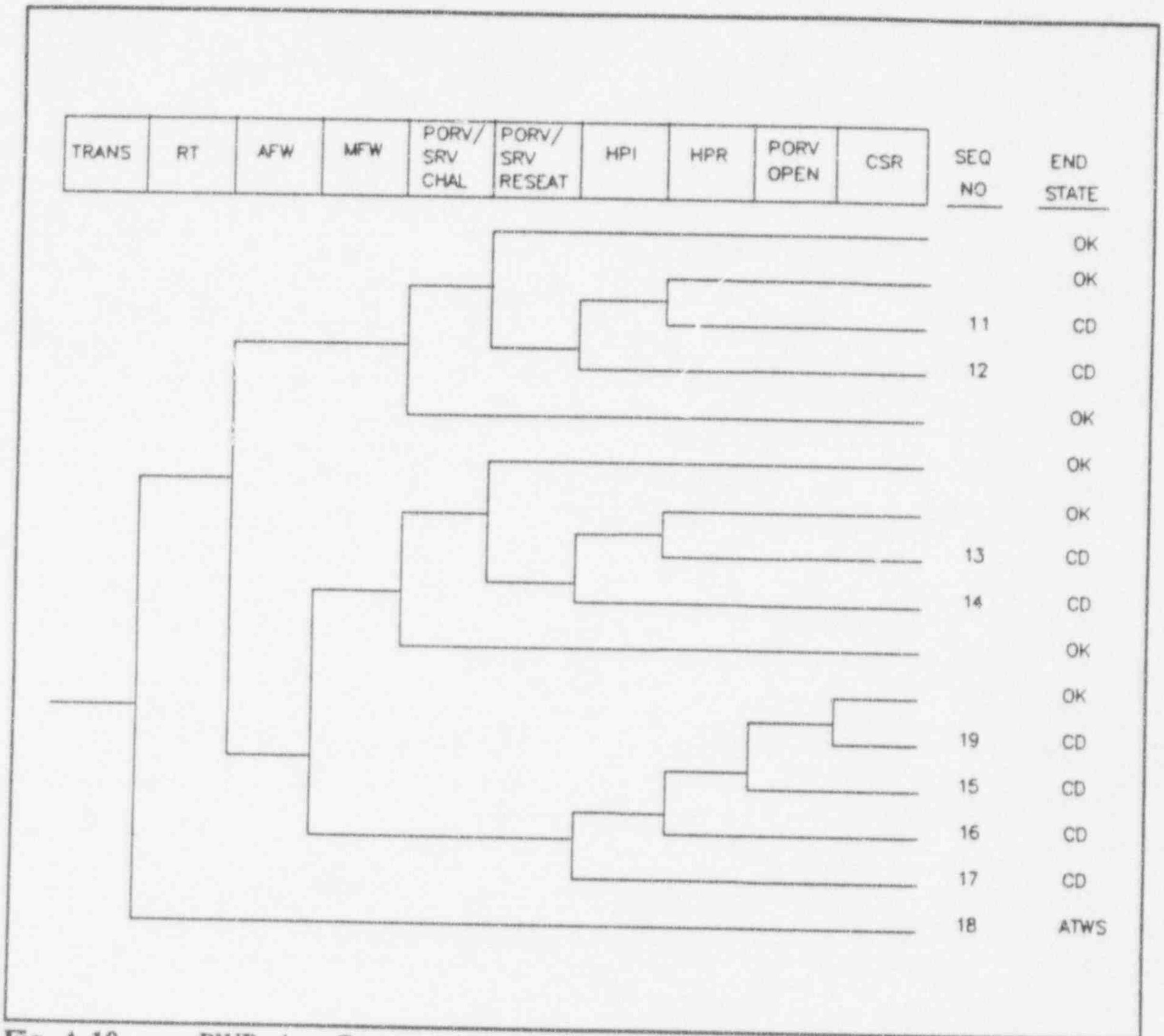

Fig. A.10. PWR class G nonspecific reactor trip

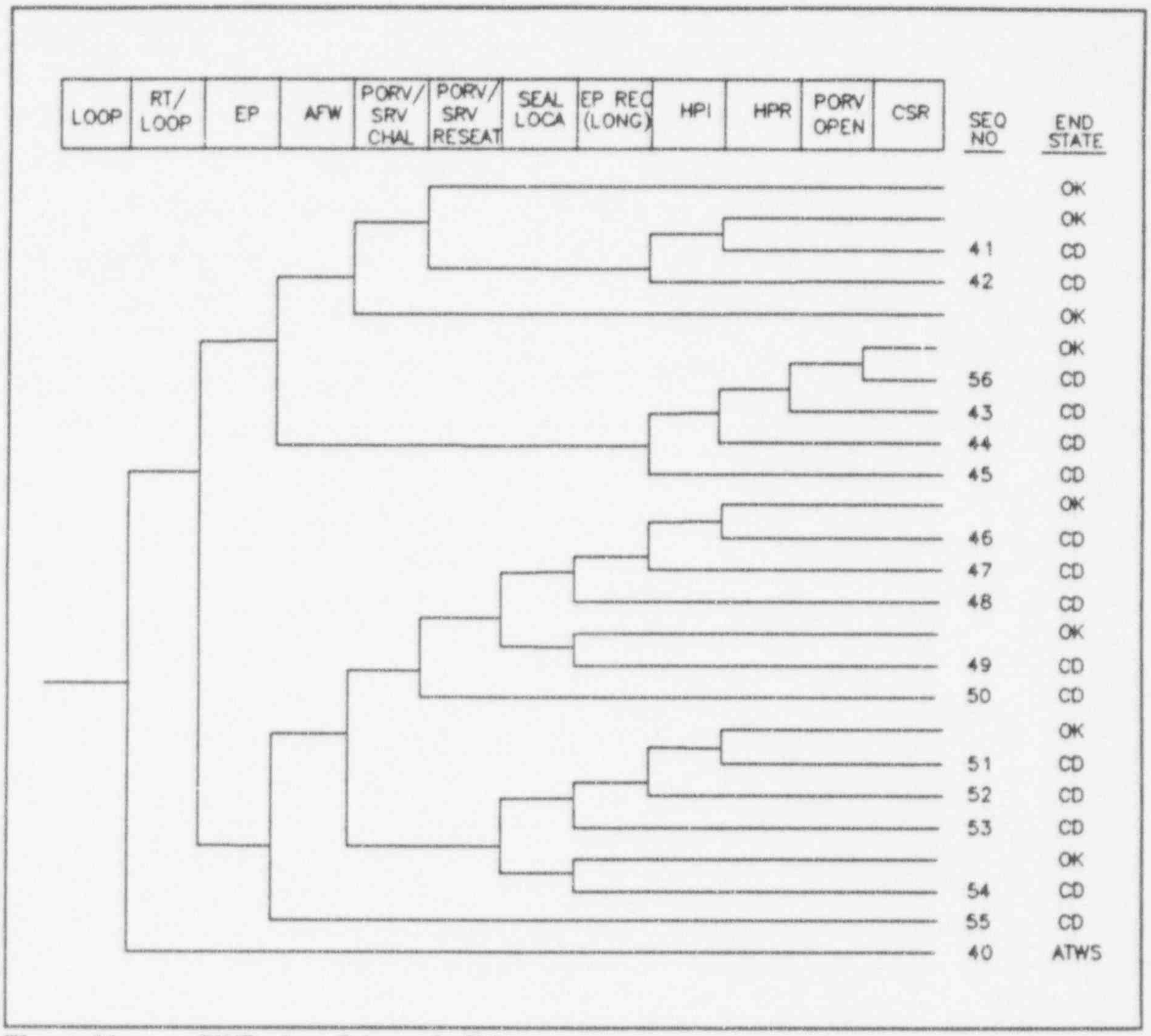

Fig. A.11. PWR class G loss of offsite power

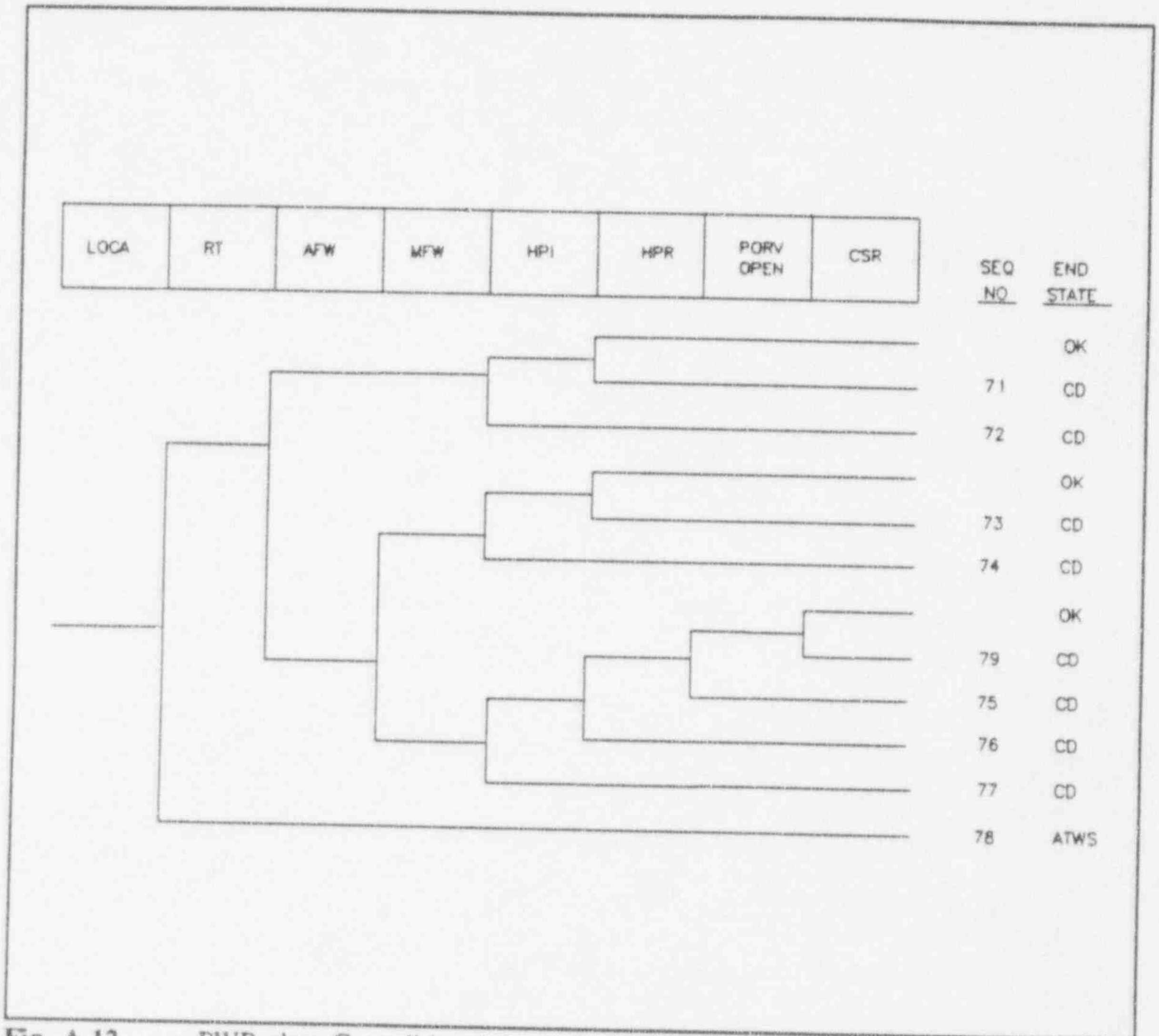

Fig. A.12. PWR class G small-break loss-of-coolant accident

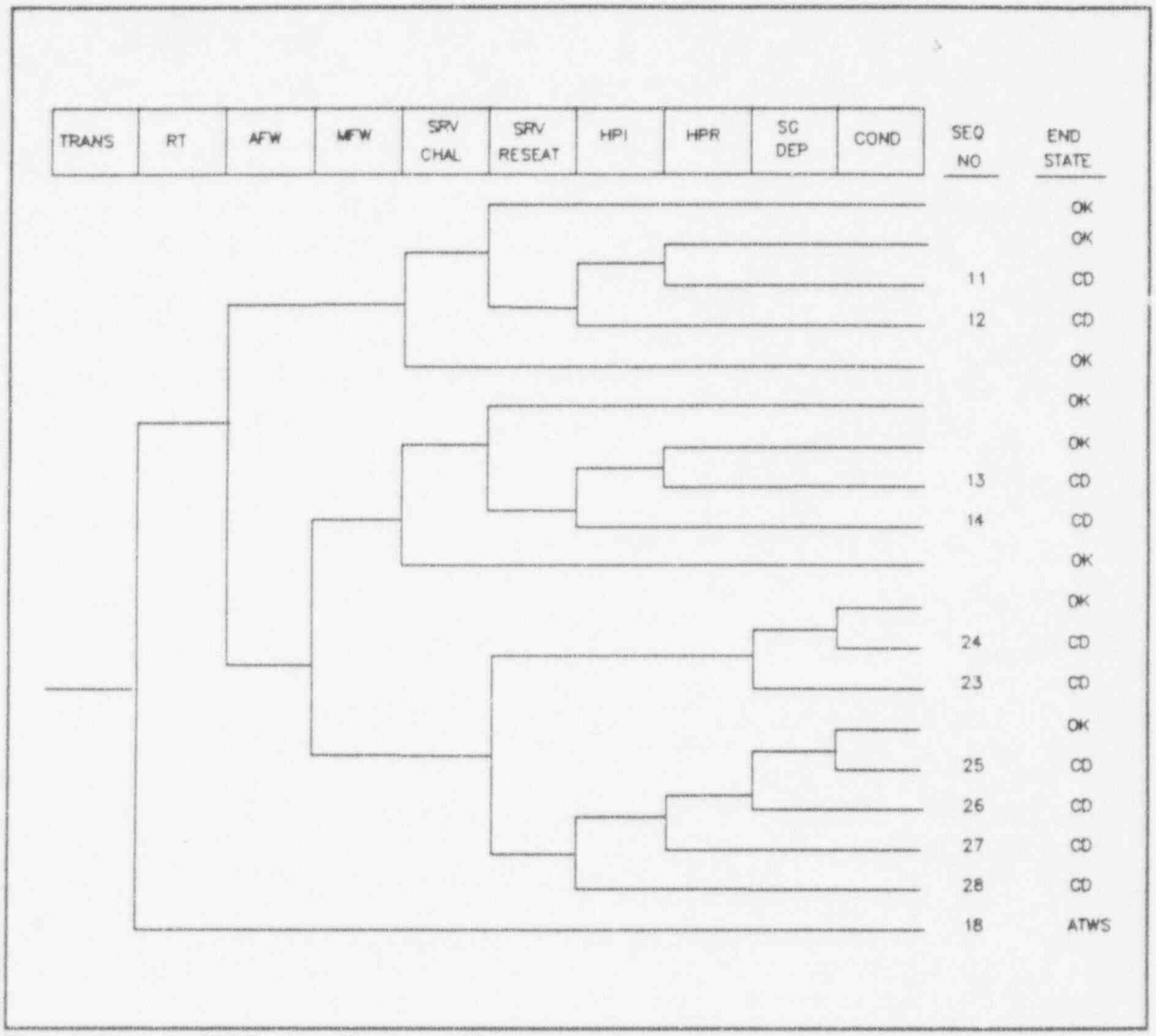

Fig. A.13. PWR class H nonspecific reactor trip

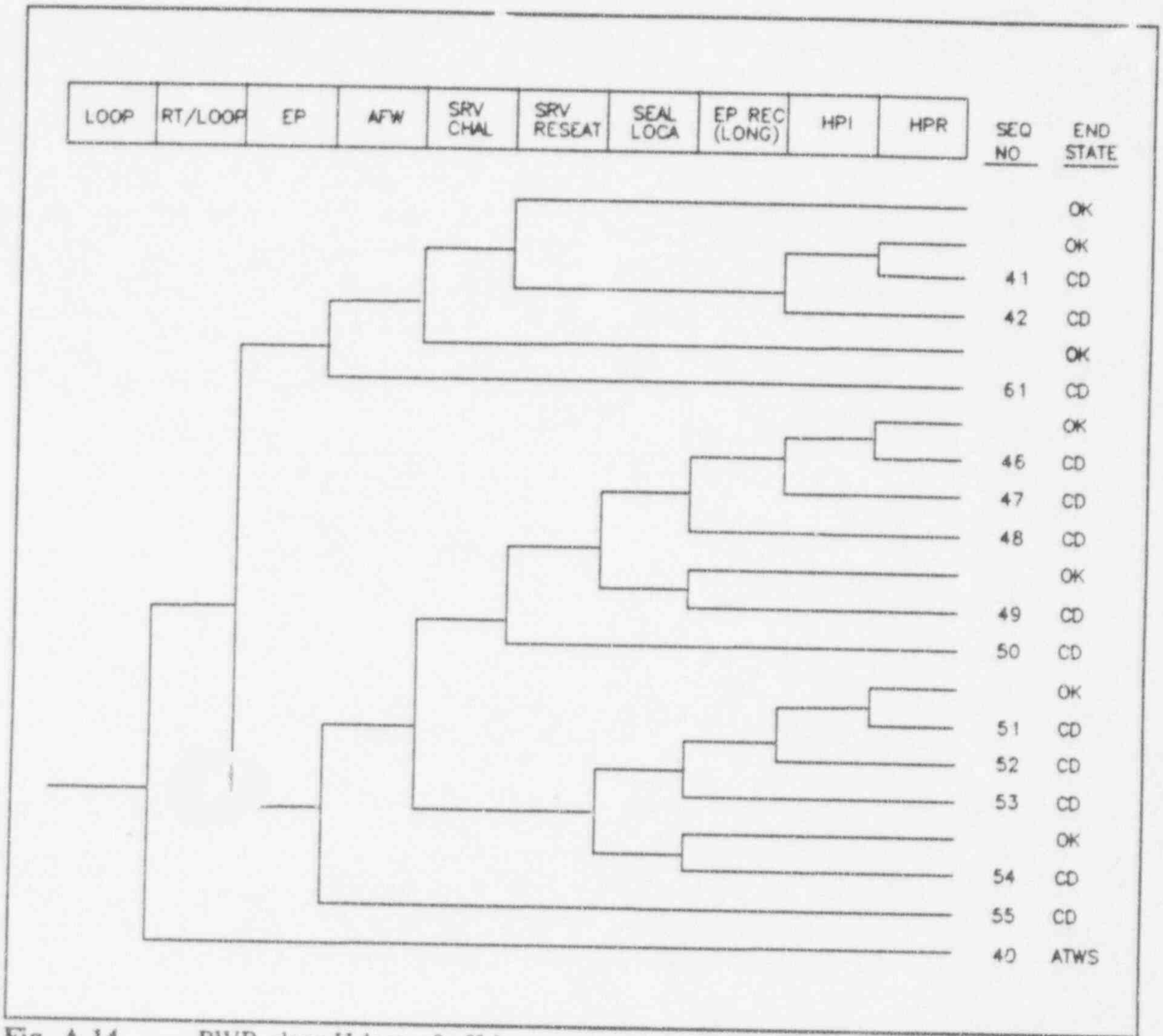

Fig. A.14. PWR class H loss of offsite power

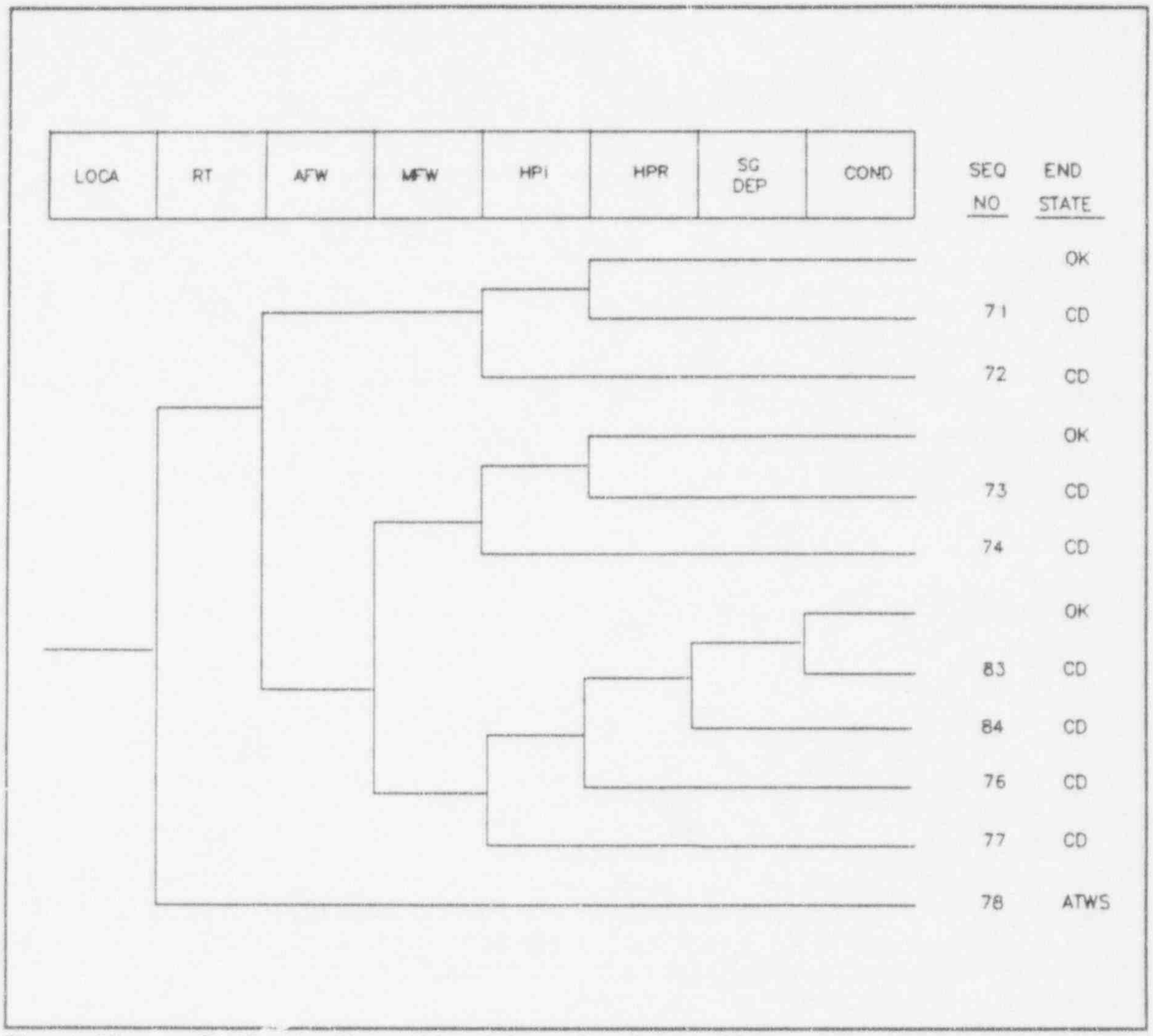

Fig. A.15. PWR class H small-break loss-of-coolant accident

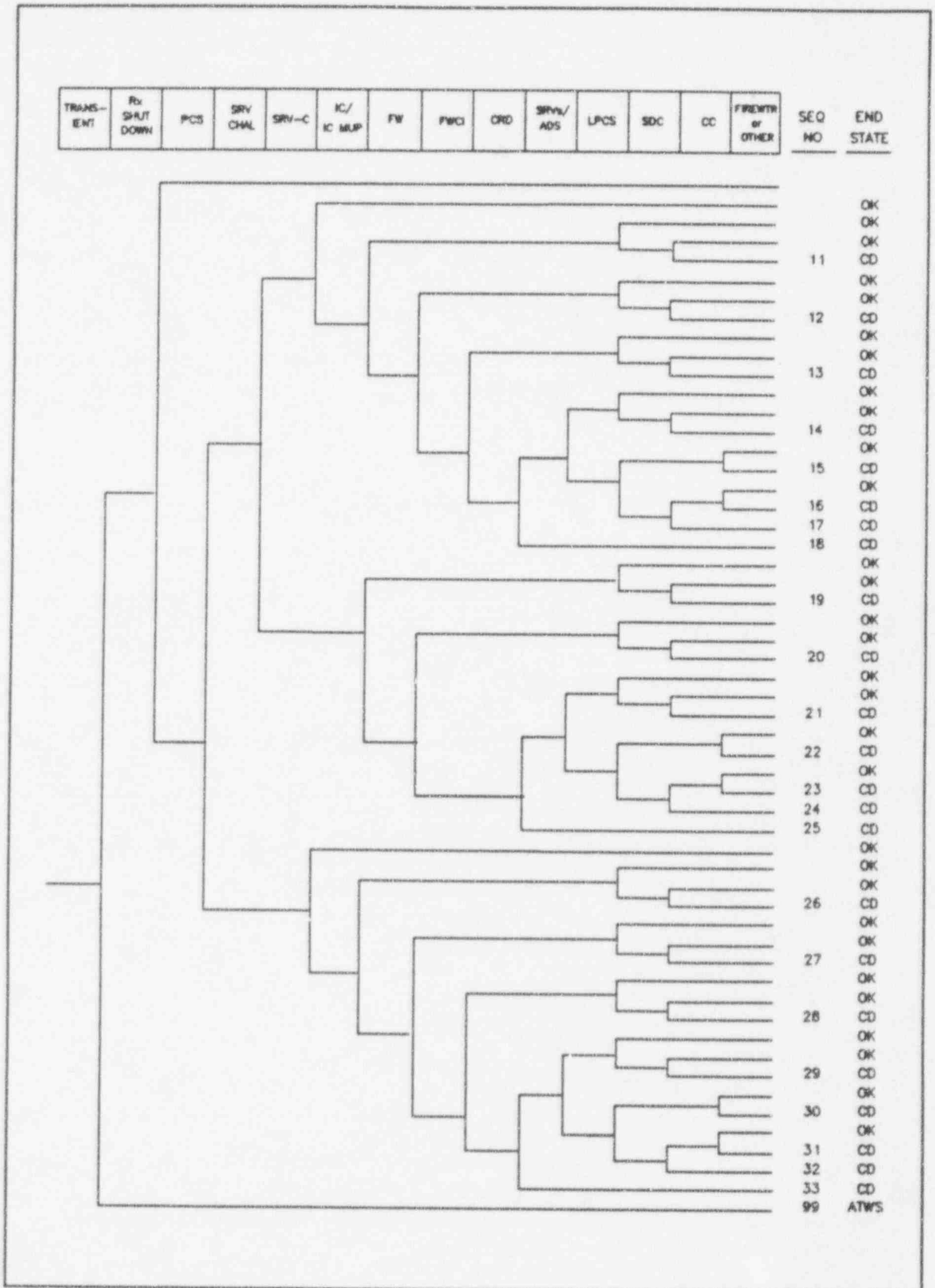

Fig. A.16. BWR class A nonspecific reactor trip

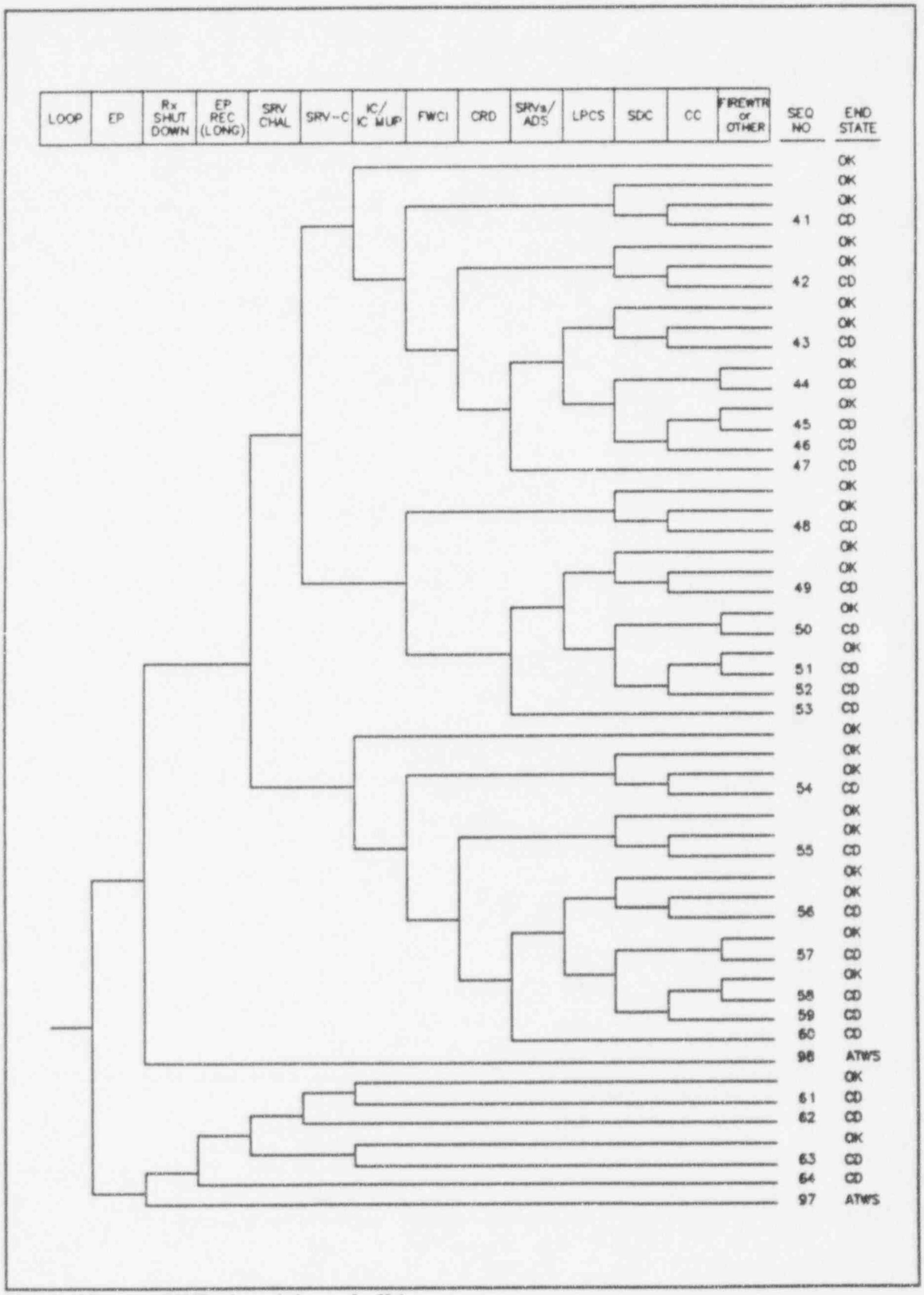

Fig. A.17. BWR class A loss of offsite power

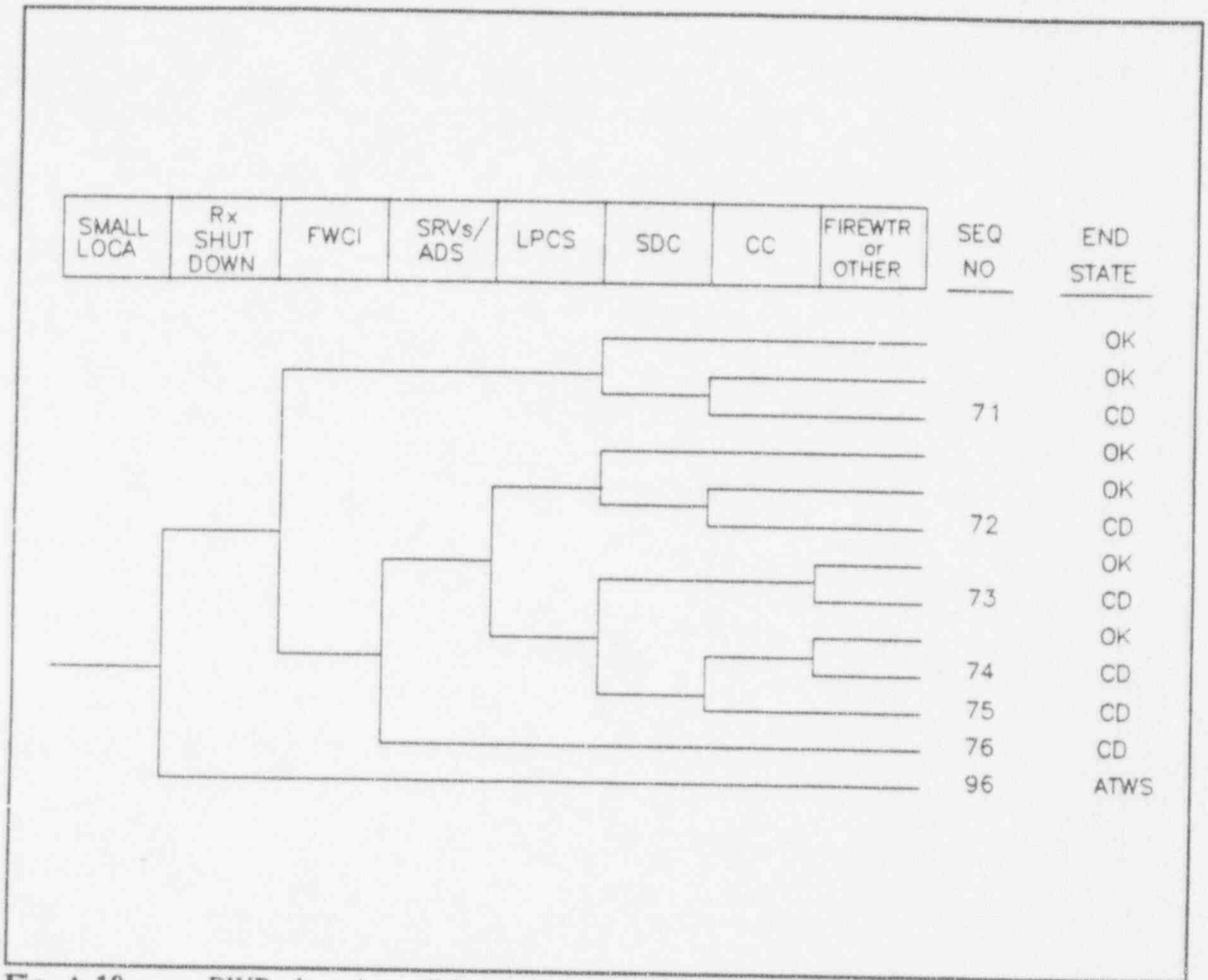

Fig. A.18. BWR class A small-break loss-of-coolant accident

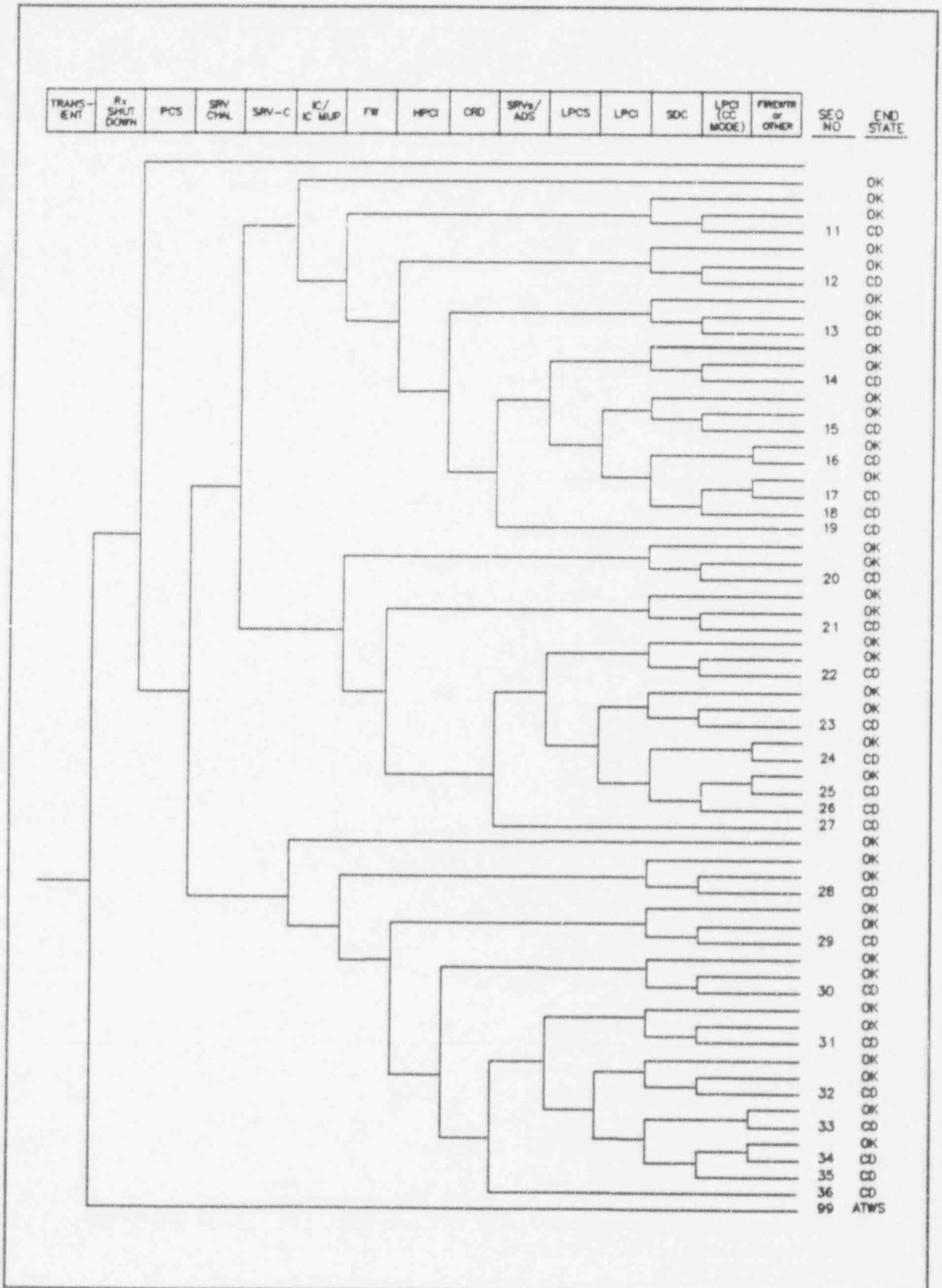

Fig. A.19. BWR class B nonspecific reactor trip

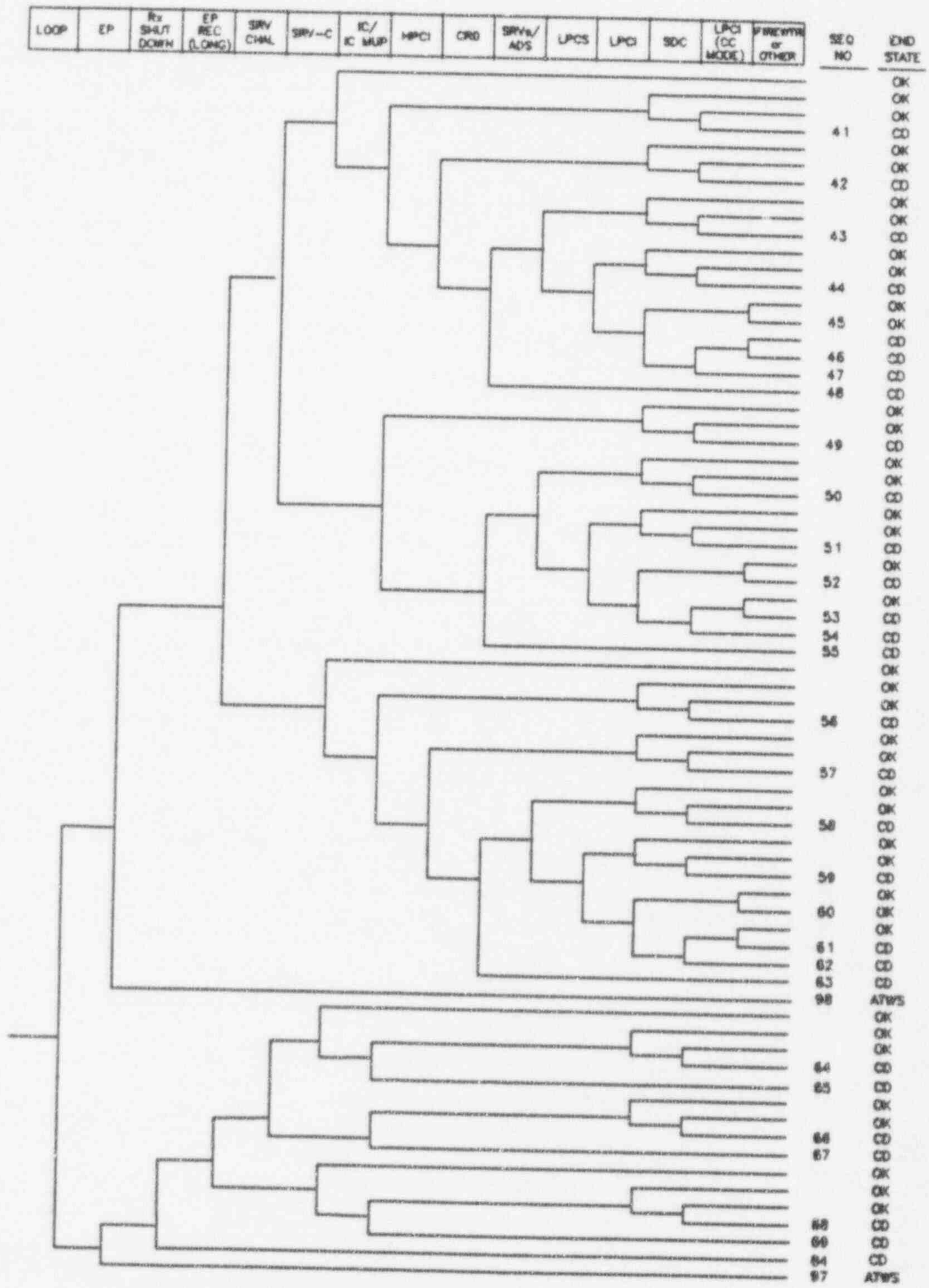

Fig. A.20. BWR class B loss of offsite power

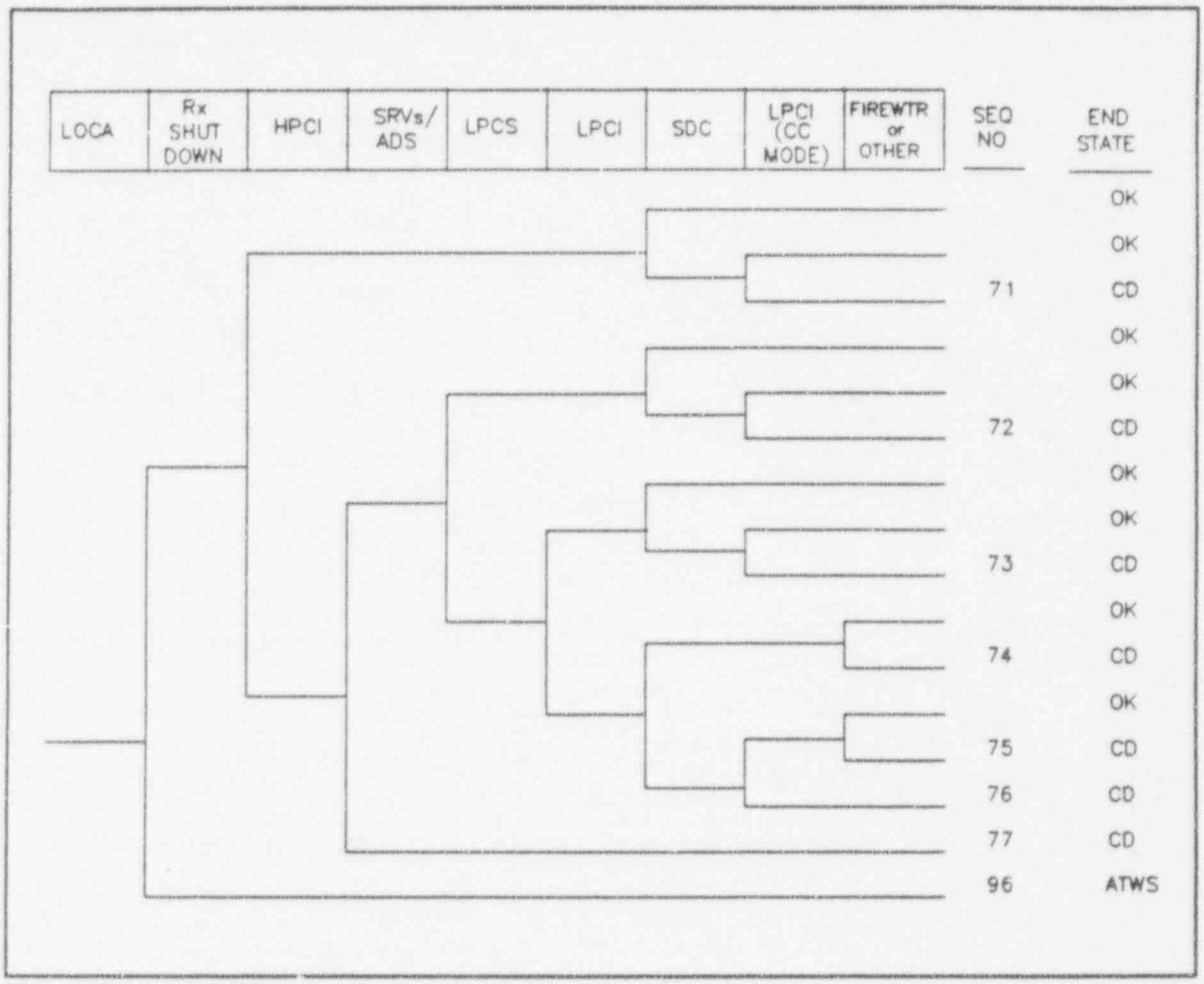

Fig. A.21. BWR class B small-break loss-of-coolant accident

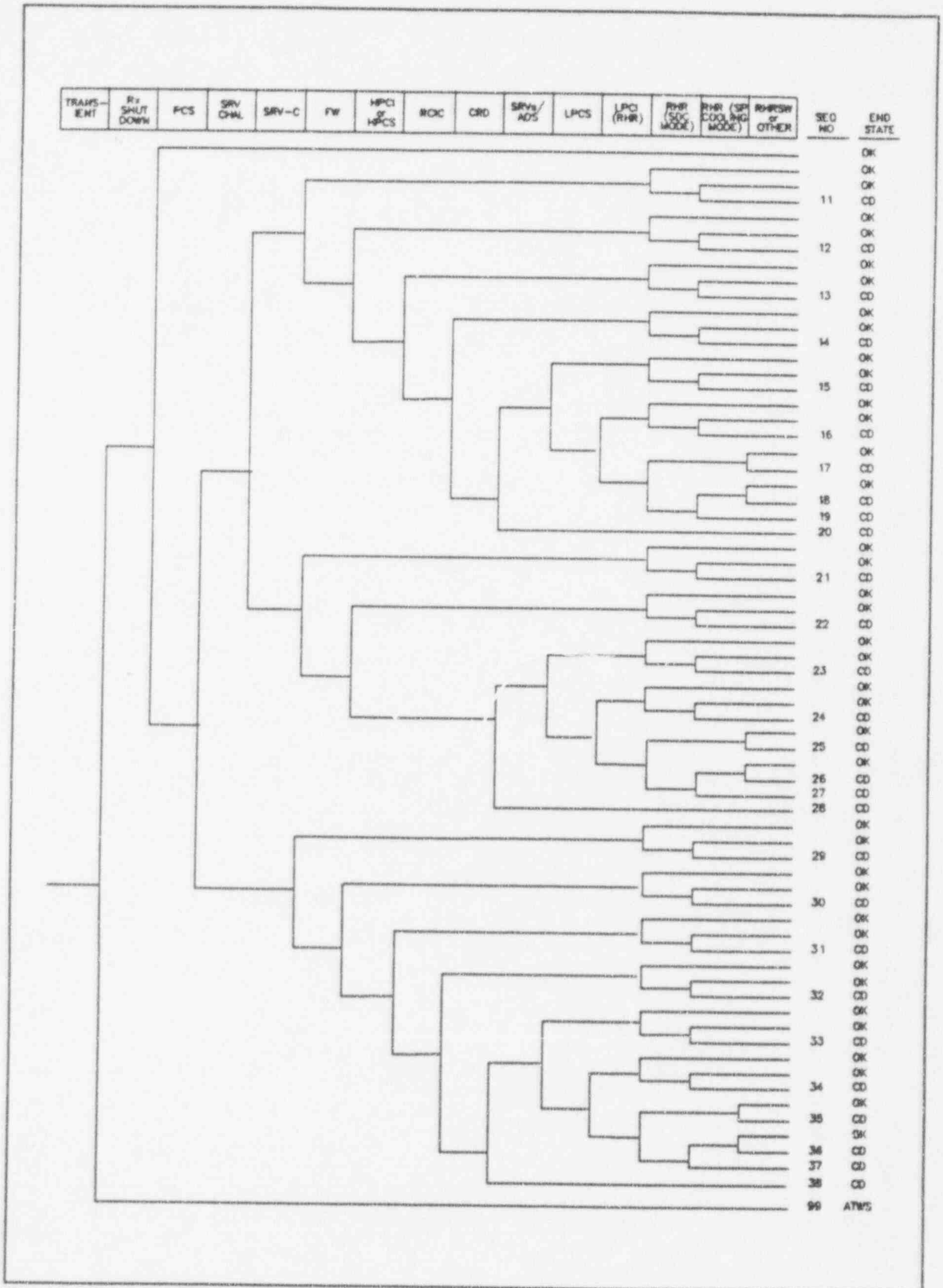

Fig. A.22. BWR class C nonspecific reactor trip

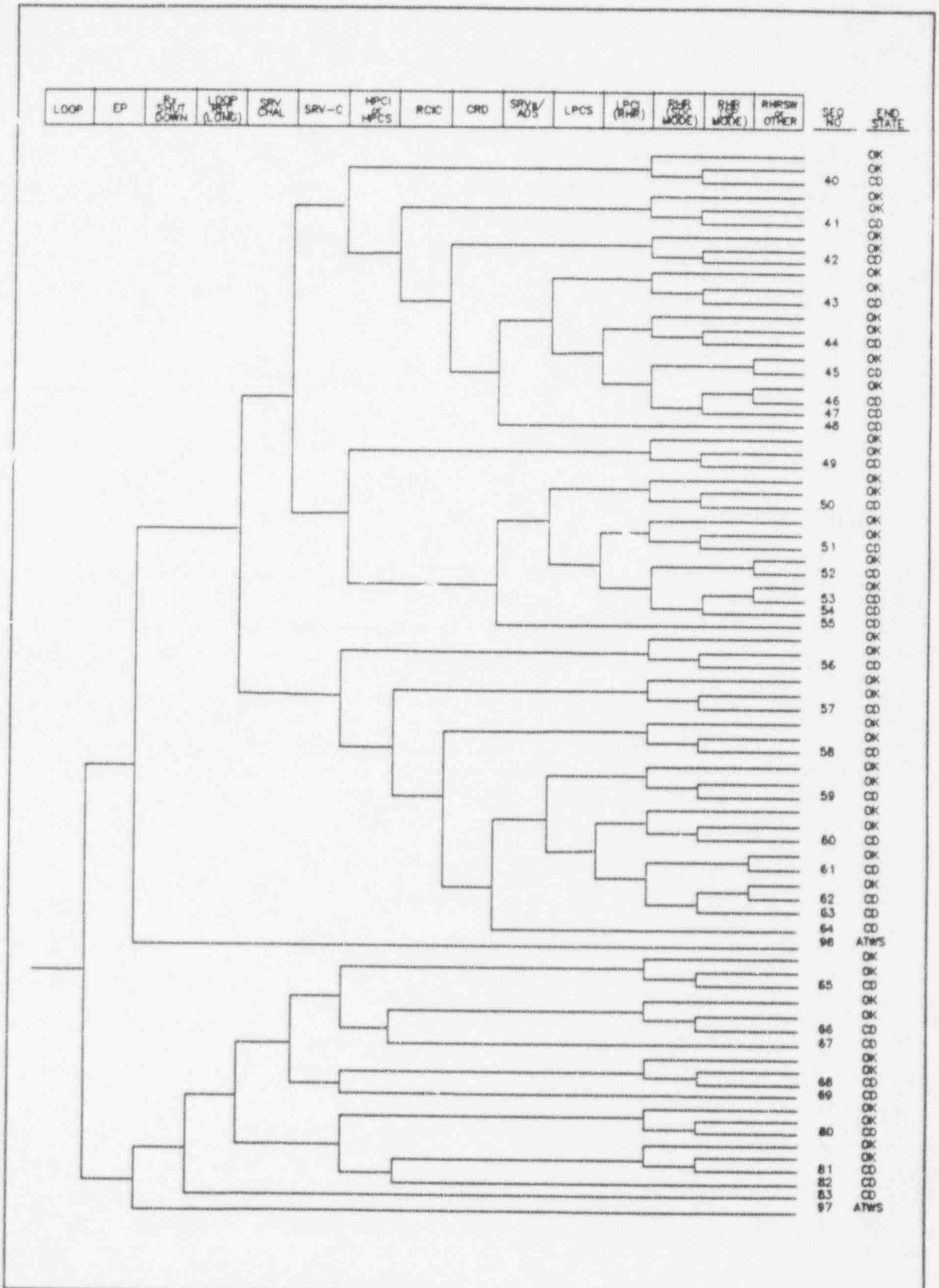

Fig. A.23. BWR class C loss of offsite power

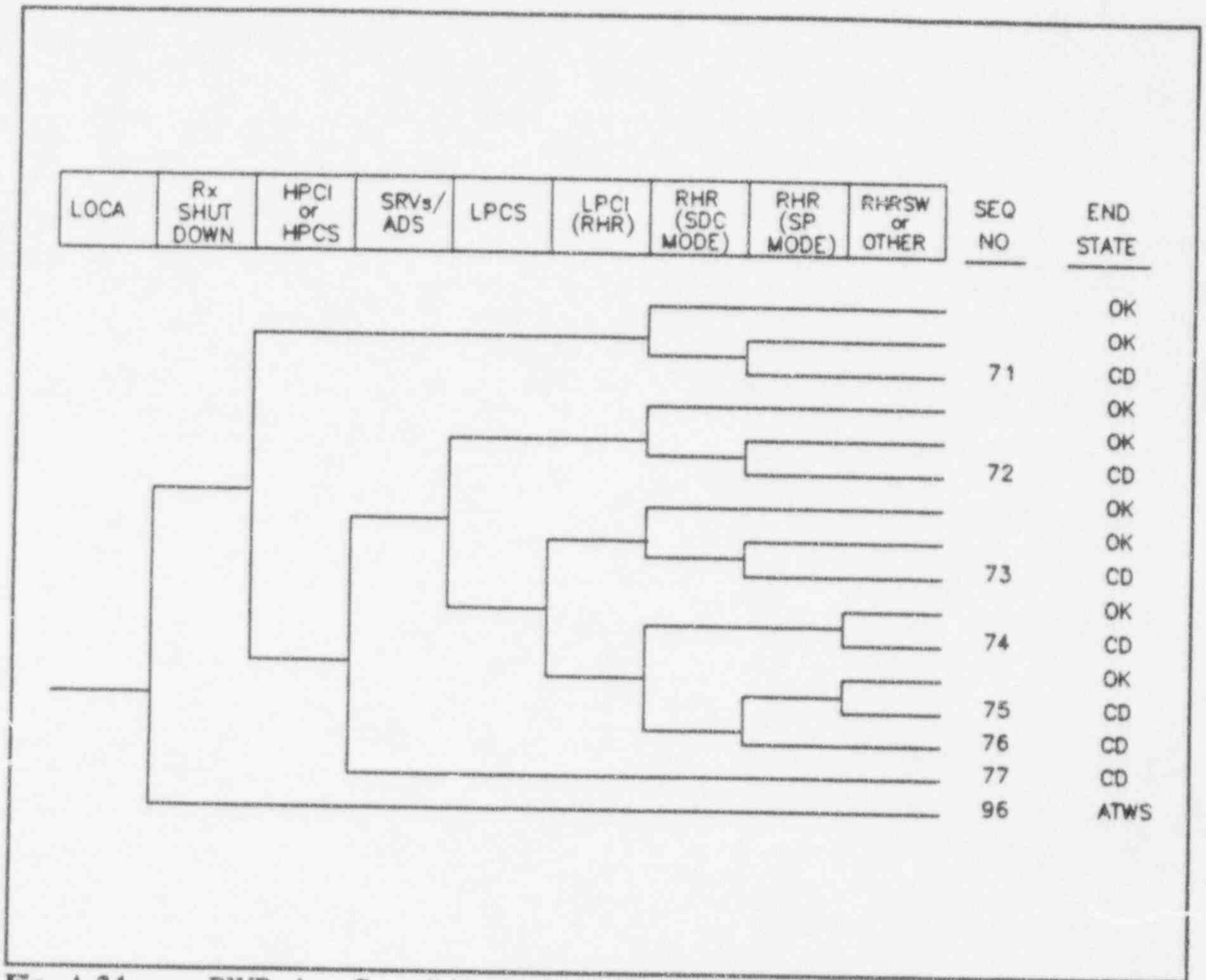

Fig. A.24. BWR class C small-break loss-of-coolant accident

UNIT 2

ENCLOSURES

FOR NOVEMBER 4-6, 1993 EVENT

ENCLOSURE 1

0.1 LER Number 412/93-012

Event Description: Failure of Both EDG Load Sequencers

Date of Event: November 4–6, 1993

Plant: Beaver Valley 2

0.1.1 Summary

On November 4, 1993, the automatic loading capability of the 2-1 emergency diesel generator (EDG) on a safety injection (SI) signal failed during a test. Two days later, on November 6, 1993, the automatic loading capability of the 2-2 EDG on SI also failed during a test. This failure will only occur when an SI signal is present coincident with a loss of the normal power supply to the ESF bus. The failure mechanism had existed since November 1990. Operator actions would have been necessary to allow manual loading of equipment on the ESF busses. The conditional core damage estimated for this event is 5.9×10^{-6}. The relative significance of this event compared to other postulated events at Beaver Valley 2 is shown in Fig. 1.

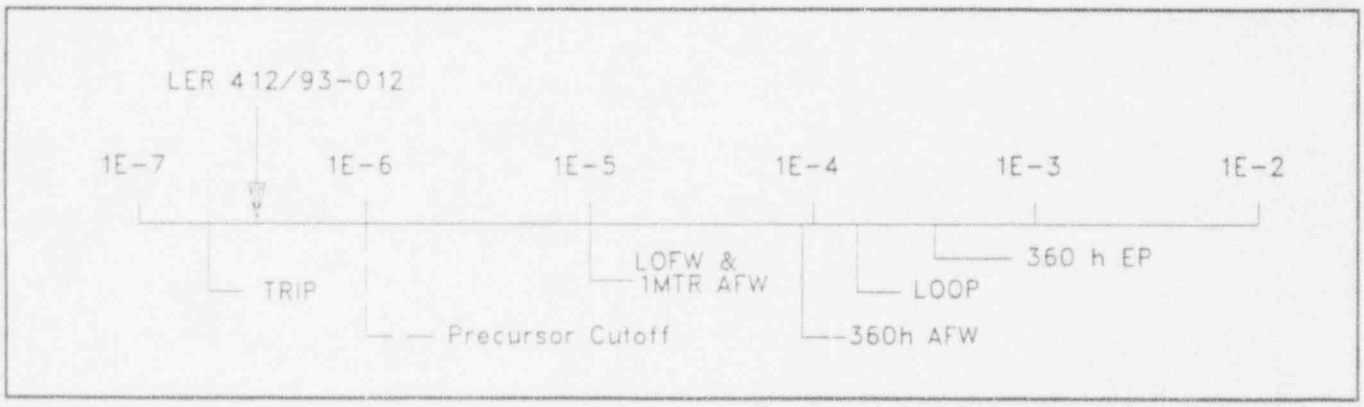

Fig. 1. Relative event significance for LER 412/93-012 compared with other potential events at Beaver Valley 2.

0.1.2 Event Description

On November 4, 1993, with the plant in cold shutdown, a test of the automatic loading capability of the 2-1 EDG on an SI signal was conducted. The test is performed during each refueling outage and verifies that the EDG circuitry will automatically load the safety-related loads on the emergency busses at the required time following the EDG start. During the test, the EDG started and reenergized the associated emergency bus, but the safety-related equipment did not automatically sequence onto the bus as expected. Approximately 2 min into the test, the SI signal was reset and the loads began to automatically sequence on the bus. An investigation following the test indicated that two relays in the solid-state protection system (SSPS) had the potential to cause the observed failures. The two relays were replaced, and the test was successfully rerun the following day.

(SSPS) had the potential to cause the observed failures. The two relays were replaced, and the test was successfully rerun the following day.

On November 6, 1993, the automatic loading capability of the 2-2 EDG on SI also failed during a test. Diagnostic test equipment which had been installed on the load sequencer identified the SI reset relay as the cause of the failure. This relay resets the sequencer if an SI signal occurs during a loss of bus voltage event. Voltage spikes caused by the opening of this relay resulted in the relay reclosing. This caused the loading sequencer to "lock-up."

This failure will only occur when an SI signal is present coincident with a loss of the normal power supply to the ESF bus. The automatic loading would have functioned properly for a postulated accident without the loss of normal power. The failure mechanism had existed since a modification of the sequencer relays in November 1990 (36 months). Operator actions would have been necessary to allow manual loading of equipment on the ESF busses. These actions include locally resetting the motor-control-centers (MCCs) to restore service water to the EDGs, the high-head SI pump coolers, and to operate essential ECCS valves.

0.1.3 Additional Event-Related Information

The EDG load sequencers automatically place vital safety-related equipment onto the ESF busses following a loss of voltage. The load sequencer is used to distribute the loads placed on the EDG in six discrete steps over a 1-min period. During the first refueling outage in 1989, problems were encountered with obtaining the necessary set-point repeatability with the existing electromechanical timer/relays used in the sequencer circuitry. During the second refueling outage in 1990, the electromechanical relays were replaced with microprocessor based timer/relays to improve set-point repeatability. The timers were also modified to be continuously energized to improve performance. During the third refueling outage, tests revealed that three of the eight timer/relays in each train had failed. The failures were due to overheating caused by the continuous energization. The timer/relay configuration was changed to be energized only when actuated. These previous failures were unrelated to the cause of the failures in 1993. Following the 1993 failures, diodes were installed to suppress the voltage spikes across the relays. The results of tests following the modification showed no failures after 80 cycles.

0.1.4 Modeling Assumptions

Two situations were modeled: (1) a postulated LOCA which induces a LOOP as a result of the effects of the plant trip on the electrical grid (see Fig. 2 for the event tree) and (2) a postulated LOOP where SI is initiated for feed and bleed (see Fig. 3 for the event tree).

<u>Case 1—LOCA with Transient-Induced LOOP</u>

In this case, a postulated LOCA is the initiating event. When the plant trips in response to the LOCA, the transient results in the loss of offsite power to the station. If offsite power is available, loads are fast transferred to the alternate offsite power source and the SI sequencer would operate properly. If offsite power is not available, then the EDGs will start. The normal feeder breaker to the safeguards busses trips open and load shedding occurs. The sequencer would then start and "lock-up." It is assumed that one-half hour is available to establish make-up to the reactor coolant system before core damage will occur. This event tree (Fig. 2) is based on the ASP LOCA tree for class A plants (see NUREG/CR-4674, Volume 17, Appendix A, Section A.3-1 and Fig. A.6). Values used in the quantification of the event tree are shown in Table 1.

LOCA Initiation Frequency

The condition was assumed to have existed for a 1-year period. The condition actually existed since the 1990 refueling outage (approximately 36 months). ASP initiating event frequencies are based on operation for 70% of a year (an approximation of the percentage of the year spent at power). Therefore the initiating event frequency is multiplied by 6132 h (= 365 × 24 × 0.7).

Loss of Offsite Power and Short-Term Offsite Power Recovery

It is assumed the probability of a LOOP induced by a LOCA is 1.0×10^{-3} (*Reactor Safety Study*, WASH-1400, NUREG-75/014, page II-90). A search of the Sequence Coding and Search System for transient-induced LOOPs from 1984 to present revealed five transient-induced LOOPs out of 3985 trips. This yields a rate of 1.25×10^{-3} per trip. This provides a degree of substantiation for the WASH-1400 value. It is assumed offsite power recovery is possible only in the first one-half hour. The nonrecovery value of 0.48 is that associated with a grid-related LOOP (from ORNL/NRC/LTR-98-11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989), since the initiating cause of the LOOP was assumed to be grid disturbance caused by the plant trip. Note that if no LOOP occurs, the event is a standard small-break LOCA, and the sequencers will operate properly. Therefore, this branch does not contribute to the conditional core damage probability.

Emergency Power

If the EDGs fail to start following the failure to recover offsite power, it is assumed that insufficient time is available to recover the EDGs and to manually load the busses. Therefore, the nonrecovery value of the EDGs for this case is set to 1.0, and the failure of the EDGs to start leads directly to core damage (sequence 24).

Loading of the ESF Equipment on the Safeguards Busses

The operator actions necessary to load required equipment onto the safeguards bus are treated as a single top event. It would be obvious to the operators that manual actions were required to load equipment on the safeguards busses since none of the safeguards equipment loads would be picked up by the EDGs. In addition, the fact that nothing loaded onto the EDGs would probably lead the operators to suspect a sequencer failure. It is assumed that the operators would have procedural guidance to direct their actions. Equipment recovery would have to be prioritized to prevent equipment damage. Service water would have to be restored to the running EDGs to cool them, and HPI pumps would be needed to provide make-up to the RCS. Local actions would be required to reset the motor-control-centers (MCCs) to restore service water (SW) to the EDGs, the high-head SI pump coolers, and to operate essential ECCS valves. Since this process requires a number of coordinated actions and local operator actions, an operator failure rate of 0.34 was assumed (ASP Recovery Class R2, see NUREG/CR-4674, Vol. 17, Appendix A, Section A.1).

If offsite power is restored within the first half-hour, it is assumed that the automatic sequencer will operate. However, since the sequencer will lock-up within the first few seconds of the event, it is assumed that the manual resetting of the MCCs will still have to be accomplished to restore SW to the EDGs, the high-head SI pump coolers, and to operate essential ECCS valves

Failure to successfully load equipment onto the safeguards bus leads to a core damage state (sequences 11 and 23). Although the turbine-driven AFW pump will operate as a heat sink, nc RCS make-up is provided. Therefore, core damage will occur. If loads are successfully loaded, then the remainder of the tree (sequences 1–10 and 13–23) is the same as the standard LOCA tree (see NUREG/CR-4674, Volume 17, Appendix A, Section A.3-1 and Fig. A.6, Sequences 71-77 and 80-82) and uses standard ASP values.

Case 2—LOOP with SI Initiated for Feed and Bleed

Case 2 involves a postulated LOOP initiator. If offsite power is not recovered and AFW and MFW fail, feed and bleed is utilized as a heat sink. It is assumed the operator will actuate high-pressure injection by manually actuating an SI signal. This will cause the sequencers to "lock-up" since offsite power is not available. Loads alre.'dy connected to the bus will not be shed. Therefore, the equipment started for the loss of voltage signal will remain operable; however, the additional equipment started by the SI signal will not start. The SI signal has to be initiated before the normal supply breaker is reclosed since this action will restore the sequencer to operation. This event tree (Fig. 3) is based on the ASP LOOP tree for class A plants (see NUREG/CR-4674, Volume 17, Appendix A, Section A.3-1 and Fig. A.5). Values used in the quantification of the event tree are shown in Table 1.

LOOP and Short-Term Nonrecovery of Offsite Power

The condition was assumed to have existed for a 1-year period. The condition actually existed since the 1990 refueling outage (approximately 36 months). ASP initiating event frequencies are based on operation for 70% of a year (an approximation of the percentage of the year spent at power). Therefore, the initiating event frequency is multiplied by 6132 h (= 365 × 24 × 0.7). It is assumed that offsite power recovery is possible only in the first one-half hour. The LOOP nonrecovery is the generic ASP value which assumes all types of initiators contribute (plant-centered, grid-related, severe-weather-related and extreme severe-weather-related). The nonrecovery in the first half hour is incorporated into the initiating event frequency value. If offsite power is recovered, the sequencers will not fail during initiation of feed and bleed. The sequences associated with this recovery do not contribute to the conditional core damage for the event; therefore, they are not depicted in Fig. 3.

Emergency Power

If the EDGs fail to start following the failure to recover offsite power, it is assumed that sufficient time is available to recover the EDGs and to manually load the busses. The manual loading of the EDGs will not be required immediately since initiation of feed and bleed will be delayed slightly. In addition, the turbine-driven AFW pump could also operate without restoration of the EDGs. Therefore, the nonrecovery value for the EDGs in this case is set to 0.80. If the EDGs are not recovered, then the nominal LOOP tree branches apply (see NUREG/CR-4674, Volume 17, Appendix A, Section A.3-1 and Fig. A.5, Sequences 46–55 and 59–60). Offsite power recovery is required for repowering ESF/ECCS equipment if the EDGs are not recovered. If offsite power is recovered, the sequencers will not fail. Therefore, the sequencer failure does not affect these branches, and they do not contribute to the conditional core damage for the event.

Loading of the ESF Equipment on the Safeguards Busses

The operator actions necessary to load required equipment onto the safeguards bus are treated as a single top event. If AFW is successful and SI is initiated in response to a stuck open PORV or SRV, it is assumed that sufficient time is available to manually load equipment. Since this process requires a number of coordinated actions and local operator actions, an operator failure rate of 0.34 was assumed (ASP Recovery Class R2, see NUREG/CR-4674, Vol. 17, Appendix A, Section A.1).

If AFW does not operate, it is assumed that the operators do not initiate feed and bleed until required to do so by procedure. In this case, it is assumed that insufficient time is available at that point to manually load equipment onto the bus and have successful feed and bleed. Therefore, the ESF loading failure rate is set to 1.0 in this situation (sequence 34).

Failure to successfully load equipment onto the safeguards bus leads to a core damage state. Although the turbine-driven AFW pump will operate as a heat sink, no RCS make-up is provided (sequence 29) or feed and bleed is not successful (sequence 34). Therefore, core damage will occur.

0.1.5 Analysis Results

The estimate of the conditional core damage probability for this event is 5.9×10^{-6}. This consists of a contribution of 2.1×10^{-6} from the postulated LOCA concurrent with the transient-induced LOOP (Case 1) and 3.8×10^{-6} from the postulated LOOP (Case 2). The dominant core damage sequence for Case 1, shown in Fig. 2, involves a postulated LOCA, transient-induced LOOP that is recovered in the first half hour, successful reactor trip, and failure to load the ESF busses (reset the MCCs). The other dominant sequence involves a postulated LOCA, transient-induced LOOP that is not recovered in the first half hour, successful reactor trip, emergency power restoration, and failure to load the ESF busses. The dominant sequence for Case 2, shown in Fig. 3, involves a postulated LOOP that is not recovered in the first half hour, successful reactor trip, emergency power restoration, failure of AFW, and failure to load the ESF busses.

Additional information concerning this event is included in Augmental Inspection Team report 50-412/93-81.

Table 1. Values Used in the Quantification of the Event Trees

Top Event	Description	Value
LOCA	LOCA initiator Initiating frequency = 2.4E-6, nonrecovery = 0.43 Duration of unavailability = 6132 h	6.30E-03
LOOP	Transient-induced LOOP Frequency = 1E-3/demand	1.0E-03
	LOOP recovery (short-term)-recovery in the first half hour for transient-induced LOOP From ORNL/NRC/LTR-98-11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989 Nonrecovery in the first half hour = 0.48	4.8E-01
	LOOP initiating frequency From ORNL/NRC/LTR-98-11, *Revised LOOP Recovery and PWR Seal LOCA Models*, August 1989 Initiating frequency = 1.63E-5, nonrecovery = 0.36 Duration of unavailability = 6132 h	3.6E-02
RT/LOOP	Reactor trip given a LOOP Failure probability = 0	0
EP	Emergency power system (LOCA and transient-induced LOOP) Failure probability (1 of 2) = Train1 × Train2 × nonrecovery Train 1 = 0.05, Train 2 = 0.057, nonrecovery = 1.0	2.9E-03
	Emergency power system (LOOP) Failure probability (1 of 2) = Train1 × Train2 × nonrecovery Train 1 = 0.05, Train 2 = 0.057, nonrecovery = 0.80	2.3E-03
ESF LOAD-ING	Loading of the safeguards busses Operator failure probability = 0.34	3.4E-01
AFW	Auxiliary Feedwater System Failure probability (1 of 3 + serial failure) = [(Train1 × Train2 × Train3) + Serial] × nonrecovery Train 1 = 0.02, Train 2 = 0.1, Train 3 = 0.05, Serial = 0.00028 Nonrecovery = 0.26	9.9E-05
MFW	Main Feedwater System Failure probability (1 of 1) = Train1 × nonrecovery Train 1 = 0.2, nonrecovery = 0.34	6.8E-02
HPI	High-pressure injection Failure probability (1 of 3) = Train1 × Train2 × Train3 × nonrecovery Train 1 = 0.01, Train 2 = 0.1, Train 3 = 0.3, nonrecovery = 0.84	2.5E-04
HPI (F/B)	High-pressure injection for feed and bleed Failure probability (1 of 3 + operator action) = (Train1 ×Train2 ×Train3 × nonrecovery) + operator action Train 1 = 0.01, Train 2 = 0.1, Train 3 = 0.3 Nonrecovery = 0.84, operator action = 0.01	1.0E-02
HPR	High-pressure recirculation Failure probability (1 of 2 + operator action) = (Train1 × Train2 × nonrecovery) + operator action Train 1 = 0.01, Train 2 = 0.015, nonrecovery = 1.0 Operator action = 0.001	1.1E-03
PORV OPEN	PORV OPEN for feed and bleed Failure probability (1 of 1) = (Train1 × nonrecovery) + operator failure Train 1 = 0.01, nonrecovery = 1.0, Operator failure = 0.0004	1.0E-02
PORV/SRV CHALL.	PORV/SRV challenge rate Challenge rate = 0.04	4.0E-02

Top Event	Description	Value
PORV/SRV RESEAT	PORV/SRV reseat rate Reseat rate = 3.3E-4	3.3E-04
CSR	Containment sump recirculation Failure probability (2 of 4) = [4(Train 1 × Train 2 × Train 3) - 3(Train 1 × Train 2 × Train 3 × Train 4)] × nonrecovery Train 1 = 0.01, Train 2 = 0.03, Train 3 = 0.1, Train 4 = 0.3 Nonrecovery = 1.0	9.3E-05

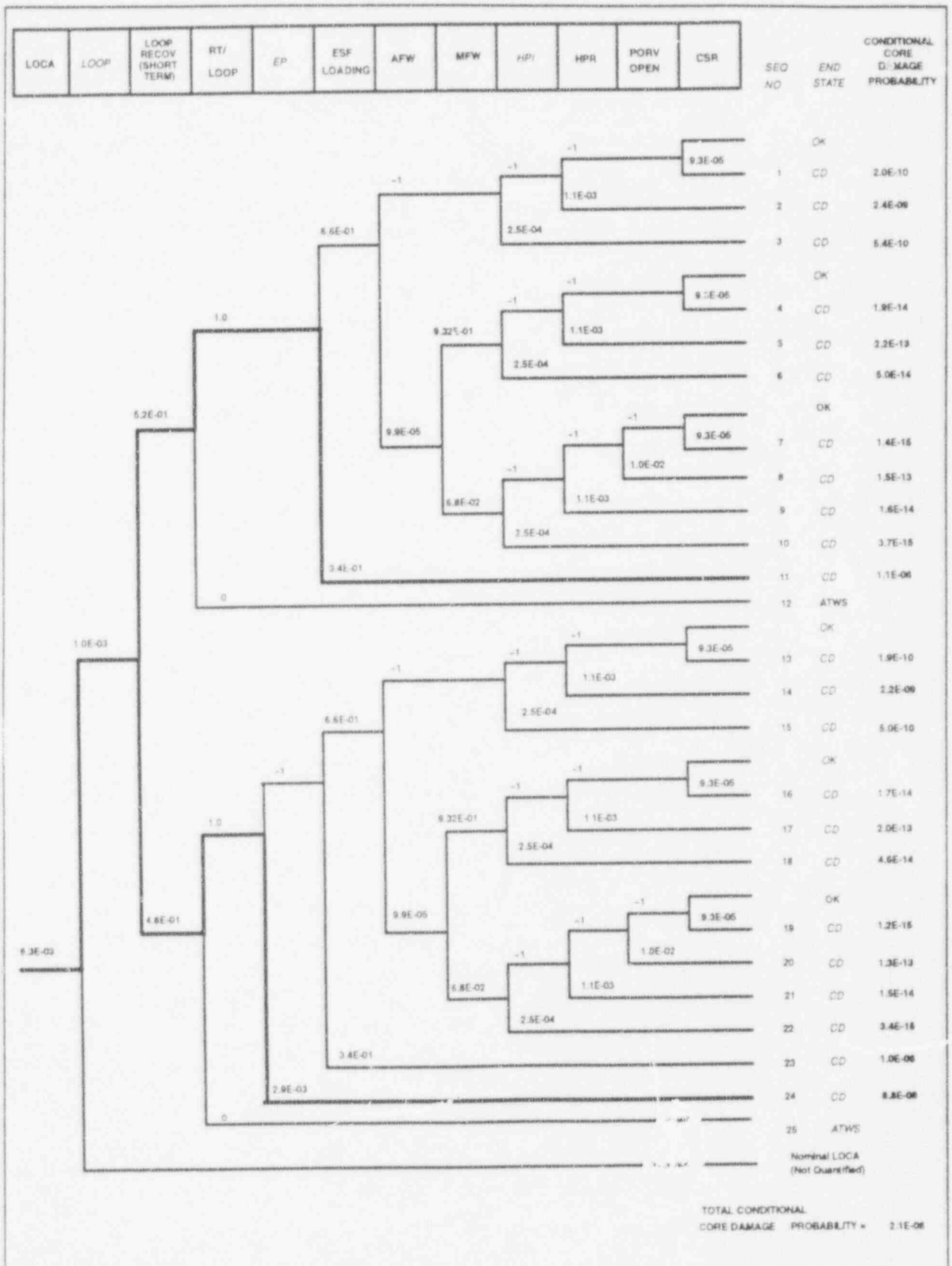

Figure 1. LOCA initiator with transient-induced LOOP Event Tree indicating dominant sequences for LER 412/93-012.

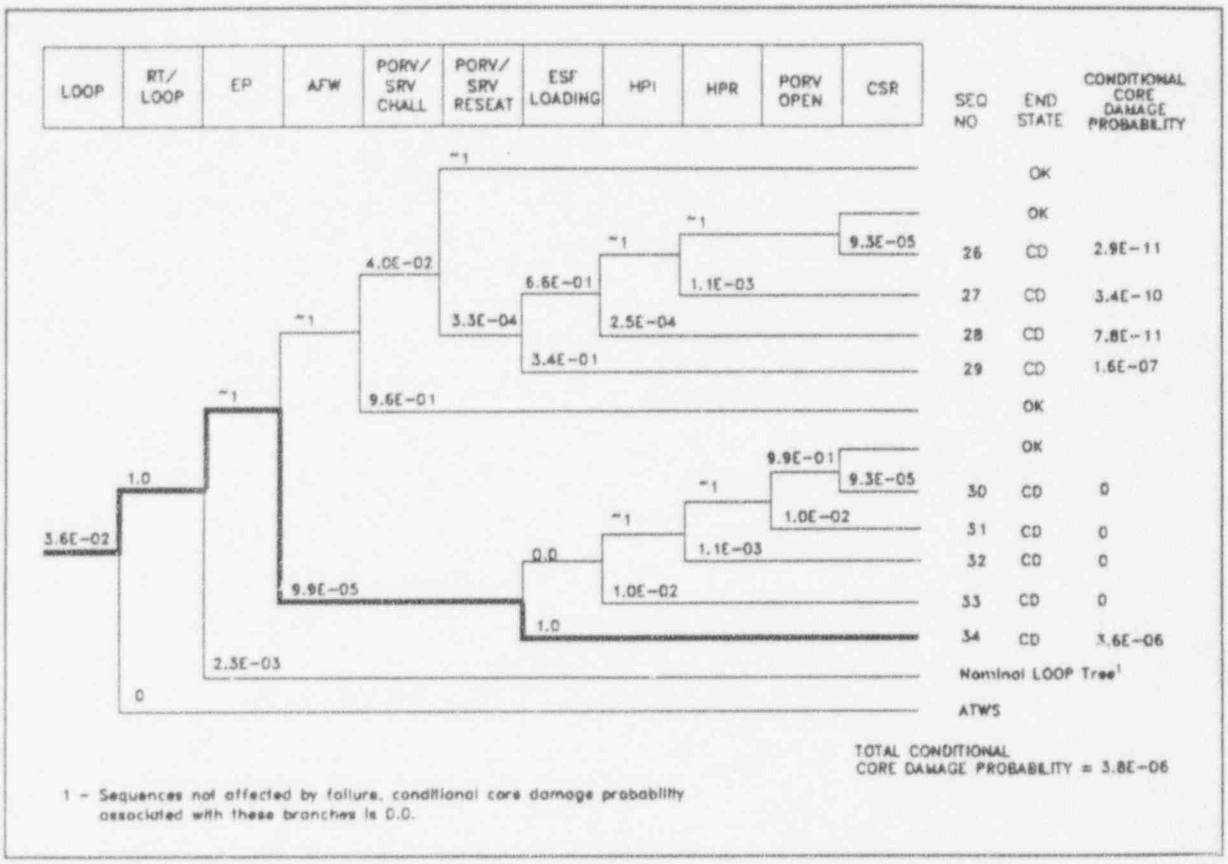

Fig. 2. LOOP Event Tree indicating dominant sequence for LER 412/93-012.

ENCLOSURE 2

GUIDANCE FOR LICENSEE PEER REVIEW OF
PRELIMINARY ASP ANALYSIS

Background

The preliminary precursor analysis of an operational event which occurred at your plant has been provided for your review. This analysis was performed as a part of the NRC's Accident Sequence Precursor (ASP) Program. The ASP Program uses probabilistic risk assessment techniques to provide estimates of operating event significance in terms of the potential for core damage. The types of events evaluated include loss of off-site power (LOOP), Loss-of-Coolant Accident (LOCA), degradation of plant conditions, and safety equipment failures or unavailabilities that could increase the probability of core damage from postulated accident sequences. This preliminary analysis was conducted using the information contained in the plant-specific final safety analysis report (FSAR), individual plant examination (IPE), and the licensee event report (LER) for this event. These sources are identified in the write-up documenting the analysis. The analysis methodology followed the process described in Section 2.1 and Appendix A of Volume 17 of NUREG/CR-4674, copies of which have been provided in this package for your use in this review.

Guidance for Peer Review and Criteria for Recovery Credit

The review of the preliminary analysis should use Section 2.1 and Appendix A of NUREG/CR-4674 for guidance. Comments regarding the analysis should address:

- Characterization of possible plant response,
- Representation of expected plant response used in the analytical models,
- Representation of plant safety equipment configuration and capabilities at the time of the event, and
- Assumptions regarding equipment recovery probabilities.

Any claims for credit for the use of additional systems, equipment, or specific actions in the recovery process must be supported by appropriate documentation in your response. The identified recovery measures must have existed at the time of the event, and should include:

- Normal or emergency operating procedures,
- Piping and instrumentation diagrams (P&IDs),
- Electrical one-line diagrams,
- Results of thermal-hydraulic analysis,
- Operator training (both procedures and simulator), etc.

Also, the documentation should address the impact of the use of the specific recovery measure on:

- The sequence of events,
- The timing of events,
- The probability of operator error in using the system or equipment, and
- Other systems/processes already modeled in the analysis.

For example, Plant A (a PWR) experiences a reactor trip and, during the subsequent recovery, it is discovered that one train of the auxiliary feedwater (AFW) system is unavailable. Absent any further information regrading this event, the ASP Program would analyze it as a reactor trip with one train of AFW unavailable. The AFW train modeling would be patterned after information gathered either from the plant PSAR or the IPE. However, if information is received about the use of an additional system (such as a standby steam generator feedwater system) in recovering from this event, the transient would be modeled as a reactor trip with one train of AFW unavailable, but this unavailability would be mitigated by the use of the standby feedwater system. The mitigation effect for the standby feedwater system would be credited in the analysis provided that the standby feedwater system characteristics are documented in the FSAR, accounted for in the IPE, procedures for using the system during recovery existed at the time of the event, the plant operators had been trained in the use of the system prior to the event, a clear diagram (one-line diagram or better) of the system is available, previous analyses have indicated that there would be sufficient time available to implement the procedure successfully, and results of an assessment that evaluates the effect that use of the standby feedwater system has on already existing processes of procedures that would normally be used to deal with the event are available.

Materials Provided for Review

The following materials have been provided in the package to facilitate your review of the preliminary analysis of the operational event:

- The specific licensee event report (LER), augmented inspection team AIT) report, or other pertinent reports as appropriate (separate enclosure).
- A calculation summary sheet indicating the dominant sequences and pertinent aspects of the modeling details (contained in the analysis writeup).
- An event tree with the dominant sequence(s) highlighted (contained in the analysis writeup).
- A copy of Section 2.1 and Appendix A of NUREG/CR-4674, Volume 17 (separate enclosures).

ENCLOSURE 3

Duquesne Light

Nuclear Group
P.O. Box 4
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Telephone (412) 393-6000

December 6, 1993
ND3MNO:3518

Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, Licensee No. NPF-73
LER 93-012-00

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

In accordance with Appendix A, Beaver Valley Technical Specifications, the following Licensee Event Report is submitted:

LER 93-012-00, 10CFR50.73.a.2.v, and 10CFR50.73.a.2.vi, "Emergency Diesel Generator Sequencer Circuit Deficiencies".

L. R. Freeland
General Manager
Nuclear Operations

JGT/ke

Attachment

JE22

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Beaver Valley Power Station Unit 2	05000 4 1 2	1 OF 06

TITLE (4)
Emergency Diesel Generator Sequencer Circuit Deficiencies

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	06	93	93	0 1 2	0 0	12	06	93	N/A	05000
										05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	000		20.405(a)(1)(i)		50.36(c)(1)	X	50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)	X	50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
L. R. Freeland, General Manager Nuclear Operations	412 643-1258

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	EK	TMR	A611	N					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 11/04/93, a test to verify the automatic loading capability, on a Safety Injection Signal (SIS), for the 2-1 Emergency Diesel Generator (EDG) failed. The Unit was in Cold Shutdown at the time of the testing. The test verifies that all loads will deenergize on the respective safety-related emergency busses and that the EDG sequencer circuitry will automatically load safety-related loads at specfied time intervals, following starting of the EDG. On 11/06/93, the 2-2 EDG also failed its respective test for automatic loading capability. The cause of the test failures was the misoperation of a digital (microprocessor based) solid state timer associated with the Load Sequencer circuitry. An inductive voltage surge was produced by the deenergization of auxiliary relays within the Load Sequencer circuit during the SIS reset of sequencer operation. This caused the timer to misoperate resulting in the failure of the sequencer. Voltage suppression diodes were added to the auxiliary relays within the sequencers' circuit to eliminate the voltage surge. This event constituted a common mode failure which could have safety implications during an event involving a loss of offsite power and safety injection actuation. Operator action may have been required to manually sequence Emergency Diesel Generator loads.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Beaver Valley Power Station Unit 2	05000 4 1 2	9 3	- 0 1 2 -	0 0	02 OF 06

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

DESCRIPTION OF EVENT

On November 4, 1993, with the Unit in Cold Shutdown, a surveillance test to verify the automatic loading capability, on a Safety Injection Signal, for the 2-1 Emergency Diesel Generator failed. The surveillance test, performed on a refueling frequency, verifies that all loads will deenergize on the respective safety-related 4160Volt and 480Volt emergency busses and that the Emergency Diesel Generator circuitry will automatically load the safety-related loads onto the emergency busses at specfied time intervals following starting of the emergency diesel generator. The emergency diesel generator 2-1 sequencer failure was originally determined to have been a safety injection relay malfunction. The safety injection relay was replaced and the emergency diesel generator was re-tested satisfactorily on November 5, 1993.

On November 6, 1993, at 1357 hours, the 2-2 Emergency Diesel Generator failed its respective surveillance test for automatic loading capability. The actual cause for both emergency diesel generator sequencer failures was determined to be the intermittent misoperation of a digital solid state timer relay associated with the individual diesel's load sequencer circuitry. Auxiliary relays within the sequencer developed inductive voltage surges which caused the solid state timer relay circuitry to misoperate, preventing the required contact closures to energize auxiliary relays which would start safety-related loads. The 2-1 and 2-2 emergency diesel generator sequencers were verified to operate correctly in response to an undervoltage condition (Loss of Offsite Power). Since Safety Injection and Containment Isolation Phase "B" (CIB) actuation are not required to function during Cold Shutdown, the 2-1 Emergency Diesel generator was maintained operable by defeating the safety injection and CIB input signals to the diesel generator circuitry.

Post-event bench testing of the digital solid state timer relay identified the intermittent misoperation condition. The misoperation occurred in approximately thirty-three (33) percent of the bench test cycles. Circuit modifications to add voltage transient suppressor diodes in parallel with the auxiliary relay coils (See Figure 1) on both emergency diesel generator sequencer circuits were performed. These voltage suppressor diodes suppress the voltage surge created when the auxiliary relay coil is deenergized.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

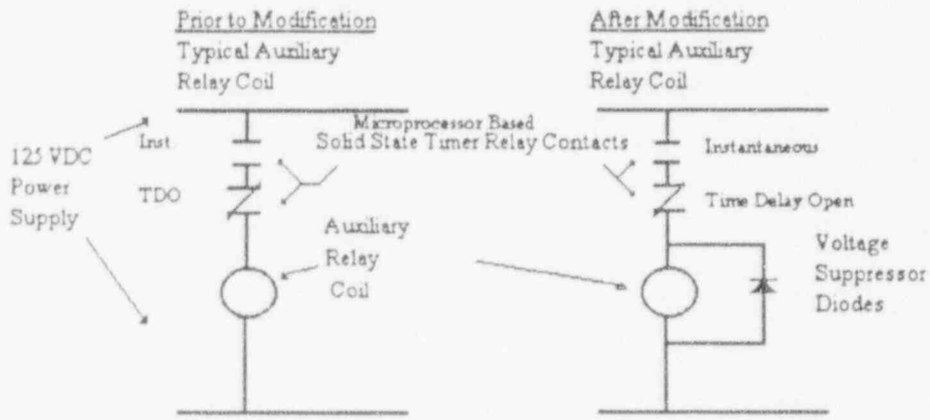


Figure 1

A problem with the start circuit for the 23A Motor Driven Auxiliary Feedwater Pump was identified during post-modification testing of the 2-1 Emergency Diesel Generator Sequencer circuit. Auxiliary relays operate to start safety-related components during sequencer operation. This start circuit has a set of parallel contacts which are closed by various start circuits, (See Figure 2 below). Following these contacts there is another set of parallel contacts associated with the auxiliary relay sequencer circuitry. The contacts for the auxiliary relays in the component start circuits are closed. The sequencer causes one set of parallel contacts to open, effectively blocking component operation until the specified step at the prescribed sequencer time interval. The remaining parallel contact is also closed and is opened by the sequencer timer relay and subsequently re-closed at the specified time interval causing the respective component to start. The addition of the voltage supressor diodes on the auxiliary relay coil for the 23A Motor Driven Auxiliary Feedwater Pump start circuit caused the drop-out time (the length of time required for relay contact opening) to increase a slight amount. This resulted in the relay contacts remaining closed upon initiation of the the first sequencer interval operation. This caused the 23A Motor Driven Auxiliary Feedwater Pump to start earlier in the loading sequence.

NRC FORM 366A
(5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Beaver Valley Power Station Unit 2	05000 4 1 2	9 3	- 0 1 2 -	0 0	04 OF 06

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Prior to Modification

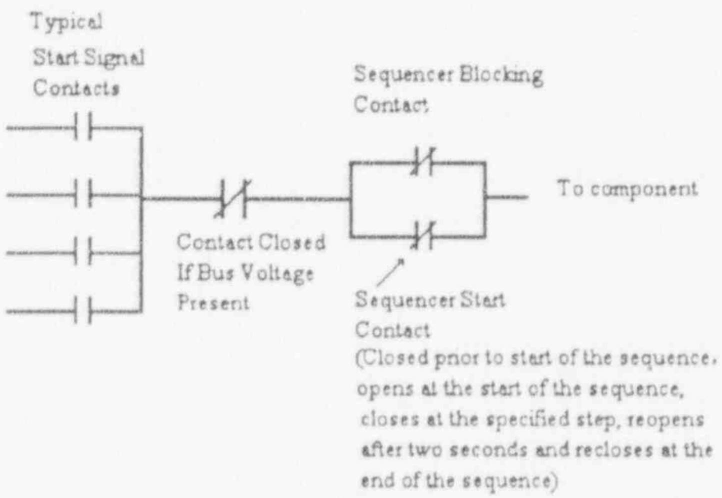


Figure 2

This problem was corrected through additional wiring changes (See Figure 3 Below). This modifiction was performed on the associated circuitry for both motor driven auxiliary feedwater pumps circuitry.

After Modification

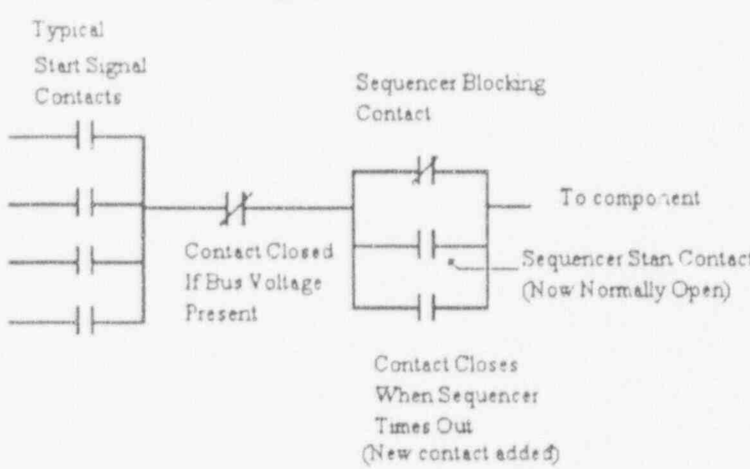


Figure 3

4. Since the digital solid state timers had been purchased commercial grade and qualified for Class 1E use, other solid state relay replacement components in Class 1E circuits that were qualification tested were evaluated to verify that they are qualified for their specific application.

5. An evaluation of the post-modification program practices will be conducted. Until completion of this evaluation, Engineering Assurance and System Engineers will review modification packages prior to installation and concur with the modification testing requirements.

6. Engineering guidelines will be developed which address engineering requirements for the application of digital solid state components as replacements for electro-mechanical or non-solid state components.

SAFETY IMPLICATIONS

This event constituted a common mode failure which could have safety implications during an event involving a loss of offsite power and safety injection actuation. Operator action may have been required to manually start Emergency Diesel Generator loads.

PREVIOUS OCCURRENCES

LER 92-004-00 involved the failure of the Emergency Diesel Generator Sequencer Timer Relays due to the application of excessive voltage to the clock circuit.

DIESEL GENERATOR RELIABILITY

The following is a summary of the past 20, 50 and 100 start and load demands for the Unit 2 emergency diesel generators, trended in accordance with NUMARC 87-00 Rev. 1, Appendix D (Data as of November 6, 1993):

Unit 2

	Start Failures	Load Failures	Total	Trigger
Past 20 Site Demands	0/20	2/20	2/20	3/20
Past 50 Site Demands	0/50	2/50	2/50	4/50
Past 100 Site Demands	0/100	2/100	2/100	5/100
EDG 2-1 Past 25 Demands	0/25	1/25	1/25	4/25
EDG 2-2 Past 25 Demands	0/25	1/25	1/25	4/25

The 2-1 Emergency Diesel Generator was functionally tested and returned to service on November 16, 1993. The 2-2 Emergency Diesel Generator was functionally tested and returned to service on November 17, 1993.

CAUSE OF THE EVENT

The cause for the emergency diesel generator (EDG) failures was identified as inadequate design understanding prior to implementation, and insufficient post modification testing following the installation of the digital (microprocesser based) solid state timers. The design changes were made to the EDG Load Sequencers during the Second Refueling Outage and also following digital solid state timer circuit modifications performed during the Third Refueling Outage. The testing conducted did not adequately validate the design change from electro-mechanical to microprocessor based solid state timers. An inductive voltage surge was produced by the deenergization of auxiliary relays within the load sequencer circuitry which caused the solid state timer to misoperate. The timer relays are Automatic Timing & Controls Company, Model 365-A, Long Range Timers. These timers were recommended by the manufacturer as direct replacements for the original electo-mechanical timers, where improved timing accuracy is desired.

REPORTABILITY

This event was reported to the Nuclear Regulatory Commission on November 6, 1993, at 1527 hours, in accordance with 10CFR50.72.b.2.i, as a condition found while the reactor is shut down, that had it been found while the reactor was in operation, would have resulted in the nuclear power plant being in an unanalyzed condition. This written report is being submitted in accordance with 10CFR50.73.a.2.v, as a condition that alone could have prevented the fulfillment of the safety function of systems that are needed to mitigate the consequences of an accident. Additionally is it being reported in accordance with 10CFR50.73.a.2.vii, as an event where a single cause or condition caused at least one train to become inoperable in a single system designed to mitigate the consequences of an accident.

CORRECTIVE ACTIONS

The following corrective actions have been taken as a result of this event:

1. Voltage supression diodes have been added to the auxiliary relays to eliminate the effects of the voltage surge.

2. The motor-driven auxiliary feedwater pump start circuitry has been modified to ensure correct operation during the emergency diesel generator sequencer operation.

3. The 2-1 Emergency Diesel Generator was functionally tested and returned to service on November 16, 1993. The 2-2 Emergency Diesel Generator was functionally tested and returned to service on November 17, 1993.

ENCLOSURE 4

2.0 ACCIDENT SEQUENCE PRECURSOR IDENTIFICATION AND QUANTIFICATION

2.1 Accident Sequence Precursor Identification

The ASP Program is concerned with the identification and documentation of operational events that have involved portions of core damage sequences, and with the estimation of frequencies and probabilities associated with them.

Identification of precursors requires the review of operational events for instances in which plant functions that provide protection against core damage have been challenged or compromised. For core damage to occur, fuel temperature must increase. Such an increase requires the heat generation rate in the core to exceed the heat removal rate. This can result from either a loss of core cooling or excessive core power. The following functions are provided at all plants to protect against these two conditions:

- Reactor subcriticality. The reactor must be placed in a subcritical condition, normally by inserting control rods into the core to terminate the chain reaction.
- Reactor coolant inventory makeup. Sufficient water must be provided to the reactor coolant system (RCS) to prevent core uncovery.
- RCS integrity. Loss of RCS integrity requires the addition of a significant quantity of water to prevent core uncovery.
- Decay heat removal (DHR). Heat generated in the core by fission product decay must be removed.
- Containment integrity. Containment integrity (containment heat removal, isolation, and hydrogen control) is not addressed in the precursor analyses unless core DHR capability is impacted.

System-based event trees were developed to model potential sequences to core damage. The event trees are specific to eight plant classes so as to reflect differences in design among plants in the U.S. LWR population. Three initiators are addressed in the event trees: trip [which includes loss of main feedwater (LOFW) within its sequences], loss of offsite power (LOOP), and small-break loss-of-coolant accident (LOCA). These three initiators are primarily associated with loss of core cooling. [Excessive core power associated with anticipated transient without scram (ATWS) is represented by a failure-to-trip sequence but is not developed.] Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with these initiators. Detailed descriptions of the plant classification scheme and the event tree models are included in Appendix A. Operational events that cannot be associated with one of these initiators are accommodated by unique modeling.

Armed with a knowledge of the primary core damage initiator types plus the systems that provide protection against core damage (based on the event tree models), ASP Program staff members examine LERs to determine the impact of operational events on potential core damage sequences. While the sequences detailed on the event tree models do not describe all possible paths to core damage, they form a primary basis for selecting an operational event as a precursor. Operational events are also reviewed in a more general sense for their impact on the protective functions described above.

Identification of precursors within a set of LERs involved a two-step process. First, each LER was reviewed by two experienced engineers to determine if the reported event should be examined in detail. This initial review was a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that were clearly unimportant. This was done by eliminating events that satisfied pre-defined criteria for rejection and accepting all others as potentially significant and requiring analysis. In some cases, events are impractical to analyze due to lack of information or inability to reasonably model within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models. Events also were eliminated from further review if they had little impact on core damage sequences or provided little new information on the risk impacts of plant operation. Such events included single failures in redundant systems and uncomplicated reactor trips and LOFWs. Any event with an impact that can be mapped onto the ASP core damage models can, in principle, be assessed.

LERs were eliminated from further consideration as precursors if they involved at most one of the following:

- a component failure with no loss of redundancy,
- a loss of redundancy in only one system,
- a seismic design or qualification error,
- an environmental design or qualification error,
- a structural degradation,
- an event that occurred prior to initial criticality (since the core is not considered vulnerable to core damage at this time and since distinguishing initial testing failures from operational failures is difficult),
- a design error discovered by reanalysis,
- an event impact bounded by a reactor trip or LOFW,
- an event with no appreciable impact on safety systems, or
- an event involving only post-core damage impacts (selected containment-related events are documented).

Events identified for further consideration typically included

- unexpected core damage initiators (LOOP and small-break LOCA);
- all events in which reactor trip was demanded and a safety-related component failed;
- all support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- any event where two or more failures occurred;
- any event or operating condition that was not predicted or that proceeded differently from the plant design basis; and
- any event that, based on the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Operational events that were not eliminated in the first review received a more extensive analysis to identify those events considered to be precursors to potential severe core damage accidents either because of an initiating event or because of failures that could have affected the course of postulated off-normal events or accidents. These detailed reviews were not limited to the LERs; they also used final safety analysis reports (FSARs), their amendments, and other information available at the Nuclear Operations Analysis Center.

The detailed review of each event considered (l) the immediate impact of an initiating event or (2) the potential impact of the equipment failures or operator errors on readiness of systems in the plant for mitigation of off-normal and accident conditions.

In the review of each selected event, three general scenarios (involving both the actual event and postulated additional failures) were considered:

1. If the event or failure was immediately detectable and occurred while the plant was at power, then the event was evaluated according to the likelihood that it and the ensuing plant response could lead to severe core damage.
2. If the event or failure had no immediate effect on plant operation (i.e., if no initiating event occurred), then the review considered whether the plant would require the failed items for mitigation of potential severe core damage sequences should a postulated initiating event occur during the failure period.
3. If the event or failure occurred while the plant was not at power, then the event was first evaluated according to whether it could have occurred while at power or at hot shutdown immediately following power operation. If the event could only occur at cold shutdown, then its impact on continued DHR was assessed.

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER, the sequence of operation of various mitigating systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if they included one of the following attributes that impacted core damage sequences and if the conditional probability of subsequent core damage (described later) was at least 1.0×10^{-6}

- an unexpected core damage initiator (such as a LOOP, steam-line break (SLB), or small-break LOCA);
- a failure of a system (all trains of a multiple train system) required to mitigate the consequences of a core damage initiator,
- concurrent degradation in more than one system required to mitigate the consequences of a core damage initiator, or
- a transient or LOFW with a degraded mitigating system.

Events of low significance are thus excluded, allowing the reader to concentrate on the more important events. This approach is consistent with the approach used to define 1987-1991 precursors, but is different from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria, regardless of conditional core damage probability.

Events that occurred in 1992 were reviewed for precursors only if they satisfied an initial significance screening. This approach, which was similar to that used in the review of 1988-1991 events, eliminated many insignificant events from review and permitted some increase in the amount of documentation provided for precursors. Two approaches were used to select events to be reviewed for precursors.

First, events were reviewed for precursors if they were identified as significant by the Nuclear Regulatory Commission's (NRCs) Office for Analysis and Evaluation of Operational Data (AEOD). AEOD's screening process identifies operating occurrences involving, in part,

- violation of a safety limit;
- an alert or higher emergency classification;
- an on-demand failure of a safety system (except surveillance failures);
- events involving unexpected system or component performance with serious safety significance or generic implications;
- events where improper operation, maintenance, or design causes a common-mode/common-cause failure of a safety system or component, with safety significance or generic implications;
- safety-significant system interactions;
- events involving cognitive human errors with safety significance or generic implications;
- safety-significant events involving earthquakes, tornadoes, floods, and fires;
- a scram, transient, or engineered safety features (ESF) actuation with failure or inoperability of required equipment;
- on-site work-related or nuclear-incident-related death, serious injury, or exposure that exceeds administrative limits;
- unplanned or unmonitored releases of radioactivity, or planned releases that exceed Technical Specification limits; and
- infrequent or moderate frequency events.

AEOD-designated significant events also involve operating conditions, where a failure or accident has not occurred but where the potential for such an event is identified.

Second, LERs were also reviewed if they were identified through a computerized search using the sequence coding and search system (SCSS) data base of LERs. This computerized search identified LERs potentially involving (1) failures in plant systems that provided the protective functions described earlier and (2) initiating events addressed in the ASP models. Based on a review of the 1984-87 precursor evaluations, this computerized search successfully identifies almost all precursors within a subset of approximately one-third of all LERs.

While review of LERs identified by AEOD and through the use of SCSS is expected to identify almost all precursors, it is possible that a few precursors exist within the set of unreviewed LERs. Some potential precursors that would have been found if all 1992 LERs had been reviewed may not have been identified. Because of this (plus modeling changes that impact precursor probability somewhat), it should not be assumed that the set of 1988-92 precursors is consistent with precursors identified in 1984-87.

Following AEOD and SCSS computerized screening, 1022 LERs from 1992 were reviewed for precursors. Twenty-seven operational events with conditional probabilities of subsequent severe core damage greater than 1.0×10^{-6} were identified as accident sequence precursors.

Individual failures of boiling-water reactor (BWR) high-pressure coolant injection (HPCI), high-pressure core spray (HPCS), and reactor core isolation cooling (RCIC) systems (all single-train systems), and trips and LOFWs without additional mitigating system failures were not selected as precursors. The impact of such events was determined on a plant-class basis. The results of these evaluations are provided in Appendix A.

In addition to accident sequence precursors, events involving loss of containment functions — containment cooling, containment spray, containment isolation (direct paths to the environment only), and hydrogen control — were identified in the review of 1992 LERs. Other events that were not selected as precursors but that provided insight into unusual failure modes with the potential to compromise continued core cooling are also identified. Events identified as precursors are documented in Appendix B, the containment-related events are documented in Appendix C, events considered "interesting" are documented in Appendix D, and events that were determined to be impractical to analyze are documented in Appendix E.

2.2 Estimation of Precursor Significance

Quantification of ASP significance involves determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto the ASP event trees, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system failure given the operational event.

In the precursor quantification process, it is assumed that the failure probabilities for systems observed to have failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Failure probabilities for systems observed to have been degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. The failure probabilities associated with observed successes and with systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities, with consideration of the potential for recovery. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage that remains once the observed failures have occurred.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the LWR population, even though they are applied to sequences that are plant-class specific in nature. Because of this, the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred.

The evaluation of precursor events in this report consider and, where appropriate, give credit for additional equipment or recovery procedures the plants have recently added. Accordingly, the evaluations this year may not be directly comparable to the results of prior years. Examples of additional equipment and recovery procedures addressed in the 1992 analyses, when information was available, include use of supplemental diesel generators (DGs) for station blackout mitigation, alternate systems for steam generator (SG) and RCS makeup, and depressurization of the primary with low pressure injection (LPI) in lieu of high pressure injection (HPI).

The ASP calculational process is described in detail in Appendix A. This appendix documents the event trees used in the 1988-1992 precursor analyses, changes to these trees from prior years, the approach used to estimate event tree branch and sequence probabilities, and sample calculations; it also provides probability values used in the calculations. The overall precursor selection process is illustrated in Fig. 1.

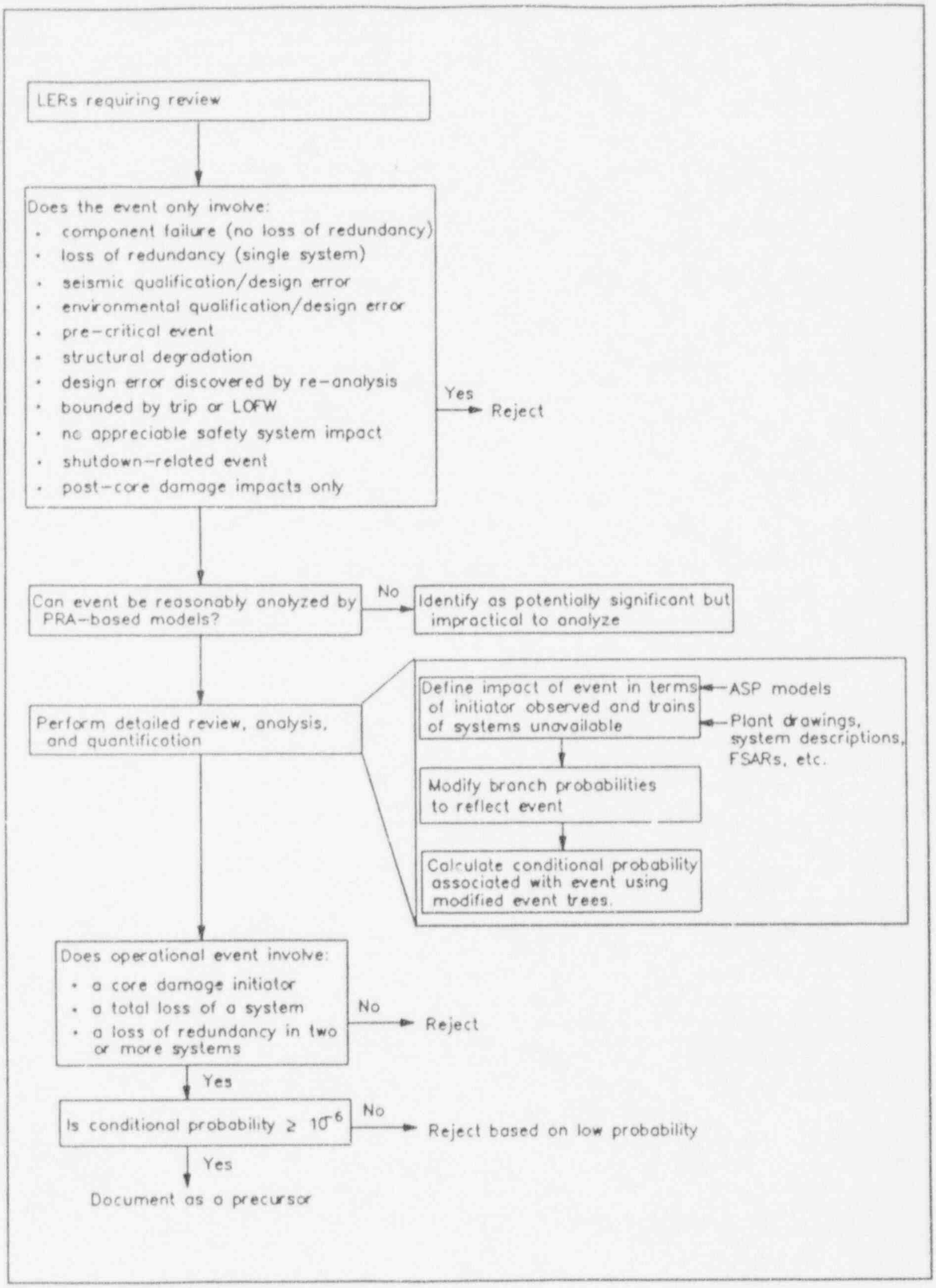


Fig. 1. ASP analysis process.

ENCLOSURE 5

APPENDIX A. ASP MODELS

A. ASP MODELS

This appendix provides information concerning the methods and models used to estimate event significance in the ASP Program. The basic models used in the analysis of 1992 precursors are the same as those used for 1989-91 precursors. However, the analysis of 1992 precursors considered the potential use of alternate equipment and procedures, beyond that addressed in the basic models, that recently have been added by the licensees to provide additional protection against core damage, if information regarding this equipment was available. This equipment is described in Sect. A.3.

A.1 Precursor Significance Estimation

Quantification of accident sequence precursor significance involves determination of a conditional probability of subsequent severe core damage given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto event trees depicting potential paths to severe core damage and calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. In the quantification processes, it is assumed that the event tree branch failure probabilities for systems observed failed during an event are equal to the likelihood of not recovering from the failure or fault that actually occurred. Event tree branch failure probabilities for systems observed degraded during an operational event are assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. Event tree branch failure probabilities used for systems observed to be successful and systems unchallenged during the actual occurrence are assumed equal to a failure probability estimated from either system failure data (when available) or by the use of system success criteria and typical train and common-mode failure probabilities. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage remaining once the observed failures have occurred.

A.1.1 ASP Event Tree Models

Models used to rank precursors as to significance consist of plant-class specific event trees that are linked to simplified plant-specific system models. These models describe mitigation sequences for three initiating events: a nonspecific reactor trip [which includes LOFW within the model], LOOP, and small-break LOCA. The event tree models are system-based and include a model applicable to each of eight plant classes: three for BWRs and five for PWRs.

Plant classes are defined based on the use of similar systems in providing protective functions in response to transients, LOOPs, and small-break LOCAs. System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar in response. Plants where certain mitigating systems do not exist, but which are largely analogous in their initiator response, are grouped into the appropriate plant class. In modeling events at such plants, the event tree branch probabilities are modified to reflect the actual systems available at the plant. For operational events that cannot be described using the plant-class specific event trees, unique models are developed to describe the potential sequences to severe core damage.

Each event tree includes two undesired end states. The undesired end states are designated as (1) core damage (CD), in which inadequate core cooling is believed to exist; and (2) ATWS, for the failure-to-scram sequence. The end states are distinct; sequences associated with ATWS are not subsets of core damage sequences. The ATWS sequence, if fully developed, would consist of a number of sequences ending in either success or core damage. Successful operation is designated "OK" in the event trees included in this appendix.

A.1.2 Precursor Impact on Event Tree Branches

The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of system failure given the operational event. This translation process is simplified in many cases through the use of train-based models that represent an event tree branch. If a train-based model exists, then the impact of the operational event need only be determined at the train level, and not at the system level.

Once the impact of an operational event on systems included in the ASP event tree models has been determined, branch probability values are modified to reflect the event, and the event trees are then used to estimate a conditional probability of subsequent core damage, given the precursor.

A.1.3 Estimation of Initiating Event Frequencies and Branch Failure Probabilities Used with the Event Tree Models

A set of initiating event frequencies and system failure probabilities was developed for use in the quantification of the event tree models associated with the precursors. The approach used to develop frequency and probability estimates employs failure or initiator data in the precursors themselves when sufficient data exists. When precursor data are available for a system, its failure probability is estimated by counting the effective number of nonrecoverable failures in the observation period, making appropriate demand assumptions, and then calculating the effective number of failures per demand. The number of demands is calculated based on the estimated number of tests per reactor year plus any additional demands to which a system would be expected to respond. This estimate is then multiplied by the number of applicable reactor years in the observation period to determine the total number of demands. A similar approach is employed to estimate initiator frequencies per reactor year from observed initiating events.

The potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes are currently used to describe the different types of recovery that could be involved:

Recovery class	Likelihood of nonrecovery	Recovery characteristic
R1	1.00	The failure did not appear to be recoverable in the required period, either from the control room or at the failed equipment.
R2	0.34	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the control room did not appear possible.
R3	0.12	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.04	The failure appeared recoverable in the required period from the control room and was considered routine and procedurally based.

The assignment of an event to a recovery class is based on engineering judgment, which considers the specifics of each operational event and the likelihood of not recovering from the observed failure in a moderate to high-stress situation following an initiating event. For analysis purposes, consistent probabilities of failing to recover an observed failure are assigned to each event in a particular recovery class. It must be noted that the actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values listed. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.*

The branch probability estimation process is illustrated in Table A.1. Table A.1 lists two operational events that occurred in 1984-86 involving failure of SG isolation. For each event, the likelihood of failing to recover from the failure is listed (Column 3). The effective number of nonrecoverable events (1.04 in this case) is then divided by an estimate of the total number of demands in the 1984-86 observation period (1968) to calculate a failure on demand probability of 5.3×10^{-4}.

The likelihood of system failure as a result of hardware faults is combined with the likelihood that the system could not be recovered, if failed, and with an estimate of the likelihood of the operator failing to initiate the system, if manual initiation were required, to estimate the overall failure probability for an event-tree branch. Calculated failure probabilities are then used to tailor the probabilities associated with train-based system models. Such an approach results in system failure probability estimates that reflect, to a certain extent, the degree of redundancy actually available and permits easy revision of these probabilities based on train failures and unavailabilities observed during an operational event.

*Programmatic constraints have prevented substantial efforts in estimating actual recovery class distributions. The values currently used were developed based on a review of events with the potential for short-term recovery, in addition to consideration of human error during recovery. These values have been reviewed both within and outside the ASP Program. While it is acknowledged that substantial uncertainty exists in them, they are believed adequate for ranking purposes, which is the primary goal of the current precursor calculations. This assessment is supported by the sensitivity and uncertainty calculations documented in the 1980-81 report. These calculations demonstrated little impact on the relative ranking of events from variance in recovery class values.

A.1.4 Conditional Probability Associated with Each Precursor

The calculation process for each precursor involves a determination of initiators that must be modeled and their probability, plus any modifications to system probabilities necessitated by failures observed in an operational event. Once the branch probabilities that reflect the conditions of the precursor are established, the sequences leading to the modeled end states (core damage and ATWS) are calculated and summed to produce an estimate of the conditional probability of each end state for the precursor. So that only the additional contribution to risk (incremental risk) associated with a precursor is calculated, conditional probabilities for precursors associated with equipment unavailabilities (during which no initiating event occurred) are calculated a second time using the same initiating event probability but with all branches assigned normal failure probabilities (no failed or degraded states) and subtracted from the initially calculated values. This eliminates the contribution for sequences unimpacted by the precursor, plus the normal risk contribution for impacted sequences during the unavailability. This calculational process is summarized in Table A.2.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the LWR population, even though they are applied to sequences that are plant-class specific in nature. Because of this, *the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred.* The probabilities calculated in the ASP study are homogenized probabilities considered representative of probabilities resulting from the occurrence of the selected events at plants representative of the plant class.

A.1.5 Sample Calculations

Three hypothetical events are used to illustrate the calculational process.

1. The first event assumes a trip and LOFW but no other observed failures during mitigation. An event tree for this event is shown in Fig. A.1. On the event tree, successful operation is indicated by the upper branch and failure by the lower branch. With the exception of relief valve lift, failure probabilities for branches are indicated. For HPI, the lowest branch includes operator action to initiate feed and bleed. Success probabilities are 1 - p(failure). The likelihood of not recovering the initiator (trip) is assumed to be 1.0, and the likelihood of not recovering MFW is assumed to be 0.34 in this example. Systems assumed available were assigned failure probabilities currently used in the ASP Program. The estimated conditional probabilities for undesirable end states associated with the event are then:

$$p(cd) = p[\text{seq. } 11] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}]$$

$$+\ p[\text{seq. } 12] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times (1 - 9.9 \times 10^{-5}) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times 8.4 \times 10^{-4}]$$

$$+\ p[\text{seq. } 13] \quad [1.0 \times (1 - 3.0 \times 10^{-5}) \times 9.9 \times 10^{-5} \times (1 - 0.34) \times 4.0 \times 10^{-2} \times 3.3 \times 10^{-4} \times (1.0 - 8.4 \times 10^{-4}) \times 1.1 \times 10^{-3}]$$

$$+ \; p[\text{seq. } 14] + p[\text{seq. } 15] + p[\text{seq. } 16] + p[\text{seq. } 17]$$

$$= 7.7 \times 10^{-7}$$

$$p(\text{ATWS}) = p[\text{seq. } 18]$$

$$= 3.0 \times 10^{-5}$$

2. The second example event involves failures that would prevent HPI if required to mitigate a small-break LOCA or if required for feed and bleed. Assume such failures were discovered during testing. This event impacts mitigation of a small-break LOCA initiator and potentially impacts mitigation of a trip and LOOP, should a transient-induced LOCA occur or should feed and bleed be required upon loss of AFW and MFW. The event tree for a postulated small-break LOCA associated with this example precursor is shown in Fig. A.2. The failure probability associated with the precursor event (unavailability of HPI) is assigned based on the likelihood of not recovering from the failure in a 20-30 min time frame (assumed to be 1.0 in this case). No initiating event occurred with the example precursor; however, a failure duration of 360 h was estimated based on one-half of a monthly test interval. The estimated small-break LOCA frequency (assumed to be 1.0×10^{-6}/h in this example), combined with this failure duration, results in an estimated initiating event probability of 3.6×10^{-4} during the unavailability. The probabilities for small-LOCA sequences involving undesirable end states (employing the same calculational method as above and subtracting the nominal risk during the time interval) are 3.6×10^{-4} for core damage and 0.0 for ATWS. Note that the impact of the postulated failure on the ATWS sequence is zero because HPI success or failure does not impact that sequence as modeled.

For most unavailabilities, similar calculations would be required using the trip and LOOP event trees, since these postulated initiators could also occur. In this example, neither of these two initiators contributes substantially to the core damage probability associated with the event.

3. The third example event involves a trip with unavailability of one of two trains of service water (SW). Assumed unavailability of the SW train results in unavailability of one train of HPI, high-pressure recirculation (HPR), and AFW, all because of unavailability of cooling to the respective pumps. In this example, SW cooling of two motor-driven AFW pumps is assumed. An additional turbine-driven pump is assumed to be self-cooled. Since SW is not explicitly addressed in the ASP event trees, the probabilities of front-line systems impacted by the loss of SW are instead modified.

Figure A.3 shows a transient event tree with branch failure probabilities modified to reflect unavailability of one train of service water. The likelihoods of not recovering failed front line systems are assumed to be unchanged, since the failure mechanisms for (observed) non-faulted trains are expected to be consistent with historically observed failures. The conditional probability of core damage given the trip and one service water train unavailable is 1.1×10^{-6}. If the second train of service water were to fail, HPI and HPR (and hence feed and bleed) would be rendered unavailable; however, the turbine-driven AFW pump would still be operable. In this case, the likelihood of not recovering HPI and HPR is assumed to be 1.0 until service water is recovered. Sequences associated with loss of both service water trains increase the core damage probability associated with the event. The extent of this increase is dependent in PWRs on the likelihood of a reactor coolant pump seal failure following the loss of service water (since seal injection and seal cooling would be typically lost). Assuming that the conditional probability of loss of the second service water train is 0.01, that the likelihood of not recovering SW is 0.34, and that the failure probability of the turbine-driven AFW pump is 0.05, the increase in core damage probability is 1.7×10^{-4} if no RCP seal failure occurs, and 3.4×10^{-3} if the likelihood of seal failure is 1.0.

A.1.6 Event Tree Changes Made to 1988–1991 Event Models

Two changes were made to the event trees used in the 1988-91 precursor assessments: core vulnerability sequences on trees used for 1984-87 assessments were reassigned as success or core damage sequences, and the likelihood of PWR RCP seal LOCA following station blackout was explicitly modeled.

In the prior models, the core vulnerability end state was assigned to sequences in which core protection was expected to be provided but for which no specific analytic basis was generally available or which involved non-proceduralized operator actions. Core vulnerability sequences were assigned to either success or core damage end states in the current models, as follows:

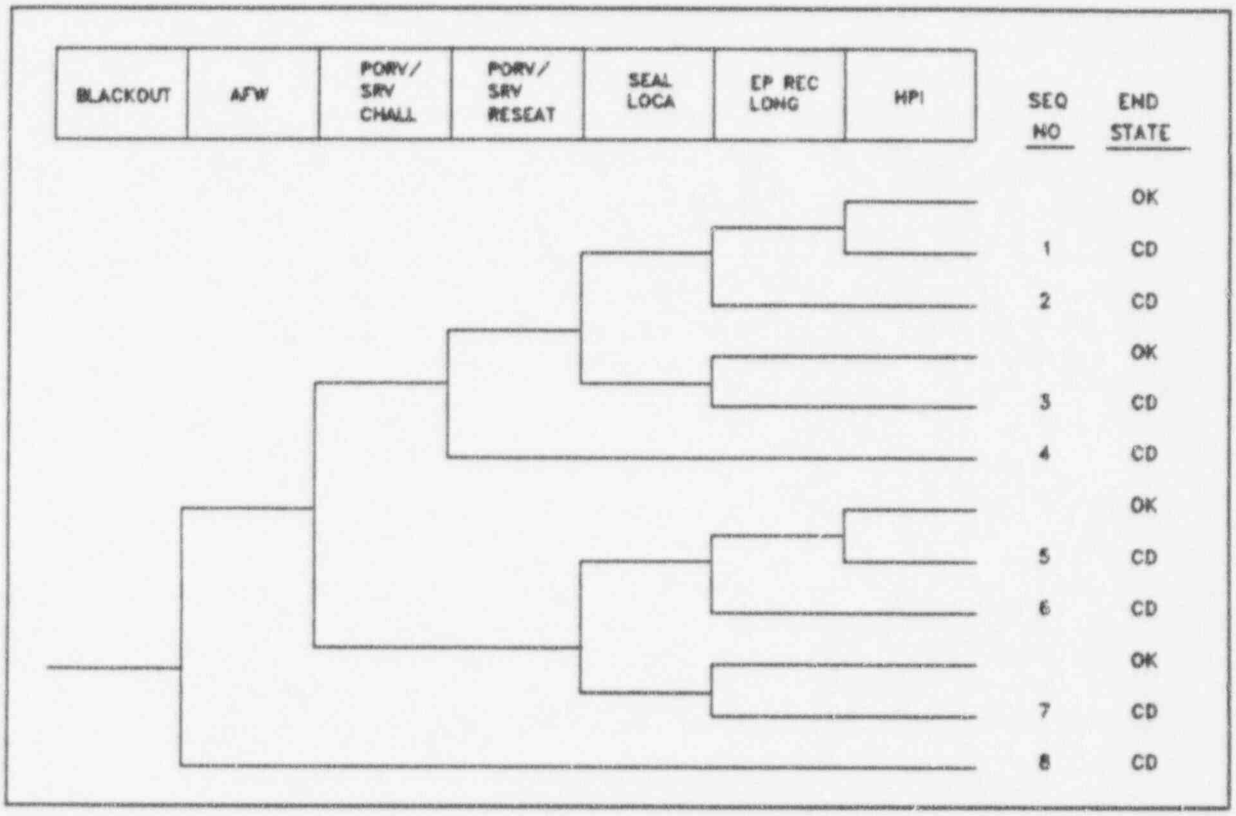
Core vulnerability sequence type	Revised end state
Stuck-open secondary-side relief valve with a failure of HPI in a PWR	Success
Steam generator (SG) depressurization and use of condensate system following failure of AFW, MFW, and feed and bleed in a PWR	Core damage (except for PWR Class H)
Use of containment venting as an alternate core cooling method in a BWR	Core damage

The net effect of this change is a significant reduction in the complexity of the event trees, with little impact on the relative significance estimated for each precursor. The impact of this modeling change on conditional probability estimates for 1987 precursors is described in Sect. 3.6 of Ref. 1. (Alternate calculations using models with the above changes were performed on 1987 events.) As illustrated in Ref. 1, modest differences existed between the core damage, core damage plus core vulnerability, and revised core damage model conditional probability estimates for most of the more significant events. Where differences did exist, the sum of probabilities of core damage and core vulnerability (all non-ATWS undesirable end states in the earlier models) was closer to the core damage probability estimated with the revised models.

Three 1987 events had substantially higher "sum" probabilities—these events involved trips with single safety-related train unavailabilities, for which the dominant core vulnerability sequence was a stuck-open secondary-side relief valve with HPI failure (assigned to success in the revised models).

The second modeling change was the inclusion of PWR RCP seal LOCA in blackout sequences. The impact of such a seal LOCA on the core damage probability estimated for an event had previously been bounded by the use of a conservative value for failure to recover ac power prior to battery depletion following a LOOP and loss of emergency power.

The PWR event trees have been revised to address potential seal LOCA during station blackout through the use of seal LOCA and electric power recovery branches, as shown below:



Two time periods are represented in the sequences in the above figure. Auxiliary feedwater, power-operated relief valve/safety relief valve (PORV/SRV) challenge, and PORV/SRV reseat are shot-term responses following loss of the diesel generators. If turbine-driven AFW is unavailable, or if an open PORV/SRV fails to close, then core damage is assumed to occur, since no high-pressure injection is available as an alternate means of core cooling or for RCS makeup. SEAL LOCA, EP REC LONG, and HPI are branches applicable in the long term. SEAL LOCA represents the likelihood of a seal LOCA prior to restoration of ac power. EP REC LONG represents the likelihood of not restoring ac power prior to core uncovery (if a seal LOCA exists) or prior to battery depletion (in the case of no seal LOCA). Once the batteries are depleted, core damage is assumed to occur, since control of turbine-driven pumps and the ability to monitor core and RCS conditions are lost. HPI represents the likelihood of failing to provide HPI following a seal LOCA to prevent core damage. The ASP models have been simplified somewhat by assuming that HPI is always adequate to make up for flow from a failed seal or seals.

The three seal LOCA-related sequences are illustrated in sequences 1, 2, and 3. In sequence 1, a seal LOCA occurs prior to restoration of ac power, ac power is successfully restored prior to core uncovery, but HPI fails to provide makeup flow. In sequence 2, a seal LOCA also occurs, and ac power is not restored prior to core uncovery. In sequence 3, no seal LOCA occurs, but ac power is not recovered prior to battery depletion. The likelihood of seal LOCA prior to ac power restoration and the likelihood of ac power recovery are time-dependent, and this time-dependency is accounted for in the analysis. A more detailed description of the changes associated with explicitly modeling RCP seal LOCA is included in Ref. 2.

In addition to elimination of core vulnerability sequences, two other changes were made to simplify the previously complex BWR event trees:

- Failure to trip with soluble boron injection success was previously developed in detail and involved a large number of low probability sequences. All failure to trip sequences are now assigned to the ATWS end state.

- The condensate system was previously modeled as an alternate source of low-pressure injection water. This use of the condensate system is now considered a recovery action. this reduces the number of sequences on the event trees without substantially impacting the core damage probability estimates developed using the trees. Systems addressed on the event trees for low-pressure injection include LPCS, LPCI, and RHRSW.

A.2 Plant Categorization

Both the 1969 – 79 and 1980 – 81 precursor reports (Refs. 1 and 2) used simplified, functionally based event trees to model potential event sequences. One set of event trees was used to model for PWR initiating events: LOFW, LOOP, small-break LOCA, and steam line break. A separate set of event trees was used to model BWR response to the same initiators. Operational events that could not be modeled using these "standardized" event trees were addressed using models specifically developed for the event.

It was recognized during the review of the 1969-79 precursor report that plant designs were sufficiently different that multiple models would be required to more correctly describe the impact of an operational event in different plants. In 1985, substantial effort was expended to develop a categorization scheme for all U.S. LWRs that would permit grouping of plants with similar response to a transient or accident at the system or functional level, and to subsequently develop eight sets of plant-class specific event tree models. Much of the categorization and early event sequence work was done at the University of Maryland (Refs. 3 and 4). The ASP Program has generally employed these categorizations; however, some modifications have been required to reflect more closely the specific needs of the precursor evaluations.

In developing the plant categorizations, each reactor plant was examined to determine the systems used to perform the following plant functions required in response to reactor trip, LOOP, and small-break LOCA initiators to prevent core damage: reactor subcriticality, RCS integrity, reactor coolant inventory, short-term core heat removal, and long-term core heat removal.

Functions related to containment integrity (containment overpressure protection and containment heat removal) and post-accident reactivity removal are not included on the present ASP event trees (which only concern core damage sequences) and are not addressed in the categorization scheme.

For each plant, systems utilized to perform each function were identified. Plants were grouped based on the use of nominally identical systems to perform each function; that is, systems of the same type and function without accounting for the differences in the design of those systems.

Three BWR plant classes were defined. BWR Class A consists of the older plants, which are characterized by isolation condensers (ICs) and feedwater coolant injection (FWCI) systems that employ the MFW pumps. BWR Class B consists of plants that have ICs but a separate HPCI system instead of FWCI. BWR Class C includes the modern plants that have neither ICs nor FWCI. However, they have a RCIC system that Classes A and B lack. The Class C plants could be separated into two subgroups, those plants with turbine-driven HPCI systems and those with motor-driven HPCS systems. This difference is addressed instead in the probabilities assigned to branches impacted by the use of these different system designs.

PWRs are separated into five classes. One class represents most Babcock & Wilcox Company plants (Class D). These plants have the capability of performing feed and bleed without the need to open the PORV. Combustion Engineering plants are separated into two classes, those that provide feed and bleed capability (Class G) and those that provide for secondary-side depressurization and the use of the condensate system as an alternate core cooling method, and for which no feed and bleed is available (Class H).[***]

The remaining two classes address Westinghouse plants – Class A is associated with plants that require the use of spray systems for core heat removal following a LOCA, and Class B is associated with plants that can utilize low-to-high pressure recirculation for core heat removal.

Plants in which initiator response cannot be described using plant-class models are addressed using unique models, for example, the now deactivated LaCrosse BWR.

Table A.17 lists the class associated with each plant.

A.3 Event Tree Models

The plant class event trees describe core damage sequences for three initiating events: a nonspecific reactor trip, a LOOP, and a small-break LOCA. The event trees constructed are system-based and include an event tree applicable to each plant class defined.

System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar. Plants where certain mitigating systems do not exist, but which are largely analogous in their transient response, were grouped into the plant classes accordingly. In modeling events at such plants, the event tree branch probabilities were modified to reflect the systems available at the plant. Certain events (such as a postulated steam line break) could not be described using the plant-class event trees presented in this appendix. In these cases, unique event trees were developed to describe the sequences of interest.

[***]Maine Yankee Atomic Power Plant was built by Combustion Engineering but has a response to initiating events more akin to the Westinghouse Electric Corporation design, so it is grouped in a class with other Westinghouse plants. Davis-Besse Nuclear Power Station was also placed in a Westinghouse plant class because its HPI system design requires the operator to open the PORV for feed and bleed, as in most Westinghouse plants. The requirement to open the PORV for feed and bleed is a primary difference between event trees for Westinghouse and Babcock and Wilcox plants. Plant response differences resulting from the use of different SG designs are not addressed in the models.

This section (1) describes the potential plant response to the three initiating events described above, (2) identifies the combinations of systems required for the successful mitigation of each initiator, and (3) briefly describes the criteria for success of each system-based function. The sequences are considered first for PWRs and then separately for BWRs. PWR Class B event trees are described first, along with those for Class D, which are similar. (The major difference between Class B and Class D plants is that PORV operability is not required for feed and bleed on Class D plants.) The event trees for the combined group apply to the greatest number of operating PWRs. Therefore, these are discussed first, followed by those for PWR Classes G, H, and then A. For the BWR event trees, the plant Class C models are described first, because these are applicable to the majority of the BWRs, followed by discussions for the A and B BWR classes, respectively. The event trees are constructed with branch (event or system) success as the upper branch and failure as the lower branch. Each sequence path is read from left to right, beginning with the initiator followed by subsequent systems required to preclude or mitigate core damage.

The event trees can be found following the discussion sections and are grouped according to plant classes, beginning with the PWR classes and followed by the BWR classes. The abbreviations used in the event tree models are defined in Table A.16 preceding the event trees. Sequence numbers are provided on the event trees for undesirable end states (core damage and ATWS). Because of the similarities among PWR sequences for different plant classes, common sequence numbers have been assigned when possible. PWR Class B sequences were used as a basis for this. Sequence numbers beyond those for Class B are used for uncommon sequences on other plant classes. This approach facilitates comparison of sequences among plant classes. This approach could not be used for BWRs because of the significant difference in systems used on plants in the three plant classes. For BWRs, sequences are numbered in increasing order moving down each event tree. The following sequence number groups are employed for all event trees: transient with reactor trip success, 11-39; LOOP with reactor trip success, 40-69; small-break LOCA with reactor trip success, 71-79; ATWS sequences, 91-99.

The trees are presented in the following order:

A.23 BWR Class C loss of offsite power
A.24 BWR Class C small-break loss-of-coolant accident

A.3.1 PWR Event Sequence Models

The PWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following three initiating events: reactor trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to an initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor trip
Reactor coolant system integrity:	Addressed in small-break LOCA models plus trip and LOOP sequences involving failure of primary relief valves to close
Reactor coolant inventory:	High-pressure injection (assumed required only following a LOCA)
Short-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure injection and PORV (feed and bleed, PWR Classes A, B, D, and G)
	Secondary-side depressurization and use of condensate system (PWR Class H)
Long-term core heat removal:	Auxiliary feedwater
	Main feedwater
	High-pressure recirculation (PWR Classes B and D) (also required to support RCS inventory for all classes)
	Secondary-side depressurization and use of condensate system (PWR Class H)
	Containment spray recirculation (PWR Classes A and G)

PWR Nonspecific Reactor Trip

The PWR nonspecific reactor trip event tree constructed for plant Classes B and D is shown in Fig. A.7. The event-tree branches and the sequences leading to severe core damage and ATWS follow.

1. Initiating event (transient). The initiating event for the tree is a transient or upset event that requires or is followed by a rapid shutdown of the plant. LOOP and small-break LOCA initiators are modeled in separate event trees. Large-break LOCA or large SLB initiators are not addressed in the models described here.

2. Reactor trip. To achieve reactor subcriticality and thus halt the fission process, the reactor protection system (RPS) is required to insert control rods into the core. If the automatically initiated RPS fails, a reactor trip may be initiated manually. Failure to trip was considered to lead to the end state ATWS and was not developed further.

3. Auxiliary feedwater. AFW must be provided following trip to remove the decay heat still being generated in the reactor core via the SGs. Successful AFW operation requires flow from one or more AFW pumps to one or more SGs over a period of time ranging from 12 to 24 h (typically, one pump to one SG is adequate).

4. Main feedwater. In lieu of AFW, MFW can be utilized to remove the post shutdown decay heat. Depending on the individual plant design, either main or AFW may be used as the primary source of secondary-side heat removal.

5. PORV or SRV challenged. For sequences in which both reactor trip and steam generator feedwater flow (MFW or AFW) have been successful, the pressurizer PORV may or may not lift, depending on the peak pressurizer pressure following the transient. (In most transients, these valves do not lift.) The upper branch indicates that the valve or valves were challenged and opened. Because of the multiplicity of relief and safety valves, it was assumed that a sufficient number would open if the demand from a pressure transient exists.

   The lower branch indicates that the pressurizer pressure was not sufficiently high to cause opening of a relief valve. For the sequence in which both AFW and MFW fail following a reactor trip, at least one PORV or SRV was assumed to open for overpressure protection.

6. PORV or SRV reseats. Success for this branch requires the closure of any open relief valve once pressurizer pressure has decreased below the relief valve set point. If a PORV sticks open, most plants are equipped with an isolation valve that allows for manual termination of the blowdown. Failure of a primary-side relief valve to close results in a transient-induced LOCA that is modeled as part of this event tree.

7. High-pressure injection. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered. Success for this branch requires introduction of sufficient borated water to keep the core covered, considering core decay heat. (Typically, one HPI train is sufficient for this purpose.)

8. HPI and PORV open. If normal methods of achieving decay heat removal via the SGs (MFW and AFW) are unavailable, core cooling can be accomplished on most plants by establishing a feed and bleed operation. This operation (1) allows heat removal via discharge of reactor coolant to the containment through the PORVs and (2) RCS makeup via injection of borated water from the HPI system. Except at Class D plants, successful feed and bleed requires the operator to open the PORV manually. At Class D plants, the HPI discharge pressure is high enough to lift the primary-side safety valves, and feed and bleed can be accomplished without the operator manually opening the PORVs. HPI success is dependent on plant design but requires the introduction of sufficient

amounts of borated water into the RCS to remove decay heat and provide sufficient reactor coolant makeup to prevent core damage.

9. High-pressure recirculation. Following a transient-induced LOCA (a PORV or SRV fails to reseat), or failure of secondary-side cooling (AFW and MFW) and initiation of feed and bleed, continued core cooling and makeup are required. This requirement can be satisfied by using HPI in the recirculation mode. In this mode the HPI pumps recirculate reactor coolant collected in the containment sump and pass it through heat exchangers for heat removal. When MFW or AFW is available, heat removal is only required for HPI pump cooling; if AFW or MFW is not available, HPR is required to remove decay heat as well. Typically, at Class B and D plants, the LPI pumps are utilized in the HPR mode, taking suction from the containment sump, passing the pumped water through heat exchangers, and providing net positive suction head to the HPI pumps.

The event tree applicable to a PWR Class G nonspecific reactor trip is shown in Fig. A.10. Many of the event tree branches and the sequences leading to successful transient mitigation and core damage are similar to those following a nonspecific reactor trip transient for plant Class B. At Class G plants, however, the HPR system performs both the high- and low-pressure recirculation (LPR), function, taking suction directly from the containment sump without the aid of the low-pressure pumps. DHR is accomplished during recirculation by the containment spray recirculation (CSR) system. The event-tree branches and sequences are discussed further.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for PWR Classes B and D. The following branches have functions and success requirements similar to those following a transient at PWR Class B.

2. Reactor trip.

3. Auxiliary feedwater or main feedwater.

4. PORV or SRV challenged reseats.

5. High-pressure injection.

6. HPI and PORV open (feed and bleed). Success requirements for feed and bleed are similar to those following the plant Class B transient. Feed and bleed with operator opening of the PORV is required in the event that both AFW and MFW are unavailable for secondary-side cooling. In addition, DHR was assumed required to prevent potential core damage. This is provided by the CSR system.

7. High-pressure recirculation. In the event of a transient-induced LOCA, continued HPI via sump recirculation is needed to provide makeup to the break to prevent potential core damage. In addition, HPR is required when both AFW and MFW are unavailable following a transient, to recirculate coolant during the feed and bleed procedure. If HPR fails and normal secondary-side cooling is also failed, core damage will occur. In Class G plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

8. Containment spray recirculation. When feed and bleed (HPI, HPR, and PORV open) is required, the CSR system operates to remove decay heat from the reactor coolant being recirculated. Without the CSR system, the feed and bleed operation could not remove decay heat. Successful operation of feed and bleed and CSR was assumed to result in successful mitigation of core damage.

The event tree for PWR Class H non-specific reactor trip is shown in Fig. A.13. This class of plants is different than other PWR classes in that PORVs are not included in the plant design and feed and bleed cannot be used to remove decay heat in the event of main and AFW unavailability. If main or AFW cannot be recovered, the atmospheric dump valves can be used to depressurize the SGs to below the shutoff head of the condensate pumps, and these can be used, if available, for RCS cooling. Because of the need for secondary-side cooling for all success sequences, a requirement for CC to prevent core damage has not been modeled.

1. Initiating event (transient). The initiating event is a non-specific reactor trip, similar to that described for the previous PWR classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with previously described PWR classes.

2. Reactor trip.

3. Auxiliary feedwater.

4. Main feedwater.

5. SRV challenged. The upper branch indicates that at least one safety valve has lifted as a result of the transient. In most transients in which reactor trip has been successful and main or AFW is available, these valves do not lift. In the case where both main and AFW are unavailable, at least one SRV is assumed to lift. The lower branch indicates that the pressurizer pressure was not sufficiently high to cause the opening of a relief valve.

6. SRV reseat. Success for this branch requires the closure of any open safety valve once pressurizer pressure has been reduced below the safety valve set point.

7. High-pressure injection. In the case of a transient-induced LOCA, HPI is required to provide RCS makeup to keep the core covered.

8. High-pressure recirculation. The requirement for continued core cooling during mitigation of a transient-induced LOCA and following depletion of the refueling water tank can be satisfied by using HPI in the recirculation mode. In Class H plants, initiation of HPR realigns the HPI pumps to the containment sump. The use of LPI pumps for suction-pressure boosting is not required.

9. Steam generator depressurization. In the event that main and AFW are unavailable, the atmospheric dump valves (or turbine bypass valves if the main steam isolation valves are open) may be used on Class H plants to depressurize the SGs to the point that the condensate pumps can be used for SG cooling. In the event of main and AFW unavailability, failure to depressurize one SG to the operating pressure of the condensate system is assumed to result in core damage.

10. Condensate pumps. As described above, use of the condensate pumps on Class H plants along with secondary-side depressurization can provide adequate core cooling. Flow from one condensate

pump to one SG is assumed adequate. Unavailability of the condensate pumps in the event of failure to recover main and AFW is assumed to result in core damage.

The event tree applicable to PWR plant Class A nonspecific reactor trip is shown in Fig. A.4. Many of the event-tree branches and the sequences leading to successful transient mitigation and severe core damage are similar to those following a nonspecific reactor trip transient for plant Classes B and G.

Like the Class G plants, the Class A plants have a CSR system that provides DHR during HPR. Use of CSR for DHR was assumed to be required if AFW and MFW were unavailable. LPI pumps are required to provide suction to the HPI pumps during recirculation. The event-tree branches and sequences are discussed further below.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip, similar to that described for the other PWR plant classes. The following branches have functions and success requirements similar to those following a transient at PWRs associated with plant Classes B, D, and G.

2. Reactor trip.

3. Auxiliary feedwater

4. Main feedwater.

5. PORV or SRV challenged.

6. PORV/SRV reseats.

7. High-pressure injection.

8. High-pressure recirculation. In the event of a transient-induced LOCA, HPR can provide sufficient makeup to the break to terminate the transient. The LPI pumps provide suction to the high-pressure pumps in the recirculation mode. In the event that feed and bleed is required (following a transient in which both AFW and MFW are unavailable), HPR success is required.

9. Containment spray recirculation. The CSR system provides DHR during HPR when AFW and MFW are not available. In transient-induced LOCA sequences, HPI and HPR success is required to mitigate the event. In the event that secondary-side cooling via AFW or MFW is unavailable, feed and bleed with CSR, for DHR is considered sufficient to prevent core damage.

10. PORV open. The PORV must be opened by the operator below its set point to establish feed and bleed operation in the event that secondary-side cooling via AFW or MFW is unavailable.

Sequences resulting in core damage or ATWS following a PWR transient, shown on event trees applicable to each plant class, are described in Table A.4.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the use of SG depressurization and condensate pumps for RCS cooling in lieu of feed and bleed on Class H. Because of this similarity, consistent sequence numbers have been used for like sequences in different PWR plant classes. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.5.

PWR Loss of Offsite Power

The event trees constructed define representative plant responses to a LOOP. A LOOP (without turbine runback on plants with this feature) will result in reactor trip due to unavailability of power to the control rod drive (CRD) mechanisms and a loss of MFW because of the unavailability of power to components in the condensate and condenser cooling systems.

The PWR LOOP tree constructed for plant Classes B and D is shown in Fig. A.8. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (LOOP). The initiating event for the tree is a grid or switchyard disturbance to the extent that the generator must be separated from the grid and all offsite power sources are unavailable to plant equipment. The capability of a runback of the unit generator from full power to supply house loads exists at some plants but is not considered in the event tree. Only LOOPs that challenge the emergency power system (EPS) are addressed in the ASP Program.

2. Reactor trip given LOOP. Unavailability of power to the CRD mechanisms is expected to result in a reactor trip and rapid shutdown of the plant. If the reactor trip does not occur, the transient was considered to proceed to ATWS and was not developed further.

3. Emergency power. Given a LOOP and a reactor trip, electric power would be lost to all loads not backed by battery power. When power is lost, DGs are automatically started to provide power to the plant safety-related loads. Emergency power success requires the starting and loading of a sufficient number of DGs to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

4. Auxiliary feedwater. The AFW system functions to remove decay heat via the SG secondary side. Success requirements for this branch are equivalent to those following a nonspecific reactor trip and unavailability of MFW. Both MFW and condensate pumps would be unavailable following a LOOP. Therefore, with emergency power and AFW failed, no core cooling would be available, and core damage would be expected to occur. Because, specific AFW systems may contain different combinations of turbine-driven and motor-driven AFW pumps, the capability of the system to meet its success requirements will depend on the state of the EPS and the number of turbine-driven AFW pumps that are available.

5. PORV or SRV challenged. The upper and lower states for this branch are similar to those following a nonspecific reactor trip. The PORV or SRV may or may not lift, depending on the peak pressure following the transient.

6. PORV or SRV reseats. The success requirements for this branch are similar to those following a nonspecific reactor trip. However, for the sequence in which emergency power is failed and the PORV fails to reseat, the HPI/HPR system would be without power to mitigate potential core damage.

7. Seal LOCA. In the event of a loss of emergency power following LOOP, both SW and component cooling water (CCW) are faulted. This results in unavailability of RCP seal cooling and seal injection (since the charging pumps are also without power and cooling water). Unavailability of seal cooling and injection may result in seal failure after a period of time, depending on the seal

design (for some seal designs, seal failure can be prevented by isolating the seal return isolation valve).

The upper event tree branch represents the situation in which seal failure occurs prior to restoration of ac power. The lower branch represents the situation in which a seal LOCA does not occur.

8. Electric power recovered (long term). For sequences in which a seal LOCA has occurred, success requirements are the restoration of ac power [either through recovery of offsite power or recovery of a DG] prior to core uncovery. For sequences in which a seal LOCA does not occur, success requires the recovery of ac power prior to battery depletion, typically 2 to 4 h.

9. High-pressure injection and recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. Because all HPI/HPR systems use motor-driven pumps, the capability of the HPI or HPR system to meet its success requirements depends on the success of the EPS.

10. PORV open (for feed and bleed). The success requirements for this branch are similar to those following a nonspecific reactor trip. The PORV is opened in conjunction with feed and bleed operations when secondary-side heat removal is unavailable. For Class D plants, the PORV does not have to be manually opened to establish feed and bleed because the HPI pump discharge pressure is high enough to lift the PORV or primary relief valve.

The event tree constructed for the PWR Class G LOOP is shown in Fig. A.11. Most of the event-tree branches and the sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, at Class G plants, DHR during recirculation is provided by the CSR system, not the HPR system. The event-tree branches and sequences are discussed further below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for PWR plant Classes B and D. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

5. PORV or SRV challenged.

6. PORV/SRV valve reseats.

7. Seal LOCA.

8. Electric power recovered (long term).

9. High-pressure injection and recirculation.

10. PORV open (for feed and bleed).

11. Containment spray recirculation. The success requirements for this branch are similar to those following a nonspecific reactor trip. The CSR system provides DHR for sequences in which secondary-side cooling is unavailable.

The event tree constructed for a PWR Class H LOOP is shown in Fig. A.14. Many of the event tree branches and sequences leading to successful mitigation and core damage are similar to those following a LOOP at Class B plants. However, Class H plants do not have feed and bleed capability and rely instead on secondary-side depressurization and the condensate system as an alternate DHR method. The condensate system is assumed unavailable following a LOOP, which limits the diversity of DHR methods on this plant class following this initiator. The event branches and sequences are discussed further below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for BWR Classes B and D. The following branches have functions and success requirements similar to those following a LOOP at PWRs associated with all of the plant classes defined.

2. Reactor trip given LOOP.

3. Emergency power.

4. Auxiliary feedwater.

5. SRV challenged. The function of this branch is similar to that described under the PWR Class H transient.

6. SRV reseat. Success requirements for this branch are similar to those described under the PWR Class H transient.

7. Seal LOCA.

8. Electric power recovered (long-term).

9. High pressure injection and recirculation.

The event tree constructed for the plant Class A LOOP is shown in Fig. A.5. All of the event-tree branches and the sequences leading to successful transient mitigation, potential core vulnerability, and severe core damage are analogous to those following a LOOP at Class B plants with the addition of the CSR branch, which is required for successful feed and bleed. At Class A plants, DHR during HPR is accomplished by the CSR system; whereas at Class B and D plants, DHR is an integral part of the HPR system. Additional information on the use of the CSR system is provided in the discussion of the PWR Class A nonspecific reactor trip event tree.

Sequences resulting in core damage and ATWS following a PWR LOOP, shown on event trees applicable to each plant class, are described in Table A.6.

Many of the sequences are the same for different plant classes, the primary differences being the use of CSR on Class G and Class A, and the unavailability of feed and bleed on Class H. As with the PWR transient sequences, this similarity permits consistent numbering of a large number of sequences. All sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.7.

PWR Small-Break Loss-of-Coolant Accident

Event trees were constructed to define the responses of PWRs to a small-break LOCA. The LOCA chosen for consideration is one that would require a reactor trip and continued HPI for core protection. Because of the limited amount of borated water available, the mitigation sequence also includes the requirement to recirculate borated water from the containment sump.

The LOCA event tree constructed for PWR plant Classes B and D is shown in Fig. A.9. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (small-break LOCA). The initiating event for the tree is a small-break LOCA that requires reactor trip and continued HPI for core protection.

2. Reactor trip. Reactor trip success is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition. Failure to trip was considered to lead to the end state ATWS.

3. Auxiliary feedwater or main feedwater. Use of AFW or MFW was assumed necessary for some small breaks to reduce RCS pressure to the point where HPI is effective. At Class D plants, the HPI pumps operate at a much higher discharge pressure and hence can function without secondary-side cooling from the AFW or MFW systems.

4. High-pressure injection. Adequate injection of borated water from the HPI system is required to prevent excessive core temperatures and consequent core damage.

5. High-pressure recirculation. Following a small-break LOCA, continued high pressure injection is required. This is typically accomplished with the residual heat removal (RHR) system, which takes suction from the containment sump and returns the lost reactor coolant to the core via the HPI pumps. The RHR system includes heat exchangers that remove decay heat prior to recirculating the sump water to the RCS.

6. PORV open. In the event AFW and MFW are unavailable following a small break LOCA, opening the PORV can result in core cooling using the feed and bleed mode. Depending on the size of the small break, opening the PORV may not be required for success. PORV open is not required for success for Class D.

The event tree constructed for a small-break LOCA at Class G plants is shown in Fig. A.12. The LOCA event tree for Class G plants is similar to that for Class B and D plants except that long-term cooling is provided by the CSR system rather than by the HPR system. The event-tree branches and sequences are discussed further below.

1. Initiating event (small-break LOCA). The initiating event is a LOCA similar to that described for PWR plant Classes B and D. The following branches have functions and success requirements similar to those following a small-break LOCA at PWRs associated with all of the plant classes defined.

2. Reactor trip.

3. Auxiliary feedwater and main feedwater

4. High-pressure injection.

5. High-pressure recirculation.

6. PORV open.

7. Containment spray recirculation. In the event that normal secondary-side cooling (AFW or MFW) is unavailable following a small LOCA, cooling via the CSR system during HPR is required to mitigate the transient.

The event tree constructed for a small-break LOCA at PWR Class H plants is shown in Fig. A.15. The event tree has been developed assuming that SG depressurization and condensate pumps can provide adequate RCS pressure reduction in the event of an unavailability of AFW and MFW to permit HPI and HPR to function in these plants. The event tree branches and sequences are discussed further below.

1. Initiating event (small-break LOCA). The initiating event is similar to that described above for PWR Classes B, D, and G. The following branches have functions and success requirements similar to those discussed previously.

2. Reactor trip.

3. Auxiliary and main feedwater.

4. High-pressure injection.

5. High-pressure recirculation.

6. SG depressurization. In the event that AFW and MFW are unavailable following a small-break LOCA, SG depressurization combined with the use of the condensate pumps can provide for RCS depressurization such that adequate HPI and HPR can be achieved. Success requirements are the same as those following a transient with unavailability of AFW and MFW.

7. Condensate pumps. Use of one condensate pump provided flow to at least one SG as required in conjunction with SG depressurization to provide for RCS depressurization and cooling.

The event tree constructed for a small LOCA at Class A plants is shown in Fig. A.6. The LOCA event tree for Class A plants is similar to that for Classes B and D except that the CSR system is required in conjunction with HPR in some sequences where secondary cooling is not provided. The sequences that follow combined AFW and MFW failure with HPR and CSR success are identical to those that follow HPR success at Class B and D plants; and sequences that follow HPR or CSR failure at Class A plants are identical to those that follow HPR failure.

Sequences resulting in core damage or ATWS following a PWR small-break LOCA, shown on event trees applicable to each plant class, are described in Table A.8.

As with the PWR transient and LOOP sequences, differences between plant classes are driven by the use of CSR on plant classes A and G, and by the use of secondary-side depressurization and condensate pumps in lieu of feed and bleed on PWR Class H. All small-break LOCA sequences, required branch success and failure states, and the applicability of each sequence to each plant class are summarized in Table A.9.

Alternate Recovery Actions

The PWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection from core damage are unavailable. In the event AFW and MFW are unavailable and cannot be recovered in the short term, the use of feed and bleed cooling is modeled on all plants except for Class H, where SG depressurization and use of the condensate pumps is modeled instead. In addition, the potential for short-term recovery of a faulted system is also included in appropriate branch models (AFW, MFW, and HPI, for example).

Alternate equipment and procedures, beyond the systems and functions included in the event trees, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental DGs, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.

- Depressurization following a small-break LOCA to the initiation pressure of the LPI systems to provide RCS makeup in the event that HPI fails. Procedures to support this action are known to exist on some plants.

- Depressurization following a small-break LOCA to the initiation pressure of the DHR system, and then proceeding to cold shutdown. While plant procedures specify the use of sump recirculation following a small LOCA or feed and bleed, sufficient RWST inventory exists to delay this action until many hours into the event, during which recovery of faulted systems may be affected. It is likely that operators will delay sump recirculation as long as possible while trying to place the plant in a stable condition through recovery of secondary-side cooling and the use of RHR.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.3.2 BWR Event Sequence Models

The BWR event trees describe the impact of the availability and unavailability of front-line systems in each plant class on core protection following the same three initiating events addressed for PWRs: trip, LOOP, and small-break LOCA. The systems modeled in the event trees are those associated with the generic functions required in response to any initiating event, as described in Sect. A.2. The systems that are assumed capable of providing these functions are:

Function	System
Reactor subcriticality:	Reactor scram
Reactor coolant system integrity:	Addressed in small-break LOCA models and in trip and LOOP sequences involving failure of primary relief valves to reseat
Reactor coolant inventory:	High-pressure injection systems [HPCI or HPCS, RCIC (non-LOCA situations), CRD (non-LOCA situations), FWCI]
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS, RHRSW or equivalent]
Short-term core heat removal:	Power conversion system
	High-pressure injection systems [HPCI, RCIC, CRD, FWCI (BWR Class A)]
	Isolation condenser (BWR Classes A and B)
	Main feedwater
	Low-pressure injection systems following blowdown [LPCI (BWR Classes B and C), LPCS]
	Note: Short-term core heat removal to the suppression pool (all cases where power conversion system is faulted) requires use of the RHR system for containment heat removal in the long term.
Long-term core heat removal:	Power conversion system
	Isolation condenser (BWR Class A)
	Residual heat removal [shutdown cooling or suppression pool cooling modes (BWR Class C)]
	Shutdown cooling (BWR Classes A and B)
	Containment cooling (BWR Class A)
	Low-pressure coolant injection [CC mode (BWR Class B)]

BWR Nonspecific Reactor Trip

The nonspecific reactor trip event tree constructed for BWR plant Class C is shown in Fig. A.22. The event tree branches and the sequences leading to potential severe core damage follow. The Class C plants are discussed first because all but a few of the BWRs fit into the Class C category.

1. Initiating event (transient). The initiating event is a transient or upset event that results in a rapid shutdown of the plant. Transients that are initiated by a LOOP or a small-break LOCA are modeled

in separate event trees. Transients initiated by a large-break LOCA or large SLB are not addressed in the event trees described here; trees applicable to such initiators are developed separately if required.

2. Reactor shutdown. To achieve reactor subcriticality and thus halt the fission process, the RPS commands rapid insertion of the control rods into the core. Successful scram requires rapid insertion of control rods with no more than two adjacent control rods failing to insert.

3. Power conversion system (PCS). Upon successful reactor scram, continued operation of the PCS would allow continued heat removal via the main condenser. This is considered successful mitigation of the transient. Continued operation of the PCS requires the MSIVs to remain open and the operation of the condenser, the turbine bypass system (TBS), the condensate pumps, the condensate booster pumps, and the feedwater pumps.

4. SRV challenged. Depending on the transient, one or more SRVs may open. The upper branch on the event tree indicates that the valves were challenged and opened. If the transient is followed by continued PCS operation and successful scram, the SRVs are not expected to be challenged. If the PCS is unavailable, at least some of the SRVs are assumed to be challenged and to open.

5. SRV close. Success for this branch requires the reseating of any open relief valves once the reactor pressure vessel (RPV) pressure decreases below the relief valve set point. If an SRV sticks open, a transient-induced LOCA is initiated.

6. Feedwater. Given unavailability of the PCS, continued delivery of feedwater to the RPV will keep the core from becoming uncovered. This, in combination with successful long-term DHR, will mitigate the transient, preventing core damage. For plants with turbine-driven feed pumps, the PCS failure with subsequent feedwater success cannot involve MSIV closure, or loss of condenser vacuum, because this would disable the feed pumps.

7. HPCI or HPCS. The primary function of the HPCI or HPCS system is to provide makeup following small-break LOCAs while the reactor is at high-pressure (not depressurized). The system is also used for DHR following transients involving a loss of feedwater. Some later Class C plants are equipped with HPCS systems, but the majority are equipped with HPCI systems. HPCI or HPCS can provide the required makeup and short-term DHR when DHR is unavailable from the condenser and the feedwater system cannot provide makeup.

8. RCIC. The RCIC system is designed to provide high-pressure coolant makeup for transients that result in LOFW. Both RCIC and HPCI (or HPCS) initiate when the reactor coolant inventory drops to the low-low level set point, taking suction from the condensate storage tank or the suppression pool. HPCI is normally secured after HPCI/RCIC initiation when pressure and water level are restored, to prevent tripping of HPCI and RCIC pumps on high water level. RCIC must then be operated until the RHR system can be placed in service. Following a transient, scram, and unavailability of the PCS, reactor pressure may increase, causing the relief valves to open and close periodically to maintain reactor pressure control.

9. CRD pumps. In transient-induced sequences where heat removal and minimal core makeup are required (i.e., not transient-induced LOCA sequences), the CRD pumps can deliver high-pressure coolant to the RPV.

10. Depressurization via SRV or the automatic depressurization system (ADS). In the event that short-term DHR and core makeup are required and high-pressure systems have failed to provide adequate flow, the RPV can be depressurized to allow use of the low-pressure, high-capacity injection systems. If depressurization fails in this event, core damage is expected to occur. The ADS will automatically initiate on high drywell pressure and low-low reactor water level, and the availability of one train of the LPCI or LPCS systems, following a time delay. The SRVs can be opened by the operators to speed the depressurization process or to initiate it if ADS fails and if additional, operable valves are available.

11. LPCS. LPI can be provided by the LPCS system if required. The LPCS system performs the same functions as the LPCI system (described below) except that the coolant, which is drawn from the SP or the condensate storage tank (CST), is sprayed over the core.

12. LPCI. The LPCI system can provide short-term heat removal and cooling water makeup if the reactor has been depressurized to the operating range of the low-head RHR pumps. At Class C plants, LPCI is a mode of the RHR system; thus, the RHR pumps operate during LPCI. LPCI takes suction from the suppression pool (SP) or the CST and discharges into the recirculation loops or directly into the reactor vessel. If LPCI is successful in delivering sufficient flow to the reactor, long-term heat removal success is still required to mitigate core damage.

13. Residual heat removal shutdown cooling (SDC) mode. In this mode, the RHR system provides normal long-term DHR. Coolant is circulated from the reactor by the RHR pumps through the RHR heat exchangers and back to the reactor vessel. Long-term core cooling success requires that heat transfer to the environment commence within 24 h of the transient. RHR SDC success following successful reactor scram and high- or low-pressure injection of water to the RPV will prevent core damage.

14. RHR SP cooling mode. If RHR SDC is unavailable, the RHR pumps and heat exchangers can be aligned to take water from the SP, cool it via the RHR heat exchangers, and return it to the SP. This alignment can provide long-term cooling for transient mitigation.

15. RHR service water or other. This is a backup measure for providing water to the reactor to reflood the core and maintain core cooling if LPCI and LPCS are unavailable. Typically, the high-pressure SW pumps are aligned to the shell side of the RHR heat exchangers for delivery of water to one of the recirculation loops.

The event tree constructed for a BWR plant Class A nonspecific reactor trip is shown in Fig. A.16. The event tree is similar to that constructed for BWR Class C plants with the following exceptions: Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. The isolation condensers can provide long-term core cooling. Class A plants do not have LPCI systems, although they are equipped with LPCS; SP cooling is provided by a system independent of the SDC system. The event tree branches and sequences are discussed further below.

1. Initiating event (transient). The initiating event is a nonspecific reactor trip similar to that described for BWR Class C plants. The following branches have functions and success requirements similar to those following a transient at BWRs associated with Class C.

2. Reactor shutdown.

3. Power conversion system.

4. SRV challenged and closed.

5. Isolation condensers and isolation condenser makeup. If PCS is not available and significant inventory has not been lost via the SRVs, then the IC system can provide for DHR and mitigate the transient. The IC system is an essentially passive system that condenses steam produced by the core, rejecting the heat to cooling water and returning the condensate to the reactor. Makeup is provided to the cooling water as needed. The system does not provide makeup to the reactor vessel.

6. FW or FWCI. Either FW or FWCI can provide short-term transient mitigation. When feedwater or FWCI is required and is successful, long-term DHR is required for complete transient mitigation. (PCS unavailability is assumed prior to feedwater or FWCI demand.) FWCI or feedwater is required for makeup in transient-induced LOCA sequences and for heat removal in sequences when the IC system would have mitigated the transient but was not available. FWCI is initiated automatically on low reactor level and uses the normal feedwater trains to deliver water to the reactor vessel.

7. CRD pumps.

8. Depressurization via SRV or ADS.

9. LPCS.

10. Fire water or other. Fire water or other raw water systems can provide a capability similar to that provided by the SW/RHR connection on Class C BWRs. As a backup source, if all normal core cooling is unavailable, fire water can be aligned to the LPCS injection line to provide water to the reactor vessel.

11. SDC. Like the RHR system at Class C plants, the SDC system is a closed-loop system that performs the long-term DHR function by circulating primary coolant from the reactor through the system's heat exchangers and back to the reactor vessel. Success requires the operation of at least one SDC loop. Long-term DHR is required to terminate transients in which high- or low-pressure injection is required to mitigate the transient.

12. Containment cooling. If the SDC system fails to provide long-term DHR, the CC system can remove decay heat. The system utilizes dedicated CC pumps, drawing suction from the SP, passing it through heat exchangers where heat is rejected to the SW system and then either returning it directly to the SP or spraying it into the dry well.

The event tree constructed for a BWR plant Class B nonspecific reactor trip is shown in Fig. A.19. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same except that Class B plants are equipped with HPCI systems instead of FWCI systems, and they are equipped with a LPCI system that represents an additional capability for providing LPCI. Also, at Class B BWRs, the CC system considered in the event tree utilizes the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR transient, shown on event trees applicable to each plant class, are described in Table A.10. Because of differences in the mitigation systems used in the three BWR classes, it is not possible to associate most sequences among different plant classes. Because of this, similar sequence numbers used for sequences in different plant classes do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss of Offsite Power

The event cores constructed define responses of BWRs to a LOOP in terms of sequences representing success and failure of plant systems. A LOOP condition will result in a generator load rejection that would trip the turbine control valves and initiate a reactor scram.

The event tree constructed for a LOOP at BWR Class C plants is shown in Fig. A.23. The event-tree branches and the sequences leading to core damage follow.

1. Initiating event (LOOP). The initiating event for a LOOP corresponds to any situation in which power from both the auxiliary and startup transformers is lost. This situation could result from grid disturbances or onsite faults.

2. Emergency power. Emergency power is provided by DGs at almost all plants. The DGs receive an initiation signal when an undervoltage condition is detected. Emergency power success requires the starting and loading of a sufficient number of DGs to support safety-related loads in systems required to mitigate the transient and maintain the plant in a safe shutdown condition.

3. Reactor shutdown. Given a load rejection, a scram signal is generated. Successful scram is the same as for the transient trees: a rapid insertion of control rods with no more than two adjacent control rods failing to insert. The scram can be automatically or manually initiated.

4. LOOP recovery (long-term). Success for this branch requires recovery of offsite power or diesel-backed ac power before the station batteries are depleted, typically 2 to 4 h.

5. SRV challenged and closed. If one or more SRV is challenged and fails to close, a transient-induced LOCA is initiated.

6. HPCI (or HPCS) or RCIC. Success requirements for these branches are identical to those following a transient at Class C BWRs. Either RCIC or HPCI (or HPCS) can provide the makeup and short-term core cooling required following most transients, including failure of the EPS. HPCI and RCIC only require dc power and sufficient steam to operate the pump turbines. HPCS systems utilize a motor-driven pump but are diesel-backed and utilize dedicated SW cooling.

7. CRD pumps. Given emergency power success, CRD pump success requirements following a LOOP are identical to those following a transient. The CRD pumps can provide sufficient makeup to remove decay heat but not enough makeup to mitigate a transient-induced LOCA. Manual restart of the CRD pumps is required following the LOOP.

8. Depressurization via SRV or the ADS.

9. LPCS, LPCI, or RHR service water.

10. RHR SDC mode or RHR SP cooling mode. For emergency power success sequences, the success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class C BWRs. Success for any one of these three branches can provide the long-term DHR required for transient mitigation. If emergency power fails, it must be recovered to power long-term DHR equipment. However, long-term DHR is not required until several hours (up to 24 h) into the transient.

The event tree constructed for a LOOP at BWR Class A plants is shown in Fig. A.17. The event tree is similar to that constructed for BWR Class C plants with the major exception that Class A plants are equipped with ICs and FWCI systems instead of RCIC and HPCI (or HPCS) systems. However, given a LOOP, FWCI would be unavailable, because it is not backed by emergency power. Also, additional long-term core cooling is not required with IC success, as long as no transient-induced LOCA is initiated. In the emergency power failure sequences, the IC system is the only system that can provide core cooling because FWCI would be without power. The event-tree branches and sequences are further discussed below.

1. Initiating event (LOOP). The initiating event is a LOOP similar to that described for Class C BWRs. The following branches have functions and success requirements similar to those following a LOOP at BWRs associated with previously described BWR classes.

2. Emergency power.

3. Reactor shutdown.

4. LOOP recovery (long-term).

5. SRV challenged and closed.

6. IC. Following successful reactor scram, the IC system can provide enough DHR, in both the short and long term, to mitigate the transient if a transient-induced LOCA has not been initiated. The IC system cannot provide coolant makeup, which would be required in a transient-induced LOCA. The IC system is an essentially passive system that does not require ac power for success.

7. FWCI. The FWCI system can provide short-term core cooling and makeup for transient mitigation. However, FWCI success requires normal power supplies and cannot be powered by emergency power following a LOOP.

8. CRD pumps.

9. Depressurization via SRV or ADS.

10. LPCS, fire water, or other water source. Success requirements for these branches are similar to those following a nonspecific reactor trip at Class A BWRs. With interim high-pressure cooling unavailable, either LPCS or, as a last resort, fire water or another water source can be used to provide low-pressure water for core makeup and cooling.

11. SDC and containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs.

The event tree constructed for a BWR plant Class B LOOP is shown in Fig. A.20. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that Class B plants are equipped with HPCI systems instead of FWCI systems and are equipped with a LPCI system, which represents an additional capability for providing LPCI. At Class B BWRs the CC system utilizes the LPCI pumps rather than having its own dedicated pumps. In emergency power failure sequences, either the IC or HPCI system can provide the required core cooling for short-term transient mitigation. However, if an SRV sticks open (transient-induced LOCA), the ICs cannot provide the makeup needed, and HPCI is required. The ICs can also provide long-term cooling, but when only HPCI is operable, recovery of emergency power is necessary to power SDC-related loads.

Sequences resulting in core damage following a BWR LOOP, as shown on each plant-class event tree, are described in Table A.11. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

BWR Loss-of-Coolant Accident

The event trees constructed define the response of BWRs to a small LOCA in terms of sequences representing success and failure of plant systems. The LOCA chosen for consideration is a small LOCA, one that would require a reactor scram and continued operation of HPI systems. A large LOCA would require operation of the high-volume/low-pressure systems and is not addressed in the models.

The LOCA event tree constructed for BWR Class C plants is shown in Fig. A.24. The event-tree branches and sequences leading to core damage and core vulnerability follow.

1. Initiating event (small LOCA). Any breach in the RCS on the reactor side of the MSIVs that results in coolant loss in excess of the capacity of the CRD pumps is considered a LOCA. A small LOCA is considered to be one in which losses are not great enough to reduce the system pressure to the operating range of the LPI systems.

2. Reactor shutdown. Successful scram is defined as the rapid insertion of sufficient control rods to place the core in a subcritical condition.

3. HPCI or HPCS. HPCI (or HPCS, depending on the plant) can provide the required inventory makeup.

4. Depressurization via SRV or ADS. The success requirements for this branch are similar to those following a nonspecific reactor trip transient. SRV/ADS success allows the use of low-pressure systems to provide short-term core cooling and makeup.

5. LPCS, LPCI, or RHR service water. The success requirements for these branches are similar to those following a nonspecific reactor trip transient. Any one of these branches can provide short-term core cooling and makeup if SRV/ADS is successful.

6. RHR (SDC mode) or RHR (SP cooling mode). Success requirements for these branches are similar to those following a nonspecific reactor trip transient, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the break size. These methods each have the capability of providing long-term DHR. Long-term DHR is required in all sequences for LOCA mitigation.

The LOCA event tree constructed for BWR Class A plants is shown in Fig. A.18. The event tree is similar to the LOCA tree constructed for BWR Class C plants except that Class A plants have FWCI instead of HPCI or HPCS systems and are, in general, not equipped with LPCI systems (only LPCS systems). In addition, SP and CC systems are independent of the SDC system. The event tree branches and sequences leading to core damage follow.

1. Initiating event (small LOCA). The initiating event is a small LOCA similar to that described for BWR Class C plants. The following branches have functions and success requirements similar to those following a small LOCA at BWRs associated with the previously described BWR classes.

2. Reactor shutdown.

3. FWCI. The FWCI system has the capability to keep the core covered and provide interim core cooling. FWCI initiates automatically on low reactor water level.

4. Depressurization via SRV or ADS.

5. LPCS or fire water (or other water source). The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs. Either of these systems (branches) can provide LPI for makeup and short-term core cooling if high-pressure systems are unavailable.

6. SDC or containment cooling. The success requirements for these branches are similar to those following a nonspecific reactor trip transient at Class A BWRs, except that heat rejection to the environment may be required sooner than 24 h into the transient, depending on the size of the break. Either of these methods can provide the long-term DHR required to mitigate a small LOCA.

The LOCA event tree constructed for BWR Class B plants is shown in Fig. A.21. The event tree is most similar to that constructed for BWR Class A plants. In fact, the branches and sequences are the same, except that some Class B plants are equipped with HPCI systems instead of FWCI systems and Class B BWRs have a LPCI system, which provides an additional capability for LPCI. At Class B BWRs the CC system uses the LPCI pumps rather than having its own dedicated pumps.

Sequences resulting in core damage following a BWR small-break LOCA, as shown on each plant-class event tree, are described in Table A.12. As in the case of BWR transients, similar sequence numbers do not imply similarity among the sequences. (Because of the lack of similarity among sequences for the three BWR classes, no sequence summary table has been provided.)

Alternate Recovery Actions

The BWR event trees have been developed on the basis that proceduralized recovery actions will be attempted if primary systems that provide protection against core damage are unavailable. If feedwater, HPCI, and RCIC are unavailable (FWCI and ICs on BWR Classes A and B) and cannot be recovered in the short term, the use of the CRD pumps (provided no LOCA exists) and the use of ADS (to depressurize below the operating pressure of low-pressure systems) are modeled. In addition, the potential for short-term recovery of a faulted system is also included in the appropriate branch model.

Alternate equipment and procedures, beyond the systems and functions included in the event tree, may be successful in mitigating the effects of an initiating event, provided the appropriate equipment or procedure is available at a particular plant. This may include:

- The use of supplemental diesel generators, beyond the normal safety-related units, to power equipment required for continued core cooling and reactor plant instrumentation. A number of plants have added such equipment, often for fire protection.
- The use of RCIC to provide RPV makeup for a single stuck-open relief valve. Thermal-hydraulic analyses performed to support a number of BWR probabilistic risk assessments have demonstrated the viability of RCIC for this purpose.
- The use of the condensate system for LPI. This recovery action requires that the condensate system be available (even though PCS and feedwater are unavailable) and that the plant has been depressurized.
- The use of containment venting for long-term DHR, provided an injection source is available. This core cooling method has been addressed in some PRAs.

The potential use of these alternate recovery actions was addressed in the analysis of the 1992 precursors when information concerning their plant specific applicability was available.

A.4 Branch Probability Estimates

Branch probability estimates used in the 1988-1992 precursor calculations were developed using information in the 1984-86 precursors when possible. Probability values developed from precursor information are shown in Table A.13. The process used to estimate branch probability values used in the precursor calculations is described in detail in Appendix C to Ref. 5 and in Ref. 6.

In addition to system failures caused by equipment failures, the likelihood of failing to actuate manually actuated systems was also included in the models. Examples of such systems are the DHR system in BWRs and feed and bleed in PWRs. For actions in the control room, revised failure to initiate probabilities consistent with those utilized for 1987 precursor calculations were also used for 1988-1992 calculations. These revised values typically assume a failure probability of 0.001 for an unburdened action and 0.01 for a burdened action. The failure probability for subsequent actions is assumed to be higher. Operator action failure probabilities used in the 1988-1992 calculations are shown in Table A.14.

A.5 Reference Event Calculations

Conditional core damage probability estimates were also calculated for nonspecific reactor trip, LOFW, and unavailabilities in certain single-train BWR systems (HPCI, HPCS, RCIC, and CRD cooling). These calculations indicate the relative importance of these events, which are too numerous to warrant individual calculation. The results of these calculations, performed without consideration of alternate recovery actions that were addressed in certain 1992 precursor assessments, are listed in Table A.15.

Table A.15 shows that nonspecific reactor trips without additional observed failures have conditional core damage probabilities below 5×10^{-6} per trip, depending on plant class. The likelihood of LOFW in conjunction with a trip is included in these calculations. LOFW conditional core damage probabilities are less than 4×10^{-5} per LOFW event, again depending on plant class, except for BWR Class A plants (1.7×10^{-4}). The conditional core damage probabilities associated with unavailabilities of HPCI and HPCS (single-train BWR systems) are also above 10^{-5}, assuming a one-half month unavailability.

A.6 References

1. J. W. Minarick and C. A. Kukielka, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., and Science Applications, Inc., *Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report*, USNRC Report NUREG/CR-2497 (ORNL/NOAC-232, Vol. 1 and 2),1982.*
2. W. B. Cottrell, J. W. Minarick, P. N. Austin, E. W. Hagen, and J. D. Haltis, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents: 1980-81, A Status Report*, USNRC Report NUREG/CR-359 1, Vols. 1 and 2 (ORNL/NSIC-217/V1 and V2), July 1984.
3. M. Modarres, E. Lois, and P. Amico, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., *LER Categorization Report*, University of Maryland, College Park, MD, Nov. 13, 1984.*
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5. J. W. Minarick et al., Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., and Science Applications International Corp., *Precursors to Potential Severe Core Damage Accidents; 1986, A Status Report*, USNRC Report NUREG/CR-4674 (ORNL/NOAC-232, Vols. 5 and 6), May 1988.
6. J. W. Minarick, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.; Science Applications International Corp., *Revised LOOP Recovery and PWR Seal LOCA Models*, Technical Letter Report ORNL/NRC/LTR-89/11, August 1989.*

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*Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

Table A.1 Branch probability estimation process

Branch failure	Observed operational event	Non-recovery likelihood for event	Effective number of non-recoverable events	Observation period	Probability estimate
Steam generator isolation	Steam line pressure transmitters (9 of 12) were found in faulty alignment, which would have prevented automatic steam line isolation on demand at Maine Yankee (LER 309/85-009, 8/7/85)	0.04	1.04	12 demands per reactor year due to testing in 164 PWR reactor years (1984–86 observation period) results in 1968 demands	5.3×10^{-4}
	All MSIVs failed to close prior to entering refueling at Point Beach 2 (LER 301/86-004, 9/28/86)	1.0			

Table A.2 Rules for calculating precursor significance

1. Event sequences requiring calculati:n.

   If an initiating event occurs as part of a precursor (i.e., the precursor consists of an initiating event plus possible additional failures), then use the event tree associated with that initiator; otherwise, use all event trees impacted by the observed unavailability.

2. Initiating event probability.

   If an initiating event occurs as part of a precursor, then the initiator probability used in the calculation is the probability of failing to recover from the observed initiating event (i.e., the numeric value of the recovery class for the event).

   If an initiating event does not occur as part of a precursor, then the probability used for the initiating event is developed using the initiating event frequency and event duration. Event durations (the period of time during which the failure existed) are based on information included in the event report, if provided. If the event is discovered during testing, then one-half of the test period (15 days for a typical 30-day test interval) is assumed, unless a specific failure duration is identified.

3. Branch probability estimation.

   For event tree branches for which no failed or degraded condition is observed, a probability equal to the estimated branch failure probability is assigned.

   For event tree branches associated with a failed system, a probability equal to the numeric value associated with the recovery class is assigned.

   For event tree branches that include a degraded system (i.e., a system that still meets minimum operability requirements but with reduced or no redundancy), the estimated failure probability is modified to reflect the loss of redundancy.

4. Support system unavailabilities.

   Systems or trains rendered unavailable as a result of support system failures are modeled recognizing that, as long as the affected support system remains failed, all impacted systems (or trains) are unavailable; but if the support system is recovered, all the affected systems are recovered. This can be modeled through multiple calculations that address support system failure and success. Calculated core damage probabilities for each case are normalized based on the likelihood of recovering the support system. (Support systems, except emergency power, are not directly modeled in the current ASP models.)

Table A.3 ASP reactor plant classes

Plant name	Plant class	Plant name	Plant class
ANO-Unit1	PWR Class D	Millstone 3	PWR Class A
ANO-Unit	PWR Class G	Monticello	BWR Class C
Beaver Valley 1	PWR Class A	Nine Mile Point 1	BWR Class A
Beaver Valley 2	PWR Class A	Nine Mile Point 2	BWR Class C
Big Rock Point	BWR Class A	North Anna 1	PWR Class A
Browns Ferry 1	BWR Class C	North Anna 2	PWR Class A
Browns Ferry 2	BWR Class C	Oconee 1	PWR Class D
Browns Ferry 3	BWR Class C	Oconee 2	PWR Class D
Braidwood 1	PWR Class B	Oconee 3	PWR Class D
Braidwood 2	PWR Blass B	Oyster Creek	BWR Class A
Brunswick 1	BWR Class C	Palisades	PWR Class G
Brunswick 2	BWR Class C	Palo Verde 1	PWR Class H
Byron 1	PWR Class B	Palo Verde 2	PWR Class H
Byron 2	PWR Class B	Palo Verde 3	PWR Class H
Callaway 1	PWR Class B	Peach Bottom 2	BWR Class C
Calvert Cliffs 1	PWR Class G	Peach Bottom 3	BWR Class C
Calvert Cliffs 2	PWR Class G	Perry 1	BWR Class C
Catawba 1	PWR Class B	Pilgrim 1	BWR Class C
Catawba 2	PWR Class B	Point Beach 1	PWR Class B
Clinton 1	BWR Class C	Point Beach 2	PWR Class B
Comanche Peak 1	PWR Class B	Prairie Island 1	PWR Class B
Comanche Peak 2	PWR Class B	Prairie Island 2	PWR Class B
Cook 1	PWR Class B	Quad Cities 1	BWR Class C
Cook 2	PWR Class B	Quad Cities 2	BWR Class C
Cooper Station	BWR Class C	Rancho Seco	PWR Class D
Crystal River 3	PWR Class D	River Bend 1	BWR Class C
Davis-Besse	PWR Class B	Robinson 2	PWR Class B
Diablo Canyon 1	PWR Class B	Salem 1	PWR Class B
Diablo Canyon 2	PWR Class B	Salem 2	PWR Class B
Dresden 2	BWR Class B	San Onofre 1	Unique
Dresden 3	BWR Class B	San Onofre 2	PWR Class H
Duane Arnold	BWR Class C	San Onofre 3	PWR Class H
Farley 1	PWR Class B	Seabrook 1	PWR Class B
Farley 2	PWR Class B	Sequoyah 1	PWR Class B
Fermi 2	BWR Class C	Sequoyah 2	PWR Class B
Fitzpatrick	BWR Class C	South Texas 1	PWR Class B
Fort Calhoun	PWR Class G	South Texas 2	PWR Class B
Ginna	PWR Class B	St. Lucie 1	PWR Class G
Grand Gulf 1	BWR Class C	St. Lucie 2	PWR Class G
Haddam Neck	PWR Class B	Summer 1	PWR Class B
Harris 1	PWR Class B	Surry 1	PWR Class A
Hatch 1	BWR Class C	Surry 2	PWR Class A
Hatch 2	BWR Class C	Susquehanna 1	BWR Class C
Hope Creek 1	BWR Class C	Susquehanna 2	BWR Class C
Indian Point 2	PWR Class B	Three Mile Island 1	PWR Class D
Indian Point 3	PWR Class B	Trojan	PWR Class B
Kewaunee	PWR Class B	Turkey Point 3	PWR Class B
LaCrosse	Unique	Turkey Point 4	PWR Class B
LaSalle 1	BWR Class C	Vermont Yankee	BWR Class C
LaSalle 2	BWR Class C	Vogtle 1	PWR Class B
Limerick 1	BWR Class C	Vogtle 2	PWR Class B
Limerick 2	BWR Class C	WNPSS 2	BWR Class C
Maine Yankee	PWR Class B	Waterford 3	PWR Class H
McGuire 1	PWR Class B	Wolf Creek 1	PWR Class B
McGuire 2	PWR Class B	Yankee Rowe	PWR Class B
Millstone 1	BWR Class A	Zion 1	PWR Class B
Millstone 2	PWR Class G	Zion 2	PWR Class B

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
11	Core damage	Unavailability of HPR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI. (PWR Classes A, B, D, G, and H)
12	Core damage	Unavailability of HPI following successful trip and AFW initiation, primary relief valve lift, and primary relief valve failure to reseat. (PWR Classes A, B, D, G, and H)
13	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
14	Core damage	Similar to sequence 12, but MFW provides SG cooling in lieu of AFW. (PWR Classes A, B, D, G, and H)
15	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but the PORV fails to open. (PWR Classes A, B, and G)
16	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is initiated, but fails in the recirculation phase. (PWR Classes A, B, D, and G)
17	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed fails in the injection phase. (PWR Classes A, B, D, and G)
18	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models. (PWR Classes A, B, D, G, and H)
19	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful but CSR is unavailable. (PWR Class G)
20	Core damage	Unavailability of CSR following successful trip and AFW initiation, primary relief valve lift and failure to reseat, and successful HPI and HPR. (PWR Class A)
21	Core damage	Similar to sequence 11, but MFW provides SG cooling in lieu of AFW. (PWR Class A)
22	Core damage	Unavailability of AFW and MFW following successful trip. Feed and bleed is successful, but CSR is unavailable for containment heat removal. This sequence is distinguished from sequence 19 because of differences in the function of CSR on Class A and G plants. (PWR Class A)

Table A.4 PWR transient core damage and ATWS sequences

Sequence No.	End state	Description
23	Core damage	Unavailability of AFW and MFW following successful trip. The SGs are successfully depressurized, but the condensate pumps fail to provide SG cooling. (PWR Class H)
24	Core damage	Unavailability of AFW and MFW following successful trip, plus failure to depressurize the SGs to allow for the use of the condensate pumps for SG cooling. (PWR Class H)
25	Core damage	Unavailability of AFW and MFW following successful trip. At least one open SRV fails to reseat, but HPI and HPR are successful. SG depressurization is successful, but the condensate pumps fail to provide SG cooling. (PWR Class H)
26	Core damage	Similar to sequence 25 except that SG depressurization fails. (PWR Class H)
27	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat. HPI is initiated but HPR fails. (PWR Class H)
28	Core damage	Unavailability of AFW and MFW following successful trip. At least one SRV fails to reseat and HPI fails. (PWR Class H)

Table A.5 PWR transient sequences summary

Seq. No.	End State	RT	AFW	MFW	RV Chall	RV Reseat	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	B	D	G	H
11	CD	S	S		S*	F	S	F					x	x	x	x	x
12	CD	S	S		S*	F	F						x	x	x	x	x
13	CD	S	F	S	S*	F	S	F					x	x	x	x	x
14	CD	S	F	S	S*	F	F						x	x	x	x	x
15	CD	S	F	F			S	S	F				x	x		x	
16	CD	S	F	F			S	F					x	x	x	x	
17	CD	S	F	F			F						x	x	x	x	
18	ATWS	F											x	x	x	x	x
19	CD	S	F	F			S	S	S	F						x	
20	CD	S	S		S*	F	S	S		F			x				
21	CD	S	F	S	S*	F	S	S		F			x				
22	CD	S	F	F			S	S	S	F			x				
23	CD	S	F	F		S					S	F					x
24	CD	S	F	F		S					F						x
25	CD	S	F	F		F	S	S			S	F					x
26	CD	S	F	F		F	S	S			F						x
27	CD	S	F	F		F	S	F									x
28	CD	S	F	F		F	F										x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
40	ATWS	Failure to trip following a LOOP. (PWR Classes A, B, D, G, and H)
41	Core damage	Unavailability of HPR following a LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat; and successful HPI. (PWR Classes A, B, D, G, and H)
42	Core damage	Unavailability of HPI following LOOP with successful trip, emergency power, and AFW; primary relief valve lift and failure to reseat. (PWR Classes A, B, D, G, and H)
43	Core damage	Failure of the PORV to open for feed and bleed cooling following successful trip and emergency power, and AFW failure. (PWR Classes A, B, and G)
44	Core damage	Failure of HPR for recirculation cooling following feed and bleed initiation. Trip and emergency power are successful, but AFW fails. (PWR Classes A, B, D, and G)
45	Core damage	Unavailability of HPI for feed and bleed cooling following successful trip and emergency power and AFW failure. (PWR Classes A, B, D, and G)
46	Core damage	Unavailability of HPR following HPI success for RCP seal LOCA mitigation. AC power is recovered following successful trip, emergency power failure, turbine-driven AFW train(s) success, primary relief valve lift and reseat, and a subsequent seal LOCA. (PWR Classes A, B, D, G, and H)
47	Core damage	This sequence is similar to sequence 46 except that HPI fails for RCP seal LOCA mitigation. (PWR Classes A, B, D, G, and H)
48	Core damage	Failure to recover AC power following an RCP seal LOCA. The seal LOCA occurs following successful trip, failure of emergency power, turbine-driven AFW train(s) success, and primary relief valve lift and closure. (PWR Classes A, B, D, G, and H)
49	Core damage	Failure to recover AC power following successful trip and emergency power system failure, AFW turbine train(s) success, and primary relief valve lift and reseat. No RCP seal LOCA occurs in the sequence. (PWR Classes A, B, D, G, and H)
50	Core damage	Failure of a primary relief valve to reseat following lift subsequent to a successful trip, emergency power system failure, and AFW turbine trains(s) success. (PWR Classes A, B, D, G, and H)

Table A.6 PWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
51	Core damage	This sequence is similar to sequence 46 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
52	Core damage	This sequence is similar to sequence 47 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
53	Core damage	This sequence is similar to sequence 48 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
54	Core damage	This sequence is similar to sequence 49 except that the primary relief valves are not challenged. (PWR Classes A, B, D, G, and H)
55	Core damage	Failure of AFW following successful trip and emergency power system failure (PWR Classes A, B, D, G, and H)
56	Core damage	Failure of CSR in conjunction with successful feed and bleed following trip, emergency power system success, and AFW failure (PWR Class G)
57	Core damage	Failure of CSR following LOOP with successful trip, emergency power and AFW, primary relief valve challenge and failure to reseat, and successful HPI and HPR. (PWR Class A)
58	Core damage	Failure of CSR in conjunction with successful feed and bleed following LOOP with successful trip and emergency power initiation, and AFW failure. (PWR Class A)
59	Core damage	Failure of CSR following successful HPI and HPR required to mitigate a seal LOCA. This sequence involves a LOOP with successful trip, emergency power system failure, primary relief valve challenge and reseat, and a subsequent seal LOCA with AC power recovery prior to core uncovery. (PWR Class A)
60	Core damage	This sequence is similar to sequence 59 except that the primary relief valves are not challenged. (PWR Class A)
61	Core damage	Failure of AFW following a LOOP with successful trip and emergency power. (PWR Class H)

Table A.7 PWR LOOP sequences summary

Seq. No.	End State	RT/ LOOP	EP	AFW	RV Chall	RV Reseat	Seal LOCA	EP Recov	HPI	HPR	PORV Open	CSR	PWR Class A	B	D	G	H
40	ATWS	F											x	x	x	x	x
41	CD	S	S	S	S*	F			S	F			x	x	x	x	x
42	CD	S	S	S	S*	F			F				x	x	x	x	x
43	CD	S	S	F					S	S	F		x	x		x	
44	CD	S	S	F					S	F			x	x	x	x	
45	CD	S	S	F					F				x	x	x	x	
46	CD	S	F	S	S*	S	S*	S	S	F			x	x	x	x	x
47	CD	S	F	S	S*	S	S*	S	F				x	x	x	x	x
48	CD	S	F	S	S*	S	S*	F					x	x	x	x	x
49	CD	S	F	S	S*	S		F					x	x	x	x	x
50	CD	S	F	S	S*	F							x	x	x	x	x
51	CD	S	F	S			S*	S	S	F			x	x	x	x	x
52	CD	S	F	S			S*	S	F				x	x	x	x	x
53	CD	S	F	S			S*	F					x	x	x	x	x
54	CD	S	F	S				F					x	x	x	x	x
55	CD	S	F	F									x	x	x	x	x
56	CD	S	S	F					S	S	S	F				x	
57	CD	S	S	S	S*	F			S	S		F	x				
58	CD	S	S	F					S	S	S	F	x				
59	CD	S	F	S	S*	S	S*	S	S	S		F	x				
60	CD	S	F	S			S*	S	S	S		F	x				
61	CD	S	S	F													x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.8 PWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
71	Core damage	Unavailability of HPR following a small-break LOCA with trip, AFW and HPI success. (PWR Classes A, B, D, G, and H)
72	Core damage	Unavailability of HPI following a small-break LOCA with trip and AFW success. (PWR Classes A, B, D, G, and H)
73	Core damage	This sequence is similar to sequence 71 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
74	Core damage	This sequence is similar to sequence 72 except that MFW is utilized for SG cooling is AFW is unavailable. (PWR Classes A, B, D, G, and H)
75	Core damage	Unavailability of AFW and MFW following a small-break LOCA and successful trip. The PORV is unavailable to depressurize the RCS to the HPI pump discharge pressure. (PWR Classes A, B, and G)
76	Core damage	Unavailability of AFW and MFW following a small-break LOCA with trip success. HPI is successful but HPR fails. (PWR Classes A, B, D, G, and H)
77	Core damage	Unavailability of AFW and MFW following trip success. HPI fails to provide RCS makeup. (PWR Classes A, B, D, G, and H)
78	ATWS	Failure of reactor trip following a small-break LOCA. (PWR Classes A, B, D, G, and H)
79	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success, AFW and MFW failure, and feed and bleed success. (PWR Class G)
80	Core damage	Unavailability of CSR following a small-break LOCA with trip, AFW, HPI and HPR success. (PWR Class A)
81	Core damage	This sequence is similar to sequence 80 except that MFW is used for SG cooling in the event AFW is unavailable. (PWR Class A)
82	Core damage	Unavailability of CSR for containment heat removal following a small-break LOCA with trip success, AFW and MFW unavailability, and feed and bleed success. (PWR Class A)
83	Core damage	Unavailability of the condensate pumps for SG cooling following a small-break LOCA with trip success, unavailability of AFW and MFW, and successful SG depressurization. (PWR Class H)
84	Core damage	This sequence is similar to sequence 83 except that SG depressurization is unavailable. (PWP. Class H)

Table A.9 PWR small-break LOCA sequences summary

Seq. No.	End State	RT	AFW	MFW	HPI	HPR	PORV Open	CSR	SG Dep	Condensate Pumps	PWR Class A	B	D	G	H
71	CD	S	S		S	F					x	x	x	x	x
72	CD	S	S		F						x	x	x	x	x
73	CD	S	F	S	S	F					x	x	x	x	x
74	CD	S	F	S	F						x	x	x	x	x
75	CD	S	F	F	S	S	F				x	x		x	
76	CD	S	F	F	S	F					x	x	x	x	x
77	CD	S	F	F	F						x	x	x	x	x
78	ATWS	F									x	x	x	x	x
79	CD	S	F	F	S	S	S	F						x	
80	CD	S	S		S	S		F			x				
81	CD	S	F	S	S	S		F			x				
82	CD	S	F	F	S	S	S	F			x				
83	CD	S	F	F	S	S			S	F					x
84	CD	S	F	F	S	S			F						x

Note: CD - Core damage.
S - Required and successfully performs its function.
F - Required and fails to perform its function.
S^* - Relief valve challenged during the transient (assumed for all losses of both AFW and MFW).

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of isolation condenser, and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater and successful feedwater coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and feedwater coolant injection, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, feedwater coolant injection and control rod drive cooling; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and success of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
16	Core damage	Similar to Sequence 15 except the shutdown cooling system fails followed by successful containment cooling.
17	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and failure of low-pressure core spray.
18	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, feedwater coolant injection, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
19	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
20	Core damage	Similar to Sequence 19 except unsuccessful main feedwater followed by successful feedwater coolant injection.
21	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief challenge and unsuccessful reseat, unsuccessful main feedwater and followed by successful vessel depressurization and low-pressure core spray.
22	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray.
23	Core damage	Similar to Sequence 22 except failure of the shutdown cooling system and successful containment spray.
24	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, unsuccessful main feedwater and feedwater coolant injection, successful vessel depressurization, and unsuccessful low-pressure core spray.
25	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and feedwater coolant injection.
26	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
27	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
28	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
30	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
11	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and failure of isolation condenser and successful main feedwater.
12	Core damage	Similar to Sequence 11 except failure of main feedwater followed by successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, followed by successful control rod drive cooling.
14	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization and low-pressure core spray.
15	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of isolation condenser; failure of main feedwater, high-pressure coolant injection, and control rod drive cooling systems; followed by successful vessel depressurization, and failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
16	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; and failure of isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.
17	Core damage	Similar to Sequence 16 except the shutdown cooling system fails followed by successful containment cooling mode of the low-pressure coolant injection system.
18	Core damage	Similar to Sequence 15 except low-pressure coolant injection system fails.
19	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, and safety relief valve challenge and successful reseat. Failure of the isolation condenser, main feedwater, high-pressure coolant injection, and control rod drive cooling.
20	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure injection) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and successful main feedwater.
21	Core damage	Similar to Sequence 20 except unsuccessful main feedwater followed by successful high-pressure coolant injection.
22	Core damage	Similar to Sequence 20 except unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray.
23	Core damage	Similar to Sequence 20 except failure of main feedwater and high-pressure coolant injection, followed by successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.
24	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
25	Core damage	Similar to Sequence 24 except failure of the shutdown cooling system and successful containment spray mode of low-pressure core injection.
26	Core damage	Similar to Sequence 23 except unsuccessful low-pressure coolant injection.
27	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection.
28	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
29	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
32	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
11	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, and successful main feedwater.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
12	Core damage	Similar to Sequence 11 except failure of main feedwater with successful high-pressure coolant injection.
13	Core damage	Similar to Sequence 11 except failure of main feedwater and high-pressure coolant injection systems, with successful reactor core isolation cooling.
14	Core damage	Similar to Sequence 11 except failure of main feedwater, high-pressure coolant injection, and reactor core isolation cooling, with successful control rod drive cooling.
15	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown cooling and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat, failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
16	Core damage	Similar to Sequence 15 except failure of low-pressure core spray and successful low-pressure coolant injection.
17	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation; safety relief valve challenge and successful reseat; failure of main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
18	Core damage	Similar to Sequence 17 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
19	Core damage	Similar to Sequence 16 except failure of low-pressure coolant injection.
20	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and successful reseat. Failure of the main feedwater, high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
21	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, and successful main feedwater.
22	Core damage	Similar to Sequence 21 except unsuccessful main feedwater with successful high-pressure coolant injection.
23	Core damage	Unavailability of long-term core cooling (residual heat removal shutdown and suppression pool cooling modes fail) following successful scram and failure of continued power conversion system operation, safety relief valve challenge with unsuccessful reseat, unsuccessful main feedwater and high-pressure coolant injection, followed by successful vessel depressurization and low-pressure core spray
24	Core damage	Similar to Sequence 23 except failure of low-pressure core spray and successful low-pressure coolant injection.
25	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of main feedwater and high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
26	Core damage	Similar to Sequence 25 except the residual heat removal system fails in the shutdown cooling mode and succeeds in the suppression pool cooling mode.
27	Core damage	Similar to Sequence 24 except failure of low-pressure coolant injection.
28	Core damage	Unavailability of vessel depressurization following successful scram and failure of continued power conversion system operation, safety relief valve challenge and unsuccessful reseat, and failure of the main feedwater and high-pressure coolant injection systems.
29	Core damage	Similar to Sequence 11 except the safety relief valves are not challenged.
30	Core damage	Similar to Sequence 12 except the safety relief valves are not challenged.
31	Core damage	Similar to Sequence 13 except the safety relief valves are not challenged.

Table A.10 BWR transient core damage and ATWS sequences

Sequence No.	End state	Description
32	Core damage	Similar to Sequence 14 except the safety relief valves are not challenged.
33	Core damage	Similar to Sequence 15 except the safety relief valves are not challenged.
34	Core damage	Similar to Sequence 16 except the safety relief valves are not challenged.
35	Core damage	Similar to Sequence 17 except the safety relief valves are not challenged.
36	Core damage	Similar to Sequence 18 except the safety relief valves are not challenged.
37	Core damage	Similar to Sequence 19 except the safety relief valves are not challenged.
38	Core damage	Similar to Sequence 20 except the safety relief valves are not challenged.
99	ATWS	Failure to trip following a transient requiring trip. ATWS sequences are not further developed in the ASP models.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat. Failure of isolation condenser and successful feedwater coolant injection.
42	Core damage	Similar to Sequence 41 except failure of the feedwater coolant injection and successful control rod drive cooling.
43	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the feedwater coolant injection and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, feedwater coolant injection, and control rod drive cooling. Successful vessel depressurization and failure of low-pressure core spray.
45	Core damage	Similar to Sequence 44 except failure of the shutdown cooling system and successful containment spray.
46	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of feedwater coolant injection and control rod drive cooling, with successful vessel depressurization and failure of the low-pressure core spray.
47	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, feedwater coolant injection, and control rod drive cooling.
48	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful feedwater coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
49	Core damage	Similar to Sequence 48 except failure of feedwater coolant injection followed by successful vessel depressurization and low-pressure core spray.
50	Core damage	Unavailability of fire water or other equivalent water source for vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of feedwater coolant injection. Successful vessel depressurization, failure of low-pressure core spray, and successful shutdown cooling system.
51	Core damage	Similar to Sequence 50 except failure of shutdown cooling system and successful containment cooling.
52	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of feedwater coolant injection, successful vessel depressurization, and failure of low-pressure core spray.
53	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the feedwater coolant injection system.
54	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
55	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
56	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged..
60	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged
61	Core damage	Unavailability of the isolation condenser following a loss of offsite power, failure of emergency power, successful scram, and safety relief valve challenge and successful reseat.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
62	Core damage	Failure of an SRV to reseat following challenge after a loss of offsite power with failure of emergency power and successful reactor scram.
63	Core damage	Similar to Sequence 61 except the safety relief valves are not challenged.
64	Core damage	Failure of recovery of electric power in the long-term following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
41	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser and successful high-pressure coolant injection.
42	Core damage	Similar to Sequence 41 except failure of high-pressure coolant injection and successful control rod drive cooling.
43	Core damage	Similar to Sequence 41 except failure of the high-pressure coolant injection and control rod drive cooling, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and reseat. Failure of isolation condenser, failure of the high-pressure coolant injection and control rod drive cooling systems, with successful vessel depressurization, failure of low-pressure core spray, and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
45	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of isolation condenser, high-pressure coolant injection, and control rod drive cooling. Successful vessel depressurization, failure of low-pressure core spray, and low-pressure coolant injection with successful shutdown cooling.
46	Core damage	Similar to Sequence 45 except failure of the shutdown cooling system and successful containment spray mode low-pressure coolant injection.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram, challenge of the safety relief valves and successful reseat with unsuccessful isolation condenser, high-pressure coolant injection, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray.
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.
52	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure core injection, and successful shutdown cooling system.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
53	Core damage	Similar to Sequence 52 except failure of shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
54	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power with successful emergency power, reactor scram, and safety relief valve challenge and unsuccessful reseat. Failure of high-pressure coolant injection, successful vessel depressurization and failure of low-pressure core spray and low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.
64	Core damage	Unavailability of long-term cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, failed isolation condenser, and successful high-pressure coolant injection.
65	Core damage	Unavailability of high-pressure core injection following a loss of offsite power, failure of emergency power, successful reactor scram, safety relief valve challenge and reseat, and failed isolation condenser and high-pressure coolant injection systems.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
66	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and successful high-pressure coolant injection.
67	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and failure to reseat, and failure of high-pressure coolant injection.
68	Core damage	Similar to Sequence 64 except the safety relief valves are not challenged.
69	Core damage	Similar to Sequence 65 except the safety relief valves are not challenged.
84	Core damage	Failure of long-term recovery of electric power following a loss of offsite power, with failure of emergency power and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
40	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
41	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection system and successful reactor core isolation cooling.
42	Core damage	Similar to Sequence 40 except failure of the high-pressure coolant injection and reactor core isolation cooling systems with successful control rod drive cooling.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
43	Core damage	Unavailability of long-term core cooling (failure of residual heat removal in shutdown and suppression cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and reseat; failure of the high-pressure coolant injection, reactor core isolation cooling and control rod drive cooling systems, with successful vessel depressurization and low-pressure core spray.
44	Core damage	Similar to Sequence 43 except failure of low-pressure core spray and successful low-pressure coolant injection.
45	Core damage	Unavailability of fire water or other equivalent water source for reactor makeup following a loss of offsite power with successful emergency power, scram, and safety relief valve challenge and successful reseat. Failure of high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling systems. Successful vessel depressurization, and failure of low-pressure core spray and low-pressure coolant injection with successful residual heat removal in shutdown cooling mode.
46	Core damage	Similar to Sequence 45 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
47	Core damage	Similar to Sequence 44 except failure of low-pressure coolant injection.
48	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Challenge of the safety relief valves and successful reseat with high-pressure coolant injection, reactor core isolation cooling, and control rod drive cooling.
49	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and successful high-pressure coolant injection.
50	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power with successful emergency power, reactor scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection followed by successful vessel depressurization and low-pressure core spray
51	Core damage	Similar to Sequence 50 except failure of low-pressure core spray and successful low-pressure coolant injection.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
52	Core damage	Unavailability of fire water or other equivalent water source following a loss of offsite power, successful emergency power and scram, safety relief valve challenge and unsuccessful reseat, and failure of high-pressure coolant injection. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal in shutdown cooling mode.
53	Core damage	Similar to Sequence 52 except failure of the residual heat removal system in shutdown cooling mode and success in suppression pool cooling mode.
54	Core damage	Similar to Sequence 51 except failure of low-pressure coolant injection.
55	Core damage	Unavailability of vessel depressurization following a loss of offsite power with successful emergency power and reactor scram. Safety relief valve challenge and unsuccessful reseat, and failure of the high-pressure coolant injection system.
56	Core damage	Similar to Sequence 40 except the safety relief valves are not challenged.
57	Core damage	Similar to Sequence 41 except the safety relief valves are not challenged.
58	Core damage	Similar to Sequence 42 except the safety relief valves are not challenged.
59	Core damage	Similar to Sequence 43 except the safety relief valves are not challenged.
60	Core damage	Similar to Sequence 44 except the safety relief valves are not challenged.
61	Core damage	Similar to Sequence 45 except the safety relief valves are not challenged.
62	Core damage	Similar to Sequence 46 except the safety relief valves are not challenged.
63	Core damage	Similar to Sequence 47 except the safety relief valves are not challenged.
64	Core damage	Similar to Sequence 48 except the safety relief valves are not challenged.

Table A.11 BWR LOOP core damage and ATWS sequences

Sequence No.	End state	Description
65	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, and successful high-pressure coolant injection.
66	Core damage	Similar to Sequence 65 except high-pressure coolant injection fails with successful reactor core isolation cooling.
67	Core damage	Unavailability of long-term core cooling (failure of the residual heat removal system in shutdown and suppression pool cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, successful long-term recovery of electric power, safety relief valve challenge and reseat, with failures of high-pressure coolant injection and reactor core isolation cooling.
68	Core damage	Similar to Sequence 65 except the safety relief valves fail to reseat.
69	Core damage	Failure of high-pressure coolant injection following a loss of offsite power, with emergency power failure, successful reactor scram, safety relief valve challenge, and unsuccessful reseat.
80	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression cooling modes) following a loss of offsite power, failure of emergency power, successful reactor scram, and long-term recovery of electric power. The safety relief valves are not challenged, and high-pressure coolant injection is successful.
81	Core damage	Similar to Sequence 66 except the safety relief valves are not challenged.
82	Core damage	Similar to Sequence 67 except the safety relief valves are not challenged.
83	Core damage	Unable to recover long-term electric power following a loss of offsite power, failure of emergency power, and successful reactor scram.
97	ATWS	ATWS following a loss of offsite power and unavailability of emergency power. ATWS sequences are not further developed in the ASP models.
98	ATWS	ATWS following a loss of offsite power, successful emergency power, and failure to scram the reactor. ATWS sequences are not further developed in the ASP models.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
		BWR Class A sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, and successful feedwater coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling) following a loss-of-coolant accident, successful scram, failure of feedwater coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of feedwater coolant injection. Successful vessel depressurization and failure of low-pressure core spray, and successful shutdown cooling system.
74	Core damage	Similar to Sequence 73 except failure of the shutdown cooling system and successful containment cooling.
75	Core damage	Similar to Sequence 72 except failure of the low-pressure core spray.
76	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the feedwater coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class B sequences
71	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of shutdown cooling system and containment cooling mode of low-pressure coolant injection) following a loss-of-coolant accident, successful scram, failure of high-pressure coolant injection, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful shutdown cooling system.

Table A.12 BWR small-break LOCA core damage and ATWS sequences

Sequence No.	End state	Description
75	Core damage	Similar to Sequence 74 except failure of the shutdown cooling system and successful containment cooling mode of low-pressure coolant injection.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.
		BWR Class C sequences
71	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, and successful high-pressure coolant injection.
72	Core damage	Unavailability of long-term core cooling (failure of residual heat removal system in shutdown and suppression pool cooling modes) following a loss-of-coolant accident, successful scram, failure of the high-pressure coolant injection system, and successful vessel depressurization and low-pressure core spray.
73	Core damage	Similar to Sequence 72 except failure of low-pressure core spray, and successful low-pressure coolant injection.
74	Core damage	Unavailability of fire water or other equivalent water source for reactor vessel makeup following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system. Successful vessel depressurization, failure of low-pressure core spray and low-pressure coolant injection, and successful residual heat removal system in shutdown cooling mode.
75	Core damage	Similar to Sequence 74 except failure of the residual heat removal system in the shutdown cooling mode and success in the suppression pool cooling mode.
76	Core damage	Similar to Sequence 73 except failure of low-pressure coolant injection.
77	Core damage	Unavailability of vessel depressurization following a loss-of-coolant accident, successful reactor scram, and failure of the high-pressure coolant injection system.
96	ATWS	ATWS following a loss-of-coolant accident. ATWS sequences are not further developed in the ASP models.

Table A.13 Average initiating event frequency and branch failure probability estimates developed from 1984-1986 precursors.

Initiator/branch	Initial estimate (no recovery attempted)	Nonrecovery estimate	Total
	PWRs		
LOOP	4.1×10^{-2}/year	0.39	1.6×10^{-2}/year[a]
Small-break LOCA	1.5×10^{-2}/year	0.43	6.4×10^{-3}/year
Auxiliary feedwater	3.8×10^{-4}	0.26	9.9×10^{-5}
High-pressure injection	6.1×10^{-4}	0.84	5.1×10^{-4}
Long-term core cooling (high-pressure recirculation)	1.5×10^{-4}	1.00	1.5×10^{-4}
Emergency power	6.4×10^{-4}	0.78	5.0×10^{-4}
SG isolation (MSIVs)	8.3×10^{-4}	0.64	5.3×10^{-4}
	BWRs		
LOOP	1.0×10^{-1}/year	0.32	3.3×10^{-2}/year[a]
Small-break LOCA	2.0×10^{-2}/year	0.50	1.0×10^{-2}/year
HPCI/RCIC	1.7×10^{-3}	0.49	8.4×10^{-4}
RV isolation	1.7×10^{-3}	1.00	1.7×10^{-3}
LPCI	1.0×10^{-3}	0.71	7.4×10^{-4}
Emergency power	1.0×10^{-4}	0.85	8.9×10^{-5}
Automatic depressurization	3.7×10^{-3}	0.71	2.6×10^{-3}

[a]Precursor calculations utilize plant-specific LOOP frequency estimates developed from information in P.W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, June 1988.

Table A.14 Operator action failure probabilities

Operation action	Failure probability
BWRs	
Condensate/feedwater recovery	0.001
Containment venting	0.01
Control rod drive water use	0.01
Initiation of RHR service water, fire water	0.01
Shutdown cooling	0.001
Standby liquid control initiation	0.01
PWRs	
Condensate/MFW recovery	0.01
Containment spray recirculation	0.001
Emergency core cooling recirculation	0.001
Fail to block stuck-open PORVs	0.001
Open PORVs for feed and bleed	0.0004
SG depressurization	0.001
Use feed and bleed to cool core	0.01

Table A.15 Reference event conditional probability values

Postulated operational event	Conditional core damage probability
BWR Class A nonspecific reactor trip	2.8×10^{-6}
BWR Class A LOFW	1.7×10^{-4}
BWR Class B nonspecific reactor trip	7.7×10^{-8}
BWR Class B LOFW	4.3×10^{-6}
BWR Class C (turbine-driven feed pumps) nonspecific reactor trip	1.2×10^{-6}
BWR Class C (turbine-driven feed pumps) LOFW	1.5×10^{-5}
PWR Class A nonspecific reactor trip	1.8×10^{-7}
PWR Class A LOFW	2.4×10^{-6}
PWR Class B nonspecific reactor trip	1.8×10^{-7}
PWR Class B LOFW	2.2×10^{-6}
PWR Class D nonspecific reactor trip	4.7×10^{-7}
PWR Class D LOFW	6.8×10^{-6}
PWR Class G nonspecific reactor trip	1.8×10^{-7}
PWR Class G LOFW	2.4×10^{-6}
PWR Class H nonspecific reactor trip	4.9×10^{-6}
PWR Class H LOFW	3.9×10^{-5}
BWR Class C HPCI unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.0×10^{-5}
BWR Class C HPCS unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	1.4×10^{-5}
BWR Class C RCIC unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	3.8×10^{-8}
BWR Class C CRD cooling unavailability (turbine-driven feed pumps, 360-h unavailability)[a]	6.2×10^{-8}

[a]The probability of a transient, LOOP, or small-break LOCA during the 360-h unavailability was estimated as described in Sect. A.1.

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	PWR event trees
AFW	auxiliary feedwater fails
ATWS	anticipated transient without scram end state
COND	condensate system fails
CD	core damage end state
CSR	containment spray recirculation fails
EP	emergency power fails
EP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
HPI	high-pressure injection fails
HPR	high-pressure recirculation fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
MFW	main feedwater fails
PORV OPEN	power-operated relief valve fails to open for feed and bleed cooling
PORV/SRV CHALL	power-operated relief valve or safety relief valves challenged (challenge rate)
PORV/SRV RESEAT	power-operated relief valve and/or safety relief valve fails to reseat
RT	reactor trip fails
RT/LOOP	reactor trip fails given a loss of offsite power
SEAL LOCA	RCP seal LOCA occurs
SEC SIDE DEP	secondary-side depressurization fails
SEQ NO	sequence number
SRV CHALL	safety relief valves challenged
SRV RESEAT	safety relief valve fails to reseat
TRANS	nonspecific reactor-trip transient

Table A.16 Abbreviations used in event trees

Abbreviation	Description
	BWR Event Trees
CC	containment cooling fails
CRD	control-rod-drive cooling fails
EP	emergency power fails
FIREWTR or OTHER	fire water or other equivalent water source fails
FW	unavailabilty of main feedwater
FWCI	failure of feedwater coolant injection system
HPCI OR HPCS	high-pressure coolant injection or high-pressure core spray fails
IC/IP MUP	isolation condenser or isolation condenser makeup fails
LOCA	small-break loss-of-coolant accident
LOOP	loss of offsite power
LOOP REC (LONG)	long-term recovery from LOOP or emergency power failure fails
LPCI	low-pressure coolant injection fails
LPCI (CC MODE)	containment cooling mode of low-pressure coolant injection system fails
LPCI (RHR)	residual heat removal mode of low-pressure coolant injection core spray fails
LPCS	low-pressure core spray fails
PCS	failure of continued power conversion system operation
RCIC	reactor core isolation cooling fails
RHR (SDC MODE)	residual-heat-removal shutdown cooling mode fails
RHR (SP COOLING MODE)	residual-heat-removal suppression pool cooling mode fails
RHR SW or OTHER	residual-heat-removal service water or other water source fails
RX SHUTDOWN	reactor fails to scram
SDC	shutdown cooling system fails
SRVs/ADS	safety relief valve(s) fail to open for depressurization or automatic depressurization system fails
SRV CHAL	safety relief valve(s) challenged (challenge rate)
SRV-C	safety relief valve fails to close
TRANSIENT	nonspecific reactor-trip transient

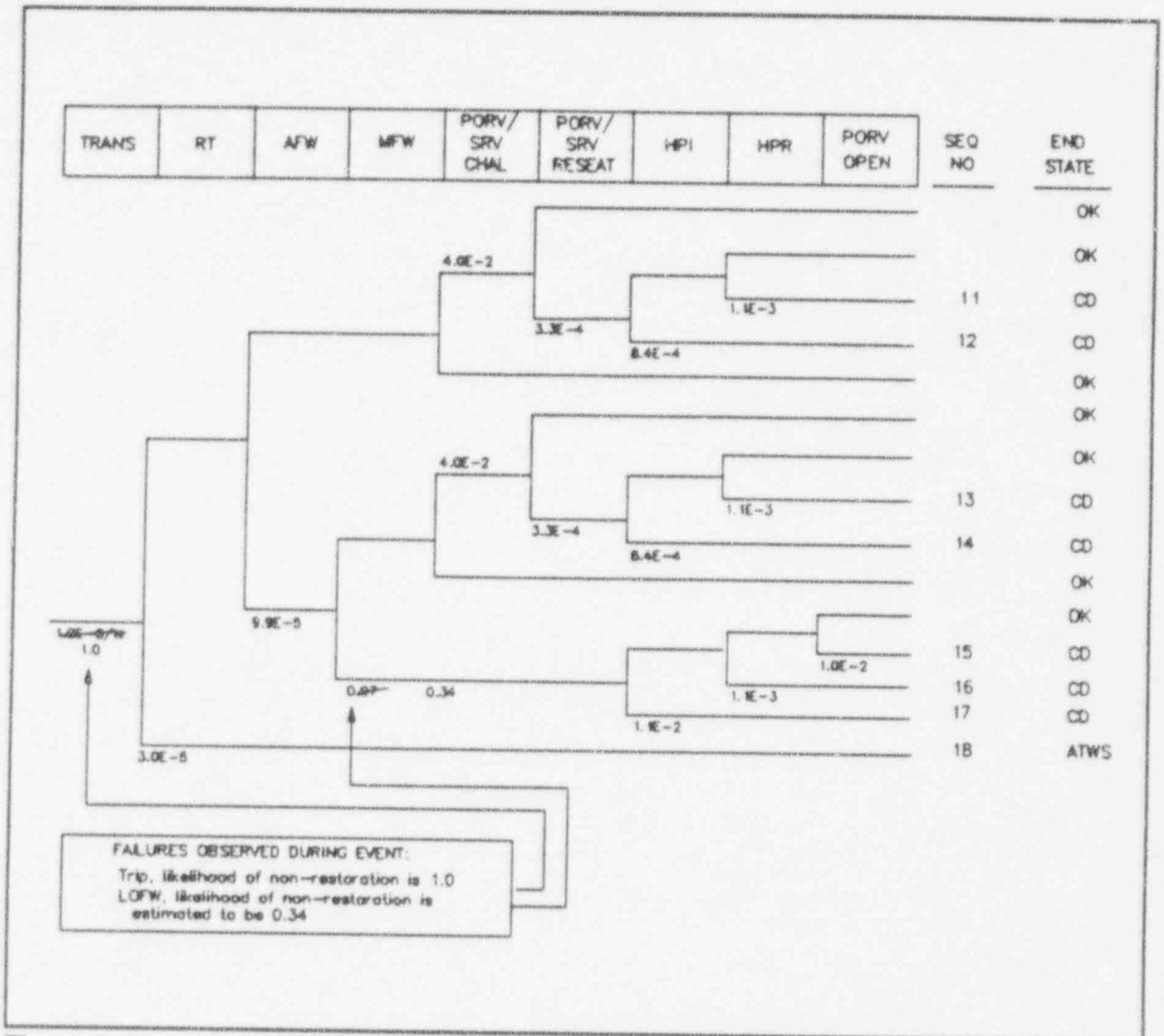

Fig. A.1. Example initiator calculation.

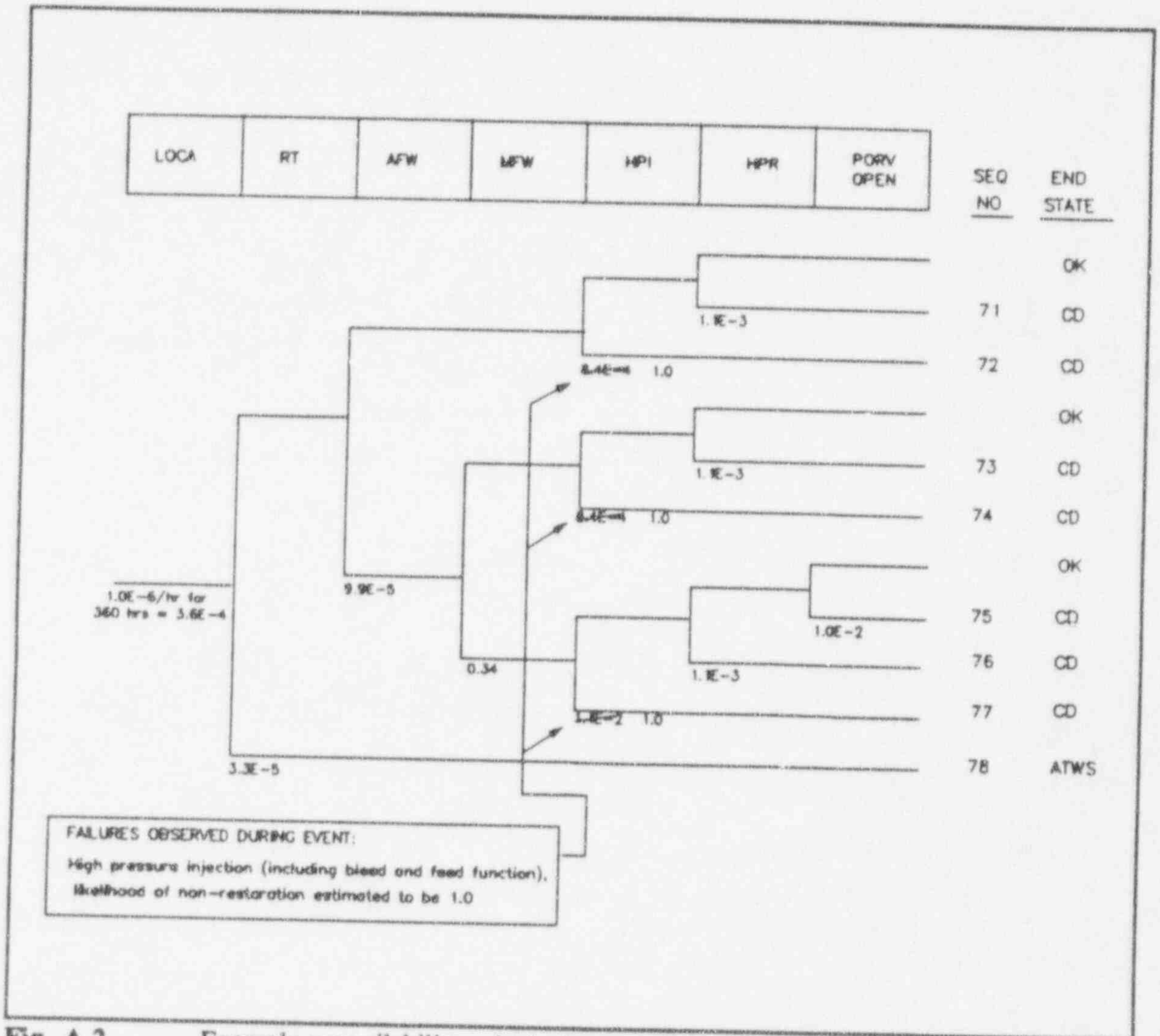

Fig. A.2. Example unavailability calculation

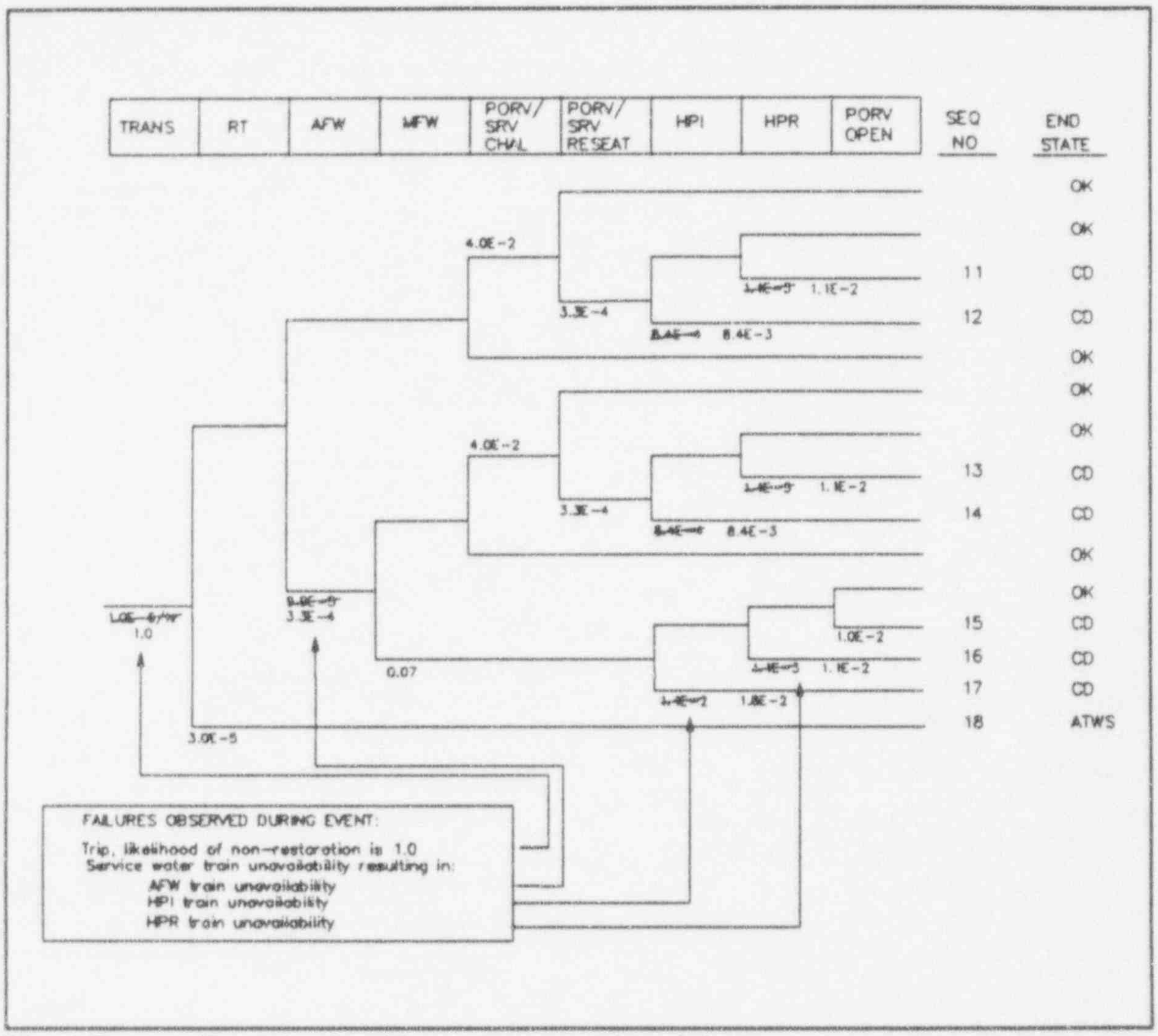

Fig. A.3. Example trip with support system degraded

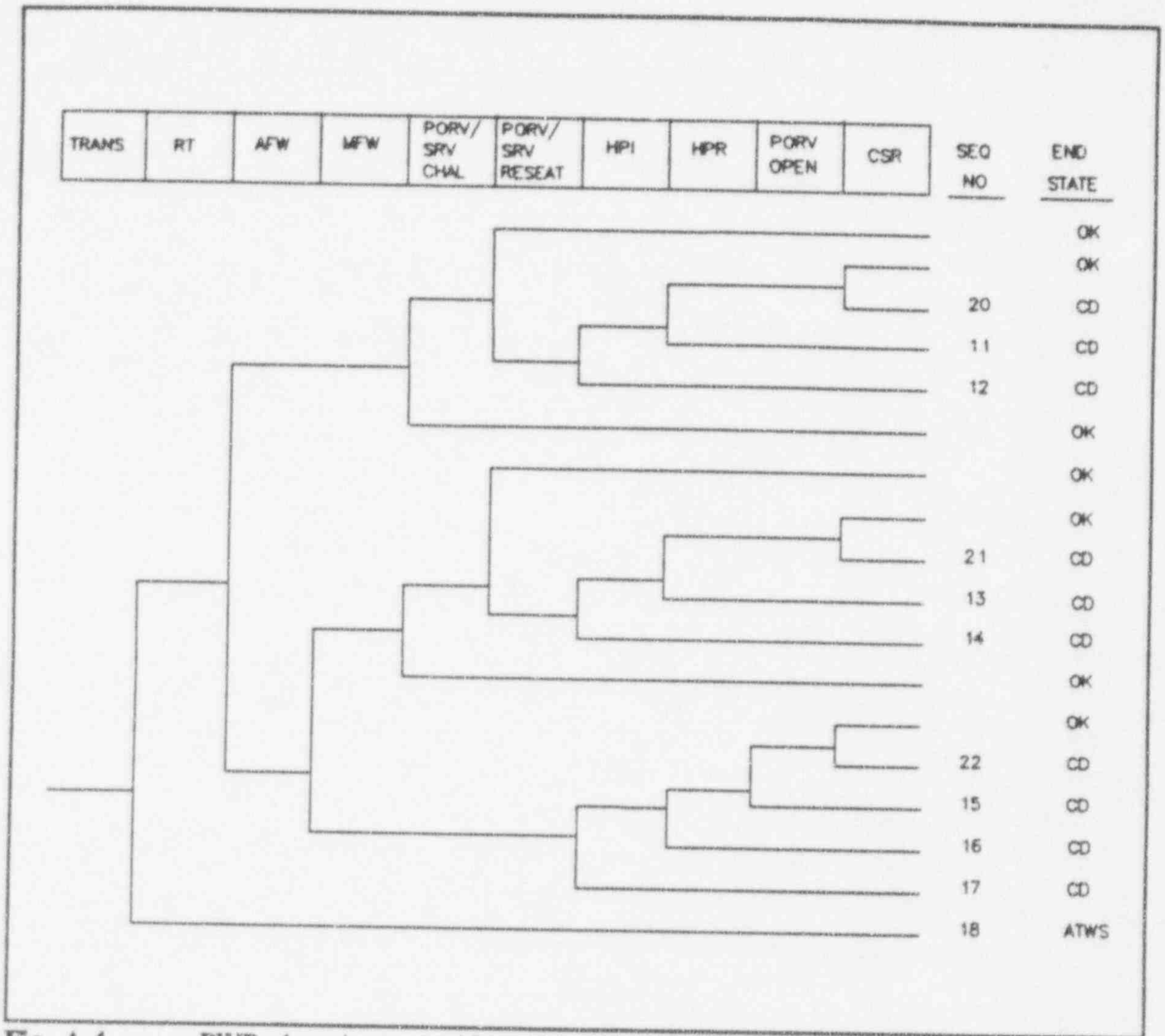

Fig. A.4. PWR class A nonspecific reactor trip

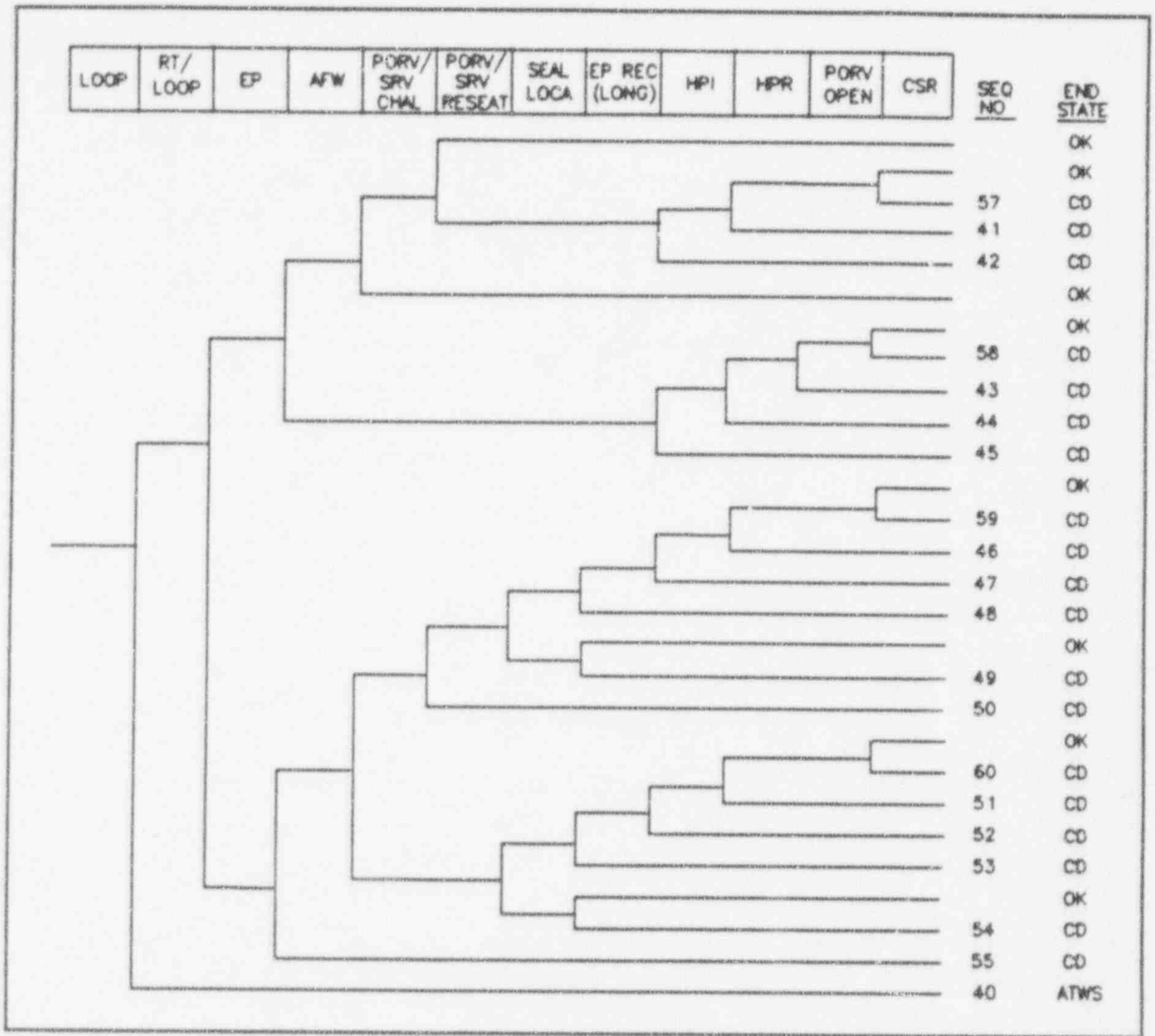

Fig. A.5. PWR class A loss of offsite power

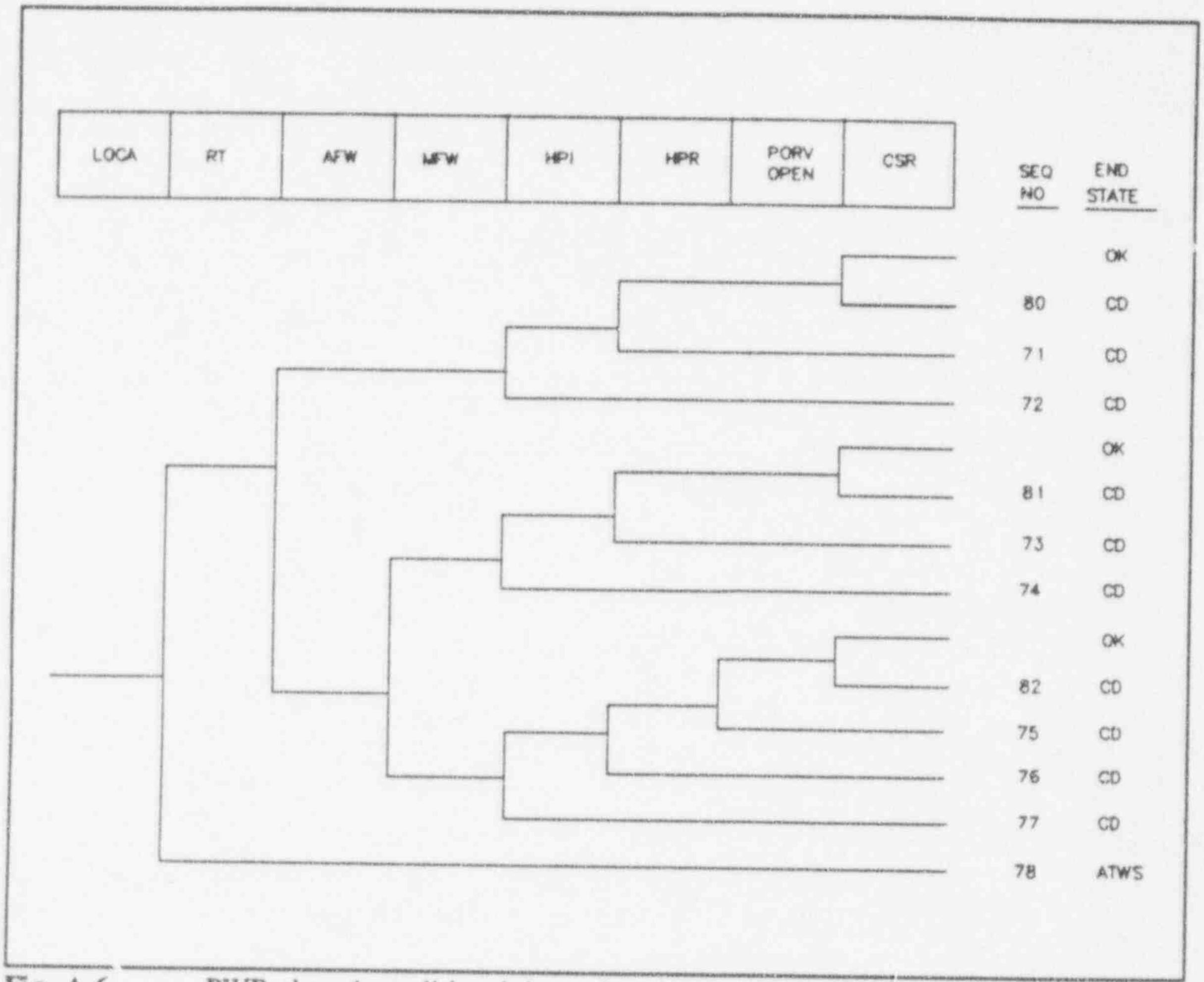

Fig. A.6. PWR class A small-break loss-of-coolant accident

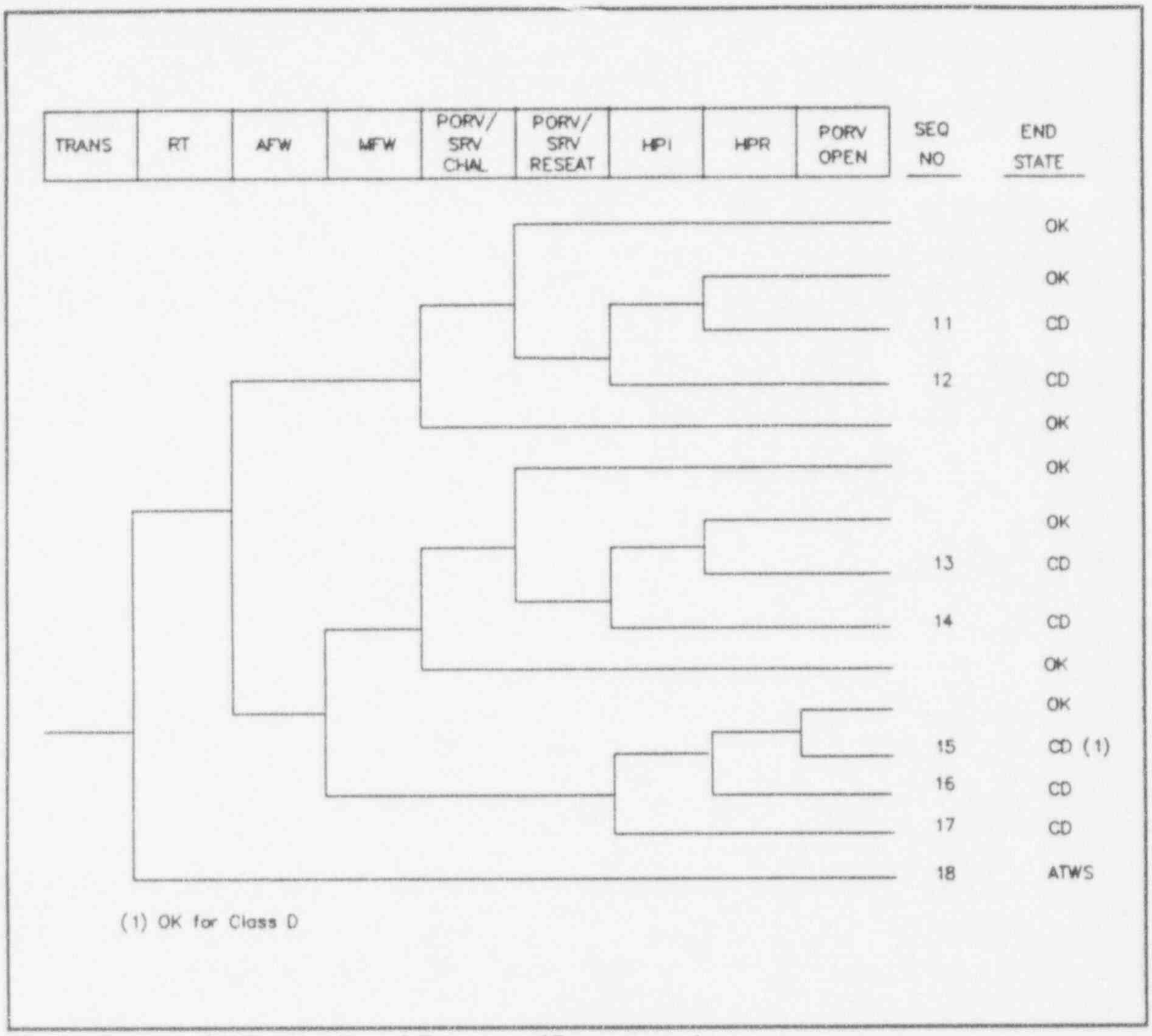

Fig. A.7. PWR class B and D nonspecific reactor trip

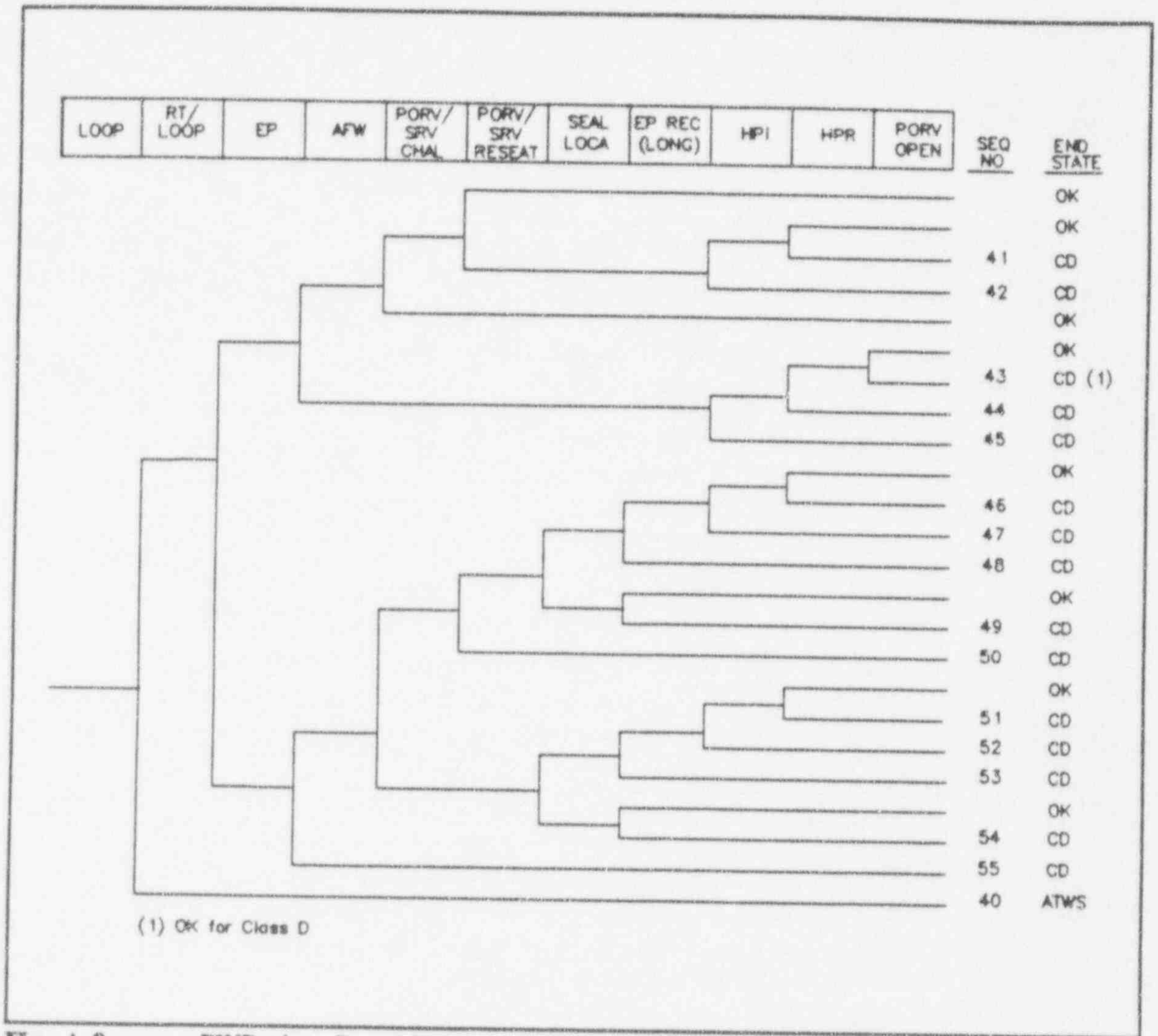

Fig. A.8. PWR class B and D loss of offsite power

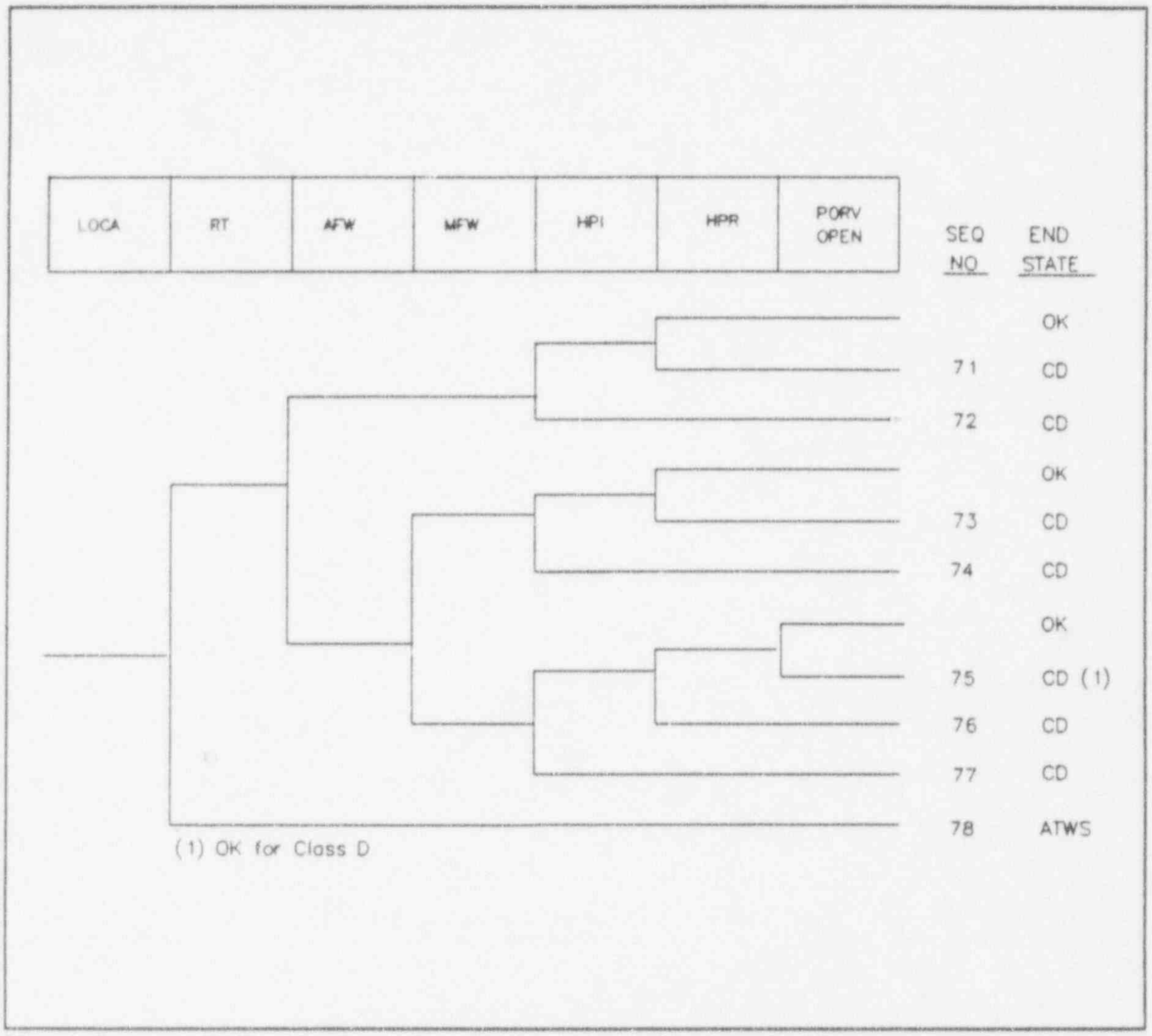

Fig. A.9. PWR class B and D small-break loss-of-coolant accident

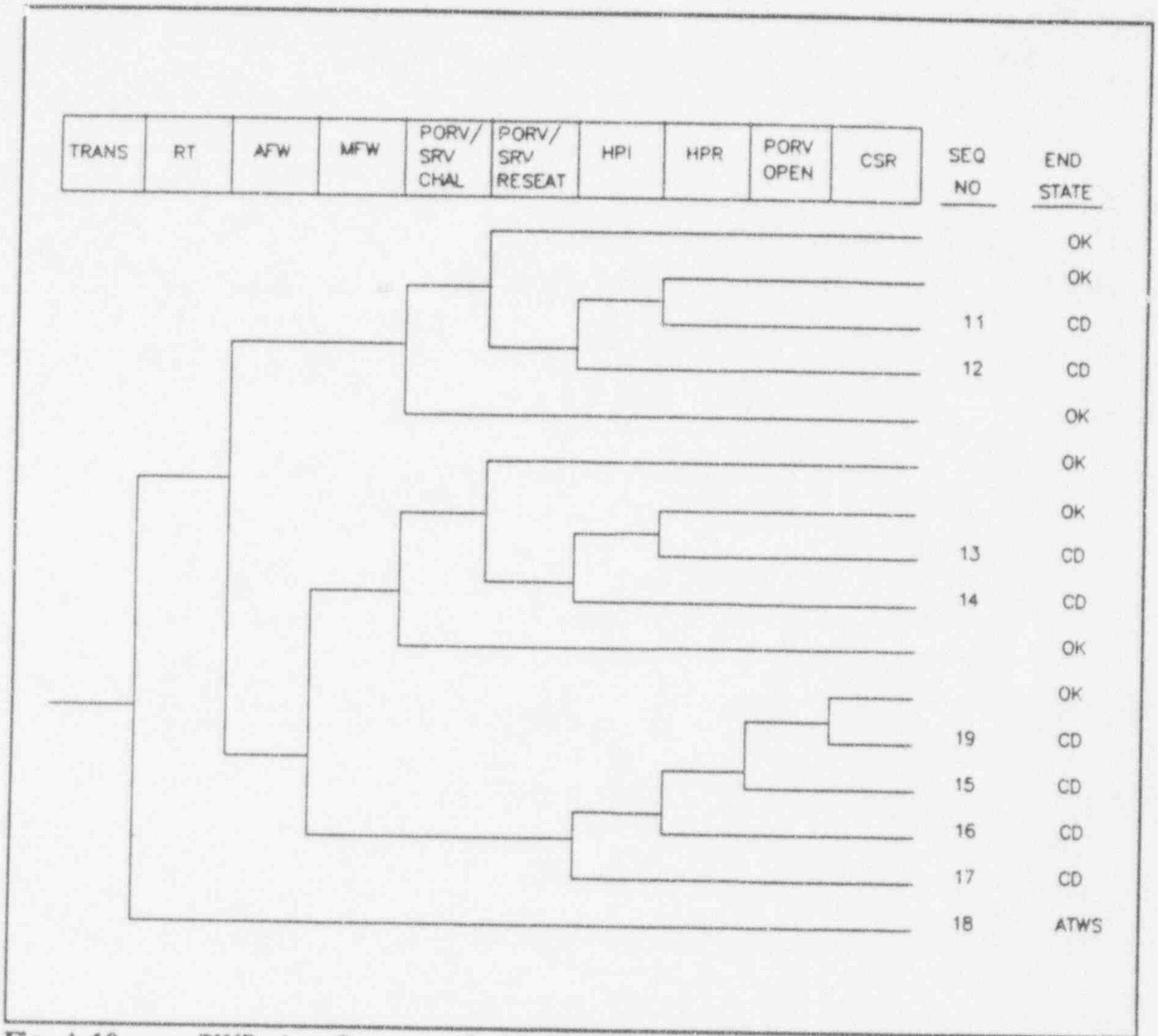

Fig. A.10. PWR class G nonspecific reactor trip

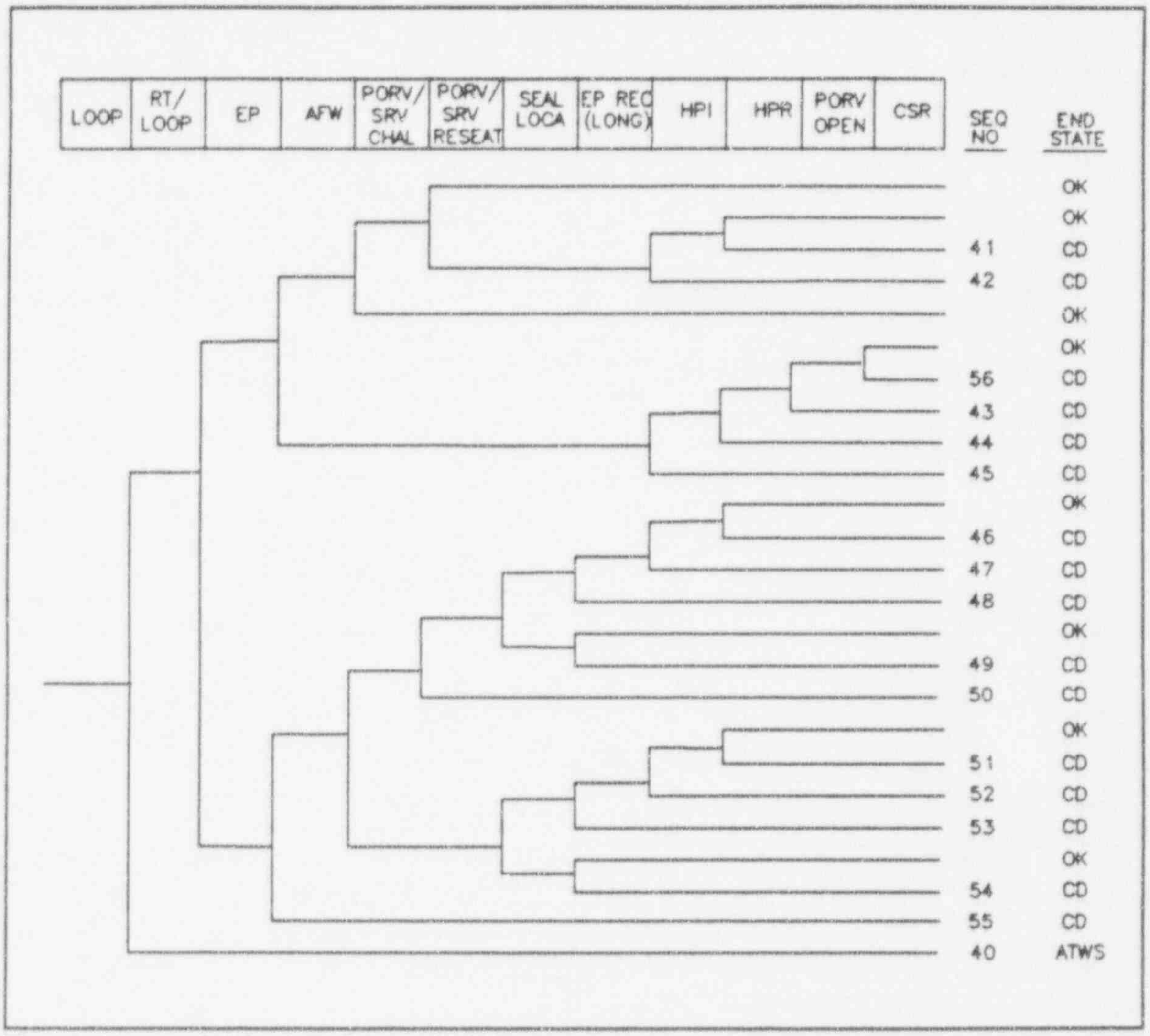

Fig. A.11. PWR class G loss of offsite power

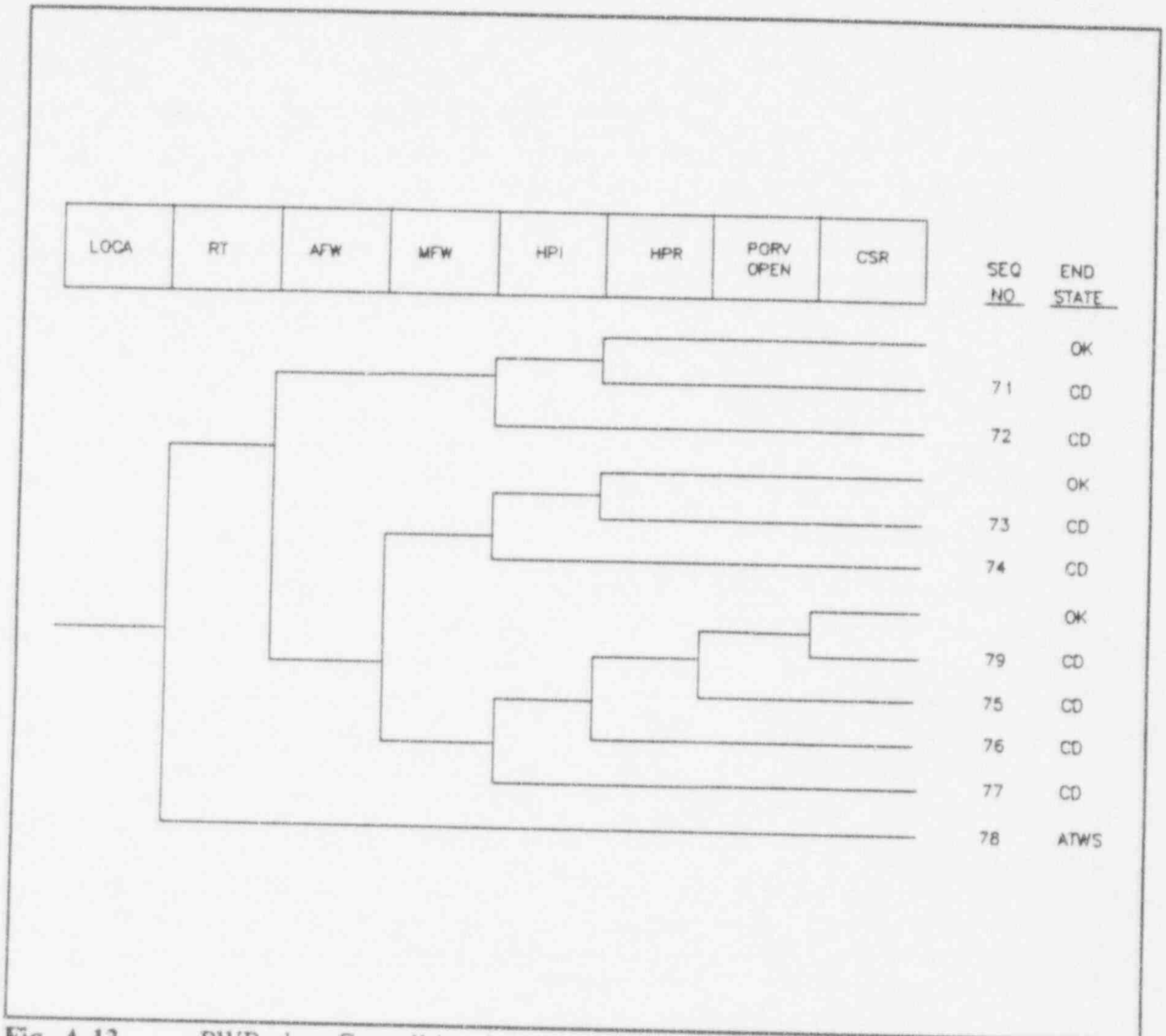

Fig. A.12. PWR class G small-break loss-of-coolant accident

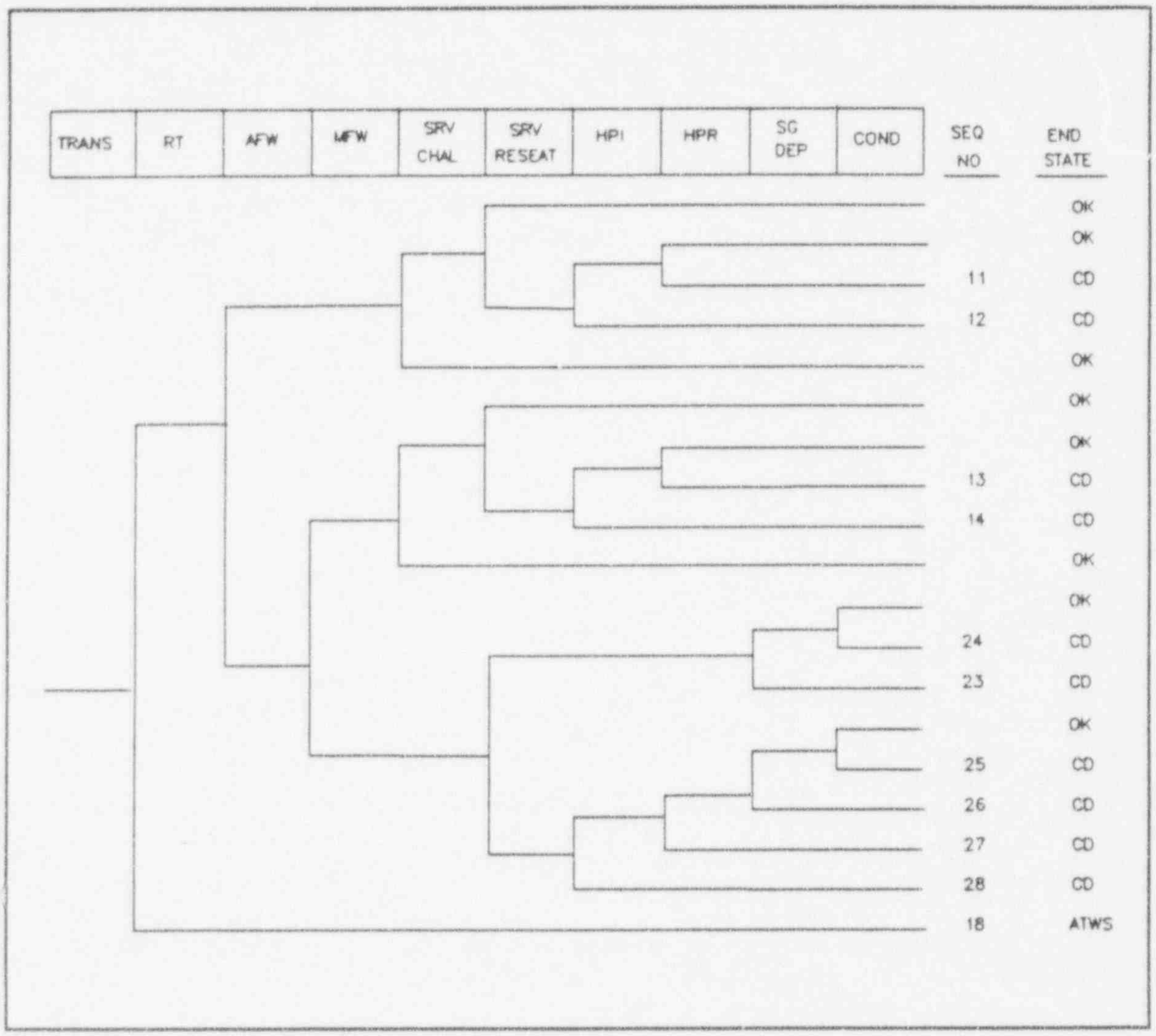

Fig. A.13. PWR class H nonspecific reactor trip

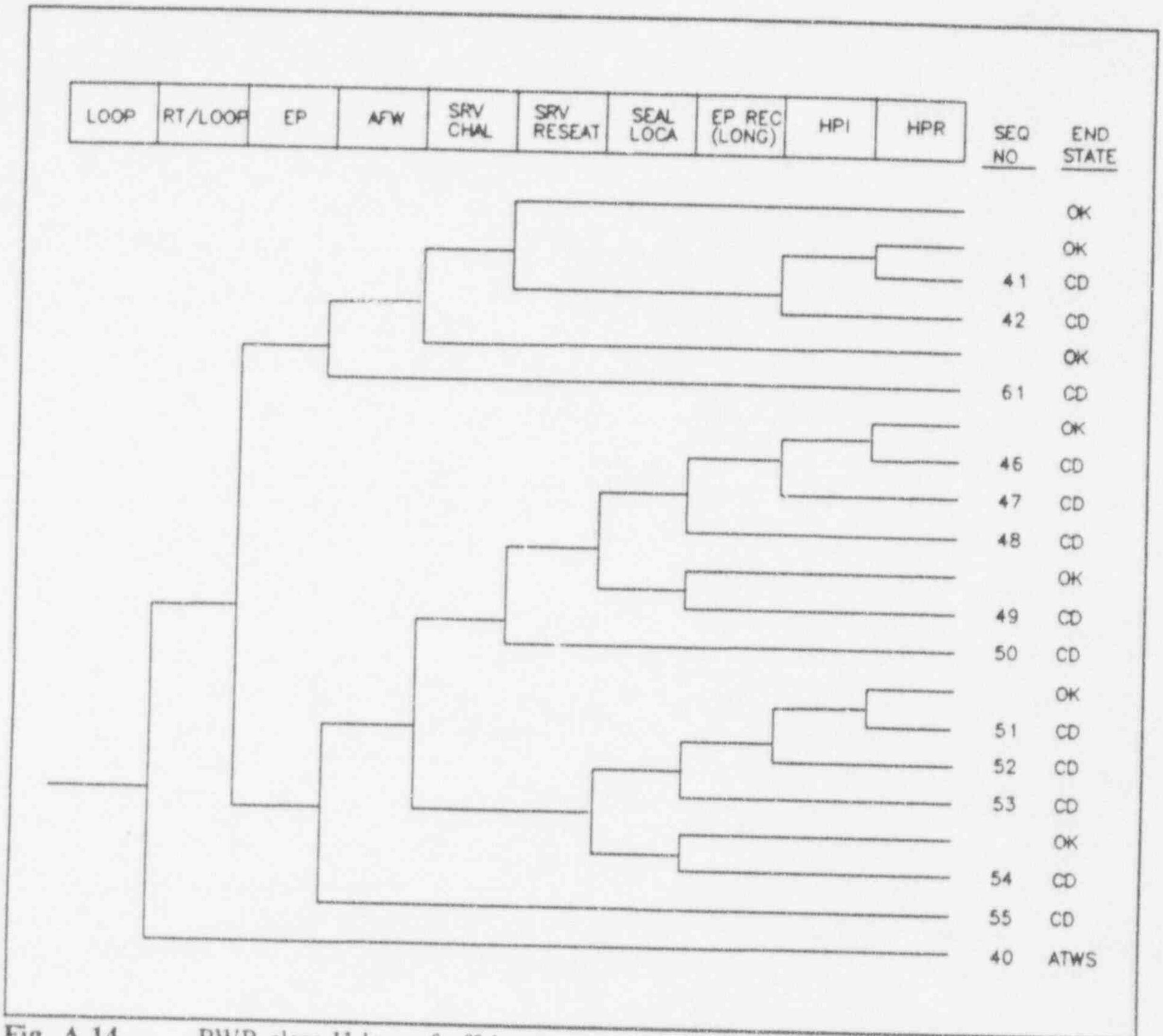

Fig. A.14. PWR class H loss of offsite power

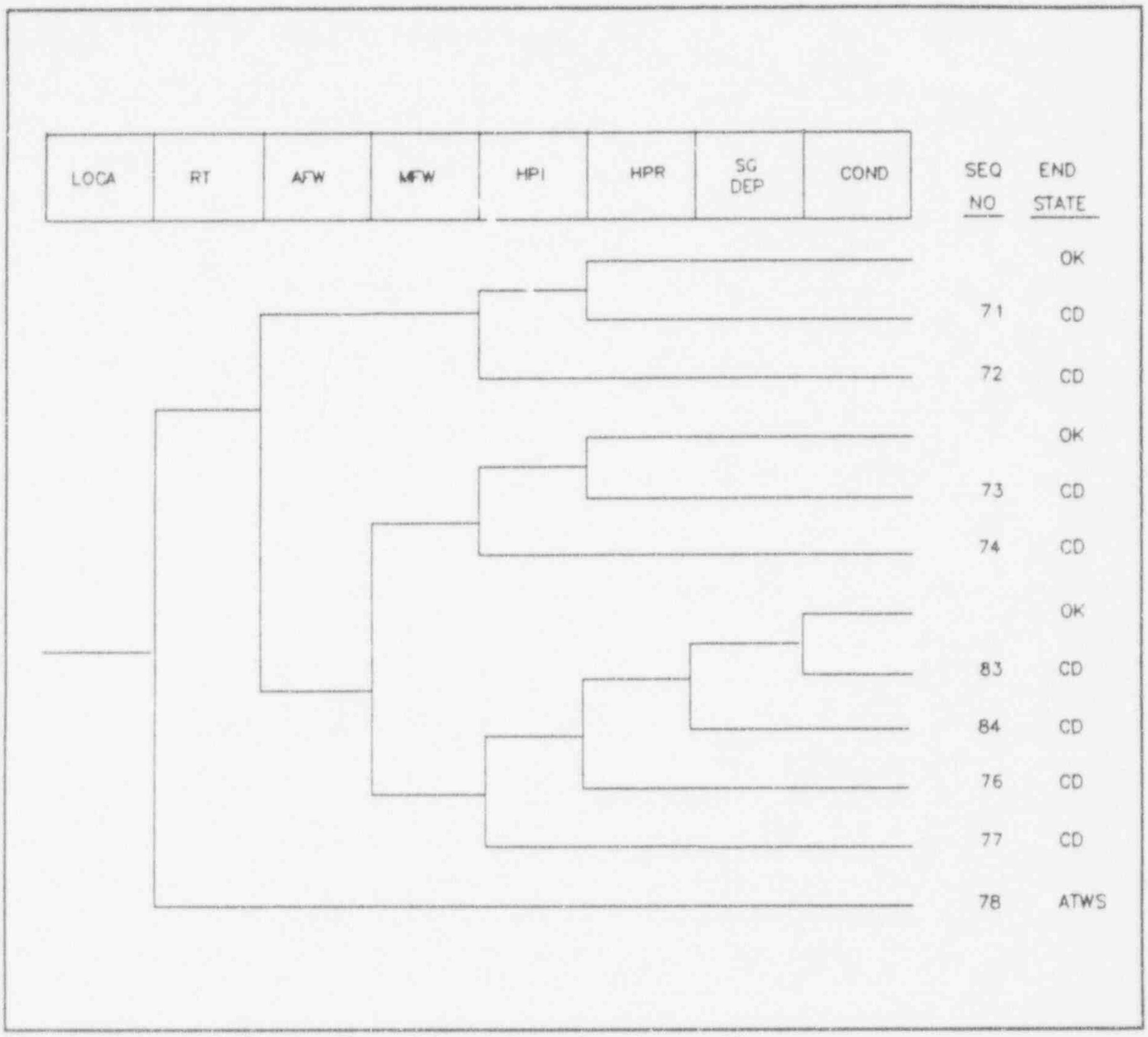

Fig. A.15. PWR class H small-break loss-of-coolant accident

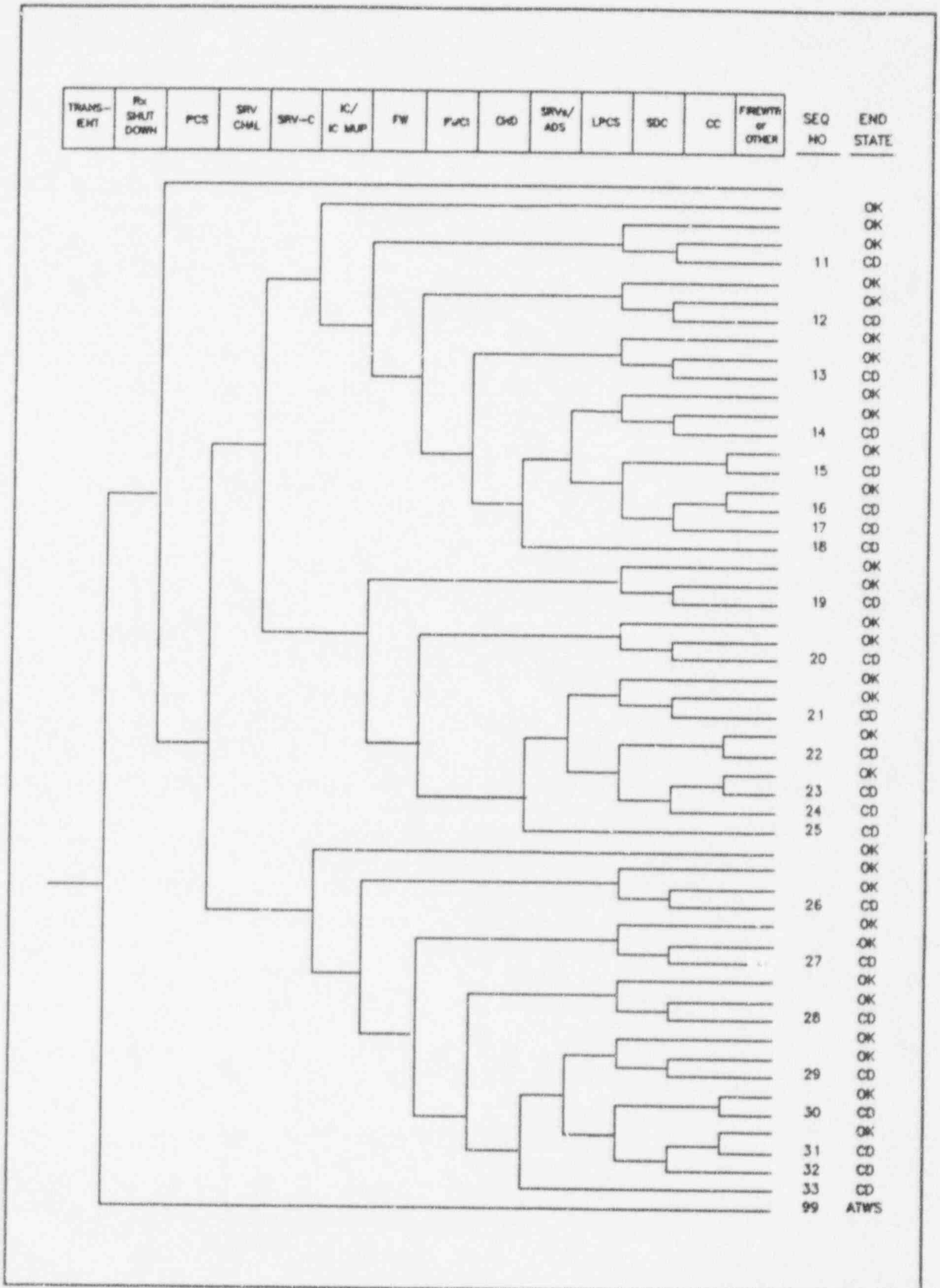

Fig. A.16. BWR class A nonspecific reactor trip

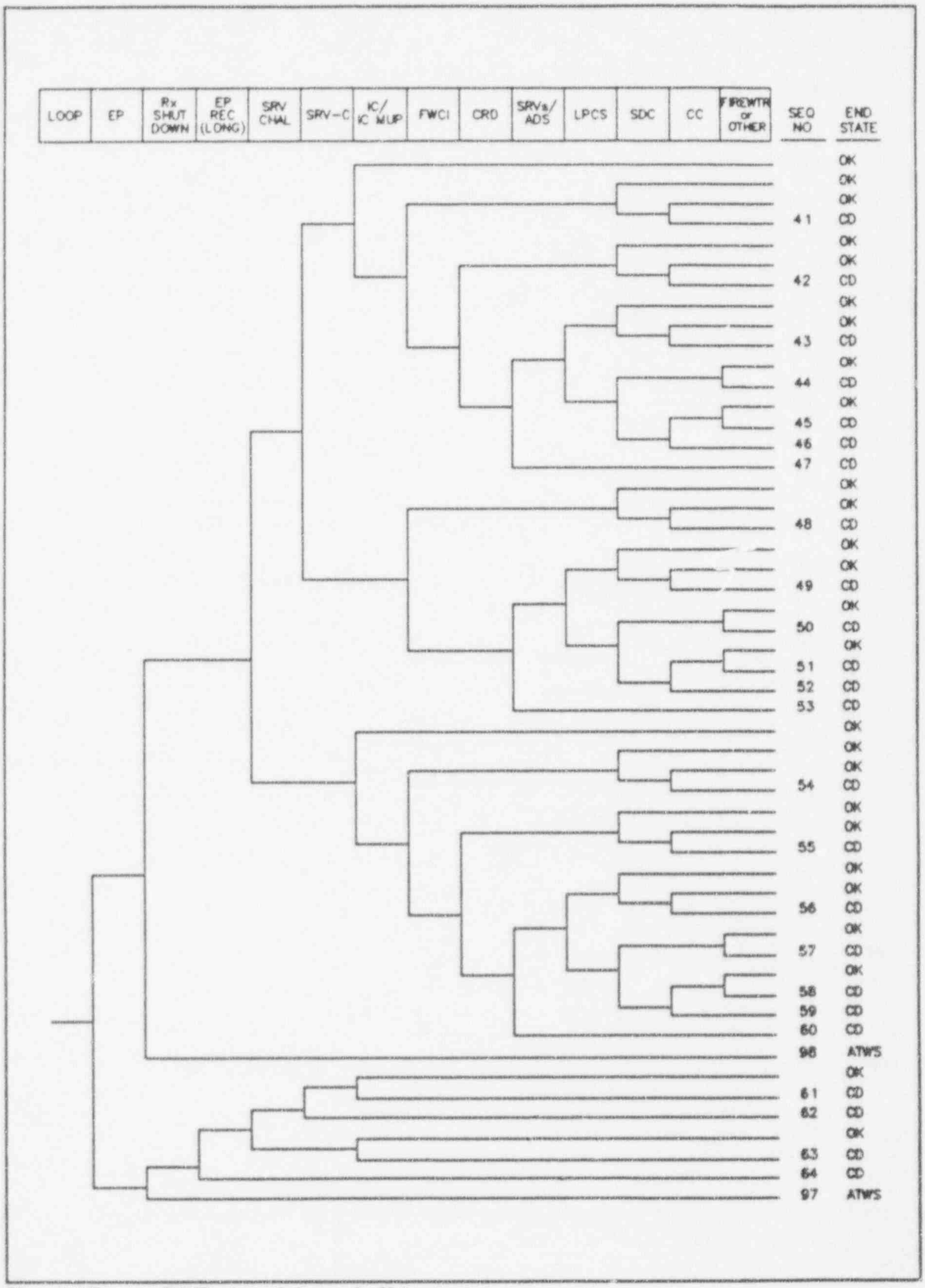

Fig. A.17. BWR class A loss of offsite power

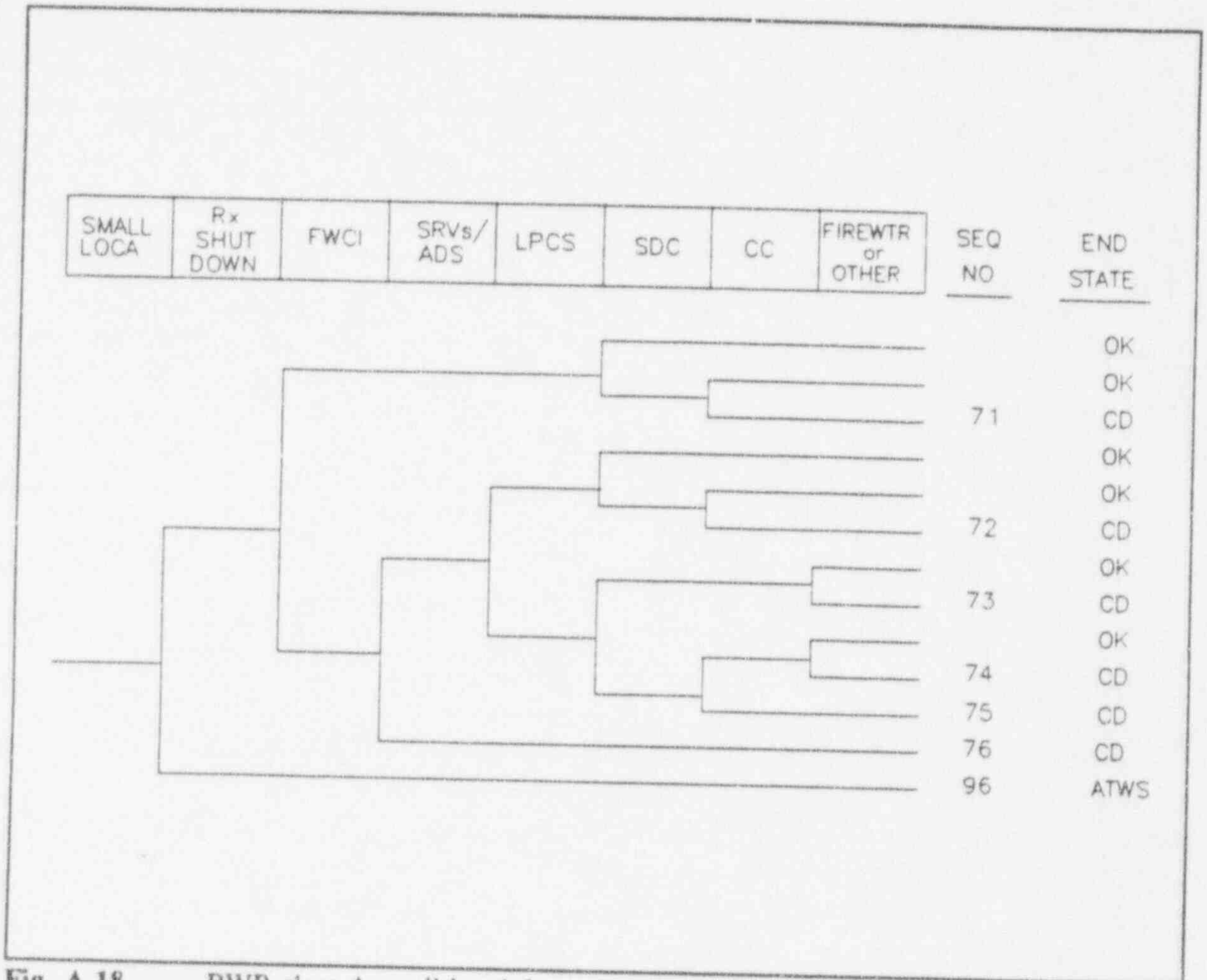

Fig. A.18. BWR class A small-break loss-of-coolant accident

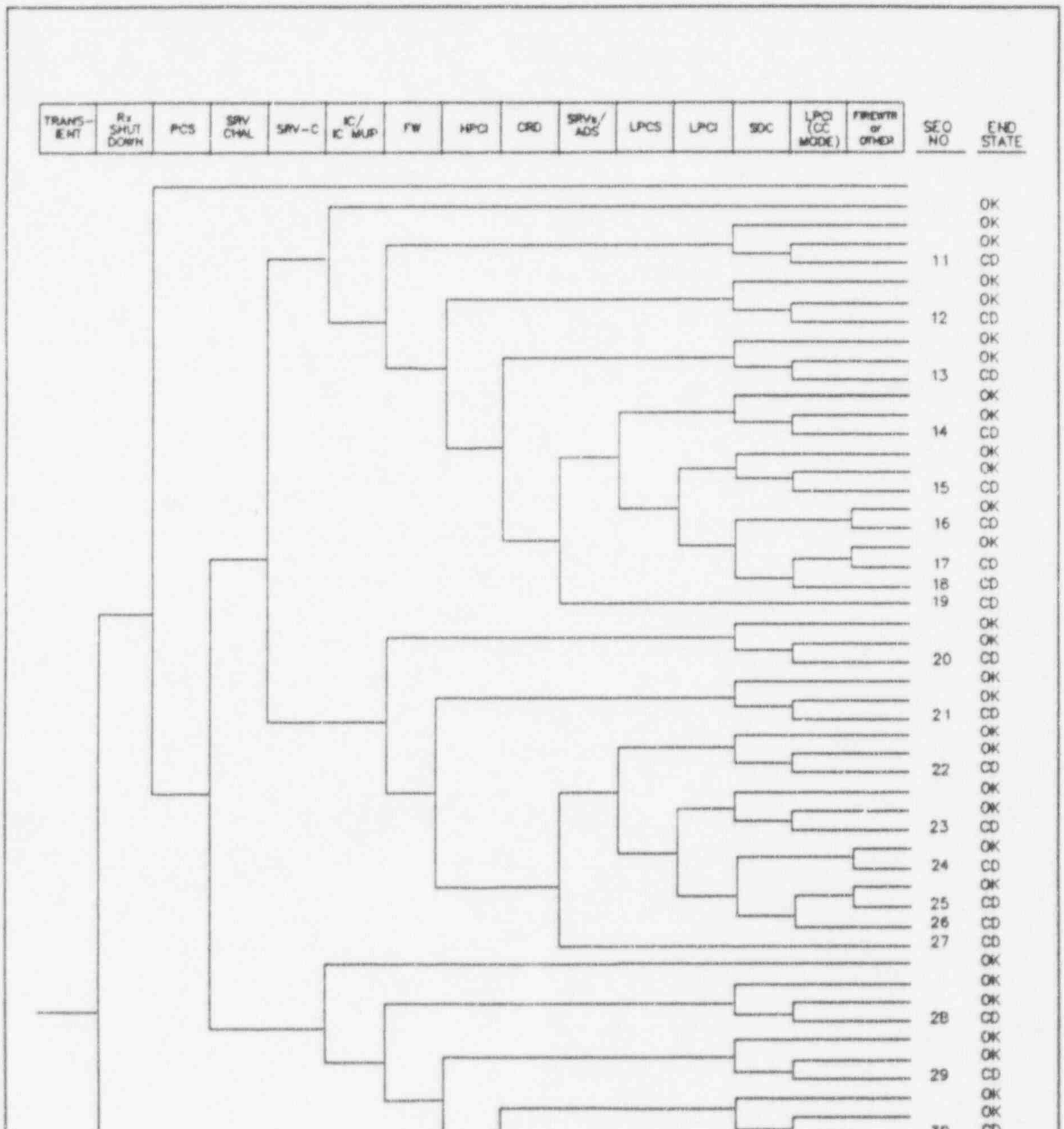

Fig. A.19. BWR class B nonspecific reactor trip

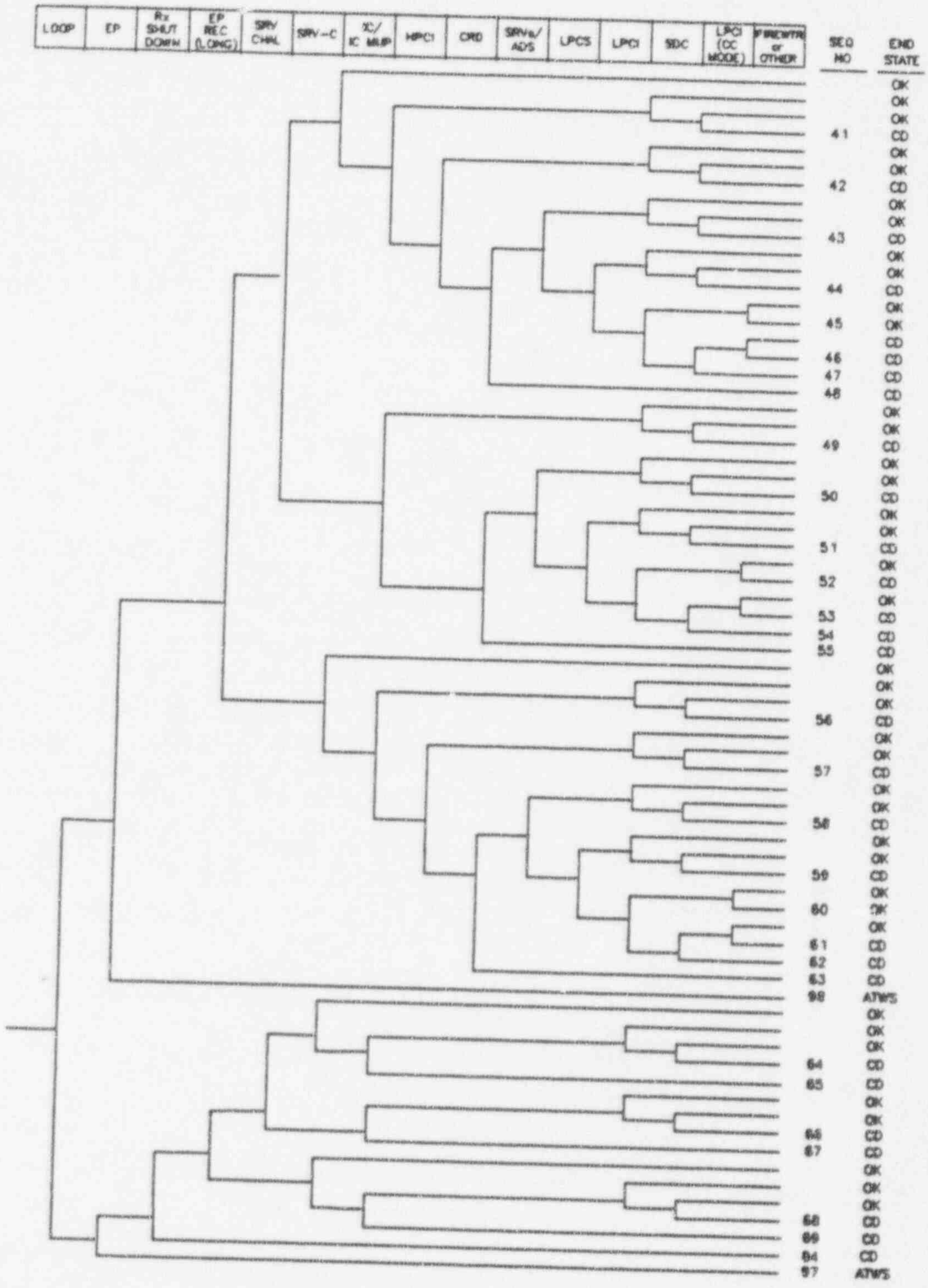

Fig. A.20. BWR class B loss of offsite power

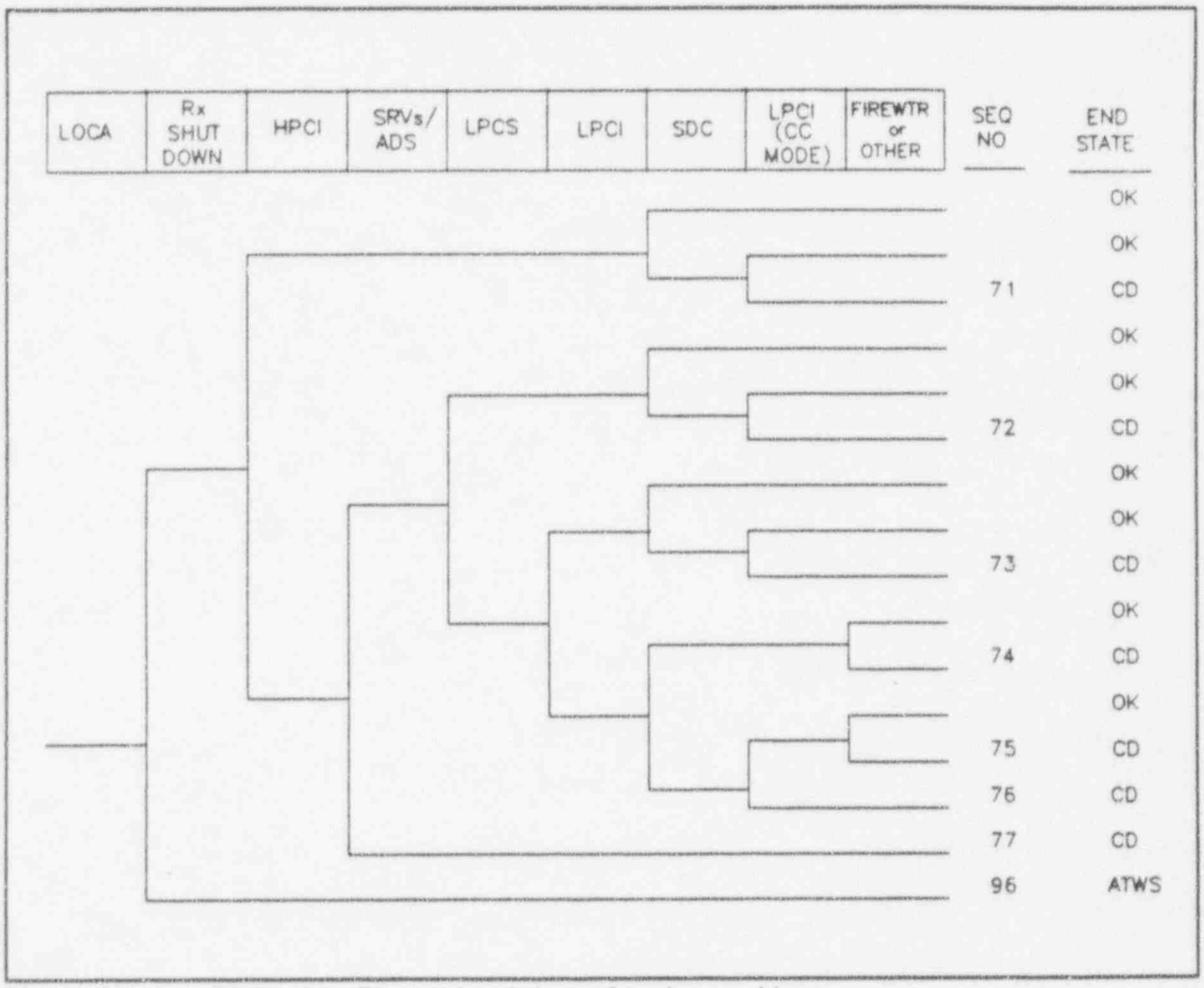

Fig. A.21. BWR class B small-break loss-of-coolant accident

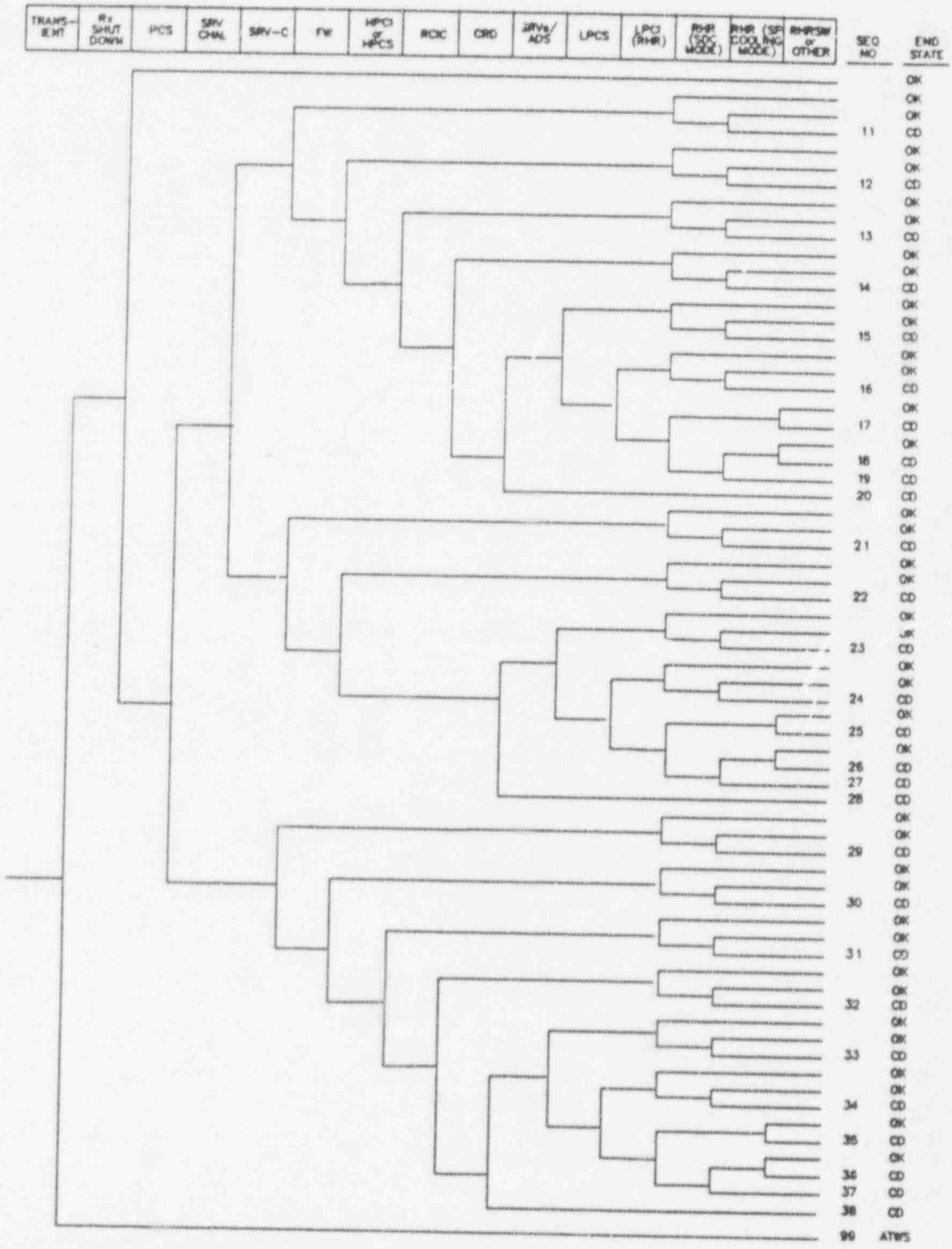

Fig. A.22. BWR class C nonspecific reactor trip

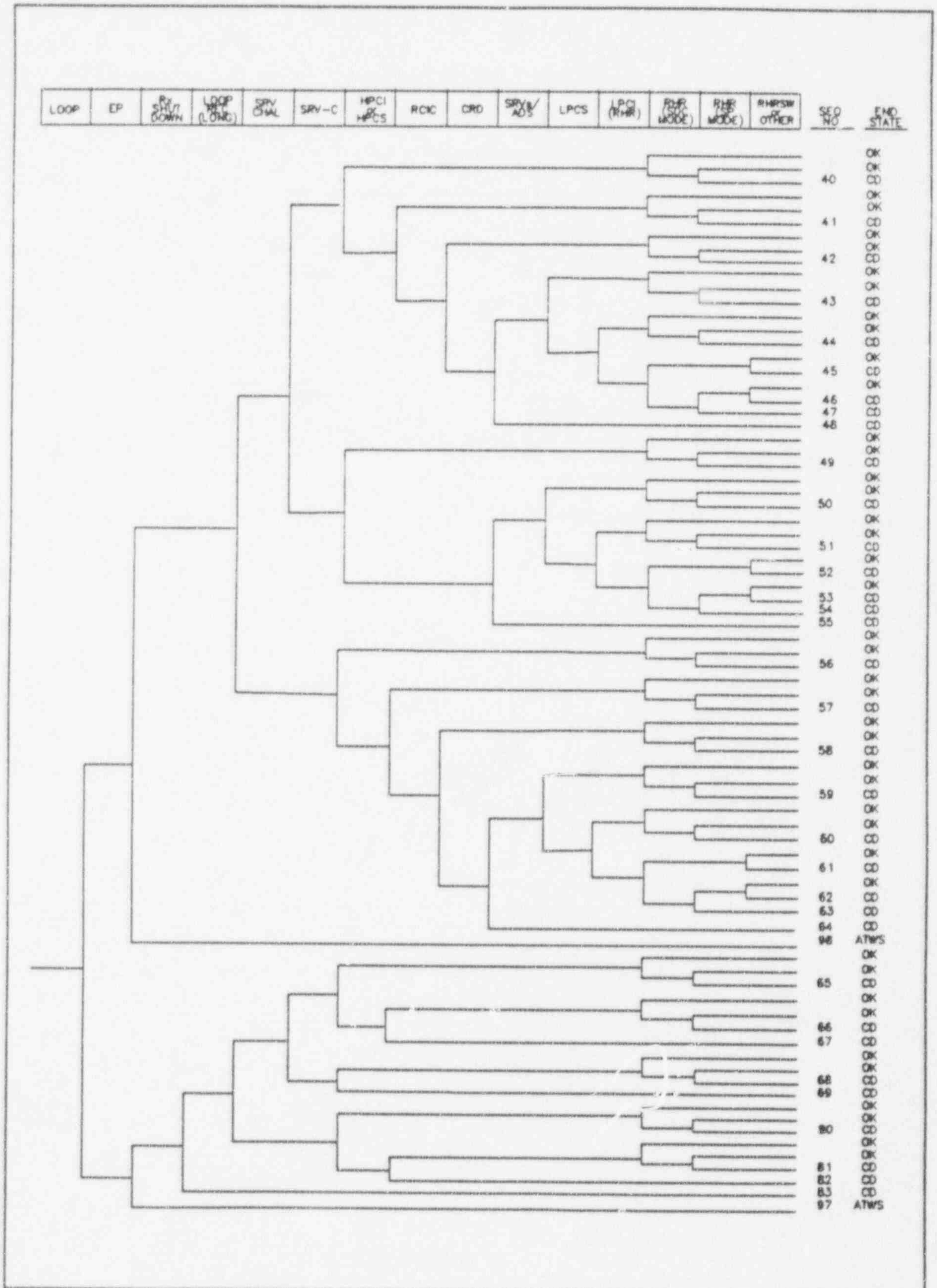

Fig. A.23. BWR class C loss of offsite power

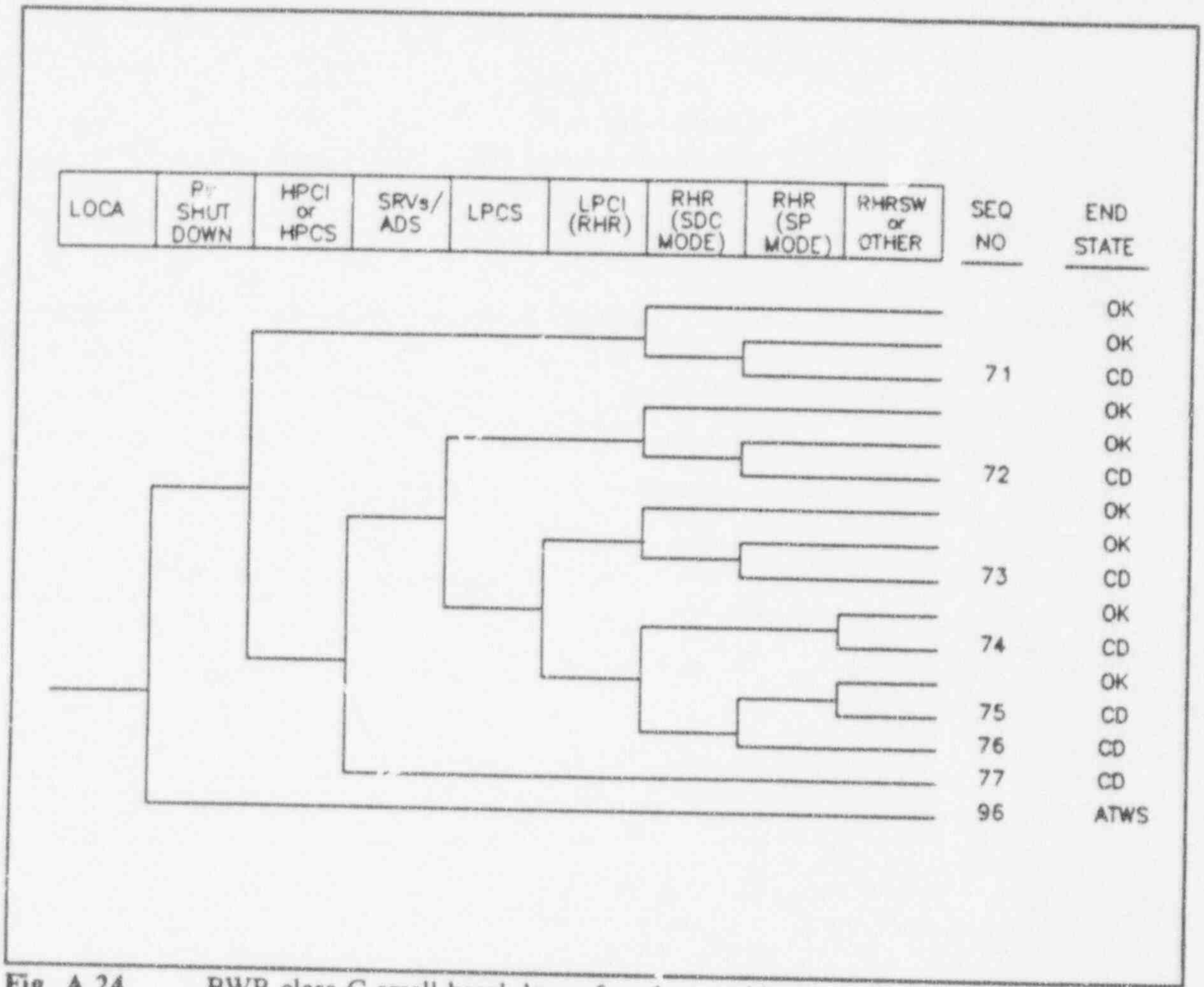

Fig. A.24. BWR class C small-break loss-of-coolant accident