50-244

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

May 7, 1996

50-244

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Company
89 East Avenue
Rochester, NY 14649

SUBJECT: DRAFT 1982-83 PRECURSOR REPORT

Dear Dr. Mecredy:

Enclosed for your information are excerpts from the draft Accident Sequence Precursor (ASP) Report for 1982-83. This report documents the ASP Program analyses of operational events which occurred during the period 1982-83. We are providing the appropriate sections of this draft report to each licensee with a plant which had an event in 1982 or 1983 that has been identified as a precursor. At least one of these precursors occurred at R. E. Ginna Nuclear Power Plant. For background information, we have also enclosed copies of Sections A.1 and A.2 of Appendix A from the 1982-83 ASP Report. Section 2.0 discusses the ASP Program event selection criteria and the precursor quantification process; Sections A.1 and A.2 contain an overview of the models that were used in these analyses. Further detail about these models may be found in various published volumes of NUREG/CR-4674, most recently in Volume 17, "Precursors to Potential Severe Core Damage Accidents: 1992, A Status Report." We emphasize that you are under no licensing obligation to review and comment on the enclosures.

The analyses documented in the draft ASP Report for 1982-83 were performed primarily for historical purposes to obtain the 2 years of precursor data for the NRC's ASP Program which had previously been missing. We realize that any review of the precursor analyses of 1982-83 events by affected licensees would necessarily be limited in scope due to: (1) the extent of the licensee's corporate memory about specific details of an event which occurred 13-14 years ago, (2) the desire to avoid competition for internal licensee staff resources with other, higher priority work, and (3) extensive changes in plant design, procedures, or operating practices implemented since the time period 1982-83, which may have resulted in significant reductions in the probability of (or, in some cases, even precluded) the occurrence of events such as those documented in this report.

The draft report contains detailed documentation for all precursors with conditional core damage probabilities $\geq 1.0 \times 10^{-5}$. However, the relatively large number of precursors identified for the period 1982-83 necessitated that only summaries be provided for precursors with conditional core damage probabilities between 1.0×10^{-6} and 1.0×10^{-5}.

We will begin revising the report about May 31, 1996, to put it in final form for publication. We will respond to any comments on the precursor analyses which we receive from licensees. The responses will be placed in a separate section of the final report. Rochester Gas and Electric Corporation is on

distribution for the final report. Please contact me at 301-415-1441 if you have any questions regarding this letter. Any response to this letter on your part is entirely voluntary and does not constitute a licensing requirement.

Sincerely,

Guy S. Vissing

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Section B.1, "Precursor Analysis of 1/25/82 Steam Generator Tube Rupture with One PORV Failed Open"
2. Section 2 "Selection Criterial and Quantification"
3. Appendix A, "ASP Models"

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ORIGINAL SIGNED BY:

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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B.1 LER No. 244/82-003 and -005

Event Description:	Steam Generator Tube Rupture with one PORV failed open
Date of Event:	January 25, 1982
Plant:	Ginna

B.1.1 Summary

On January 25, 1982, while operating at 100% power, the Ginna B steam generator experienced a tube rupture. The resulting plant transient included significant primary system depressurization, actuation of the safety injection system and minor releases of radioactive materials from the plant. During the transient, a pressurizer PORV failed to close after being used to reduce primary and secondary pressure below the steam safety valve setting. The estimated conditional core damage probability for this event is 3.0×10^{-4}.

B.1.2 Event Description

On January 25, 1982, at 0925, while the plant was operating at 100% power, a tube rupture occurred in steam generator B.. Multiple control room alarms alerted the operators to a Reactor Coolant System (RCS) rapid depressurization. The air ejector radiation monitor alarm indicated that the rupture was likely in steam generator B. The continuing pressure drop resulted in an automatic reactor trip and an automatic safety injection actuation. All three HPI pumps started. The safety injection actuation resulted in an automatic containment isolation and trip of the operating charging pumps. All safety systems functioned properly. Both reactor coolant pumps were manually stopped, and natural circulation cooling in the RCS was verified. The pressurizer emptied, and the RCS depressurization reached a minimum of 1200 psig. A small steam bubble formed during natural circulation in the upper head but was collapsed when safety injection flow refilled the RCS.

Initially, operators cooled down the reactor by sending steam from both steam generators to the main condensor. The B steam generator was isolated at 0940, and natural circulation cooling in loop B was terminated. The B steam generator water level continued to rise in spite of the termination of feedwater flow to the steam generator due to flow through the ruptured tube. At 0955, the narrow-range water level indicator on B steam generator went off scale high and subsequently the B main steam line started to fill.

At 0957, the safety injection actuation circuitry was reset thus resetting the containment isolation system. Instrument air and thus control of the air-operated valves inside containment were restored.

At 1007, operators attempted to equalize the pressure differential between the RCS and the secondary side of the B steam generator to stop flow through the tube rupture. A pressurizer PORV was cycled three times before it stuck open. The operator attempted to close the valve, but the valve would not close. The operator then closed the block valve to prevent further RCS water loss. Steam bubbles in the reactor vessel upper head and in the top of the B steam generator tubes occurred as well. The growth of the bubbles and increased safety injection flow resulted in the rapid filling of the pressurizer. Loop A natural circulation was not affected by the steam bubbles.

One of the B steam generator safety valves cycled three times as a result of the over-pressurization caused by continued flow through the ruptured tube. At 1038, safety injection was terminated to prevent further water discharge through the safety valve. At 1040, the condensate system was secured to prevent further radioactive contamination of the condensate storage tanks and demineralizers. The operators used the SG PORV to relieve steam from the A steam generator.

At 1042, the pressurizer heaters were reenergized (after having tripped at 0928 from low pressurizer level) to re-establish a steam bubble in the pressurizer. At 1052, the rupture disk on the pressurizer relief tank burst due to the addition of water from the letdown line relief valve, the pressurizer PORV, and the relief valve for the RCP seal return line.

At 1107, one safety injection pump was started in anticipation of an RCS pressure drop due to the restart of the A RCP. At 1119, the B steam generator safety valve lifted and closed. At this time, the B steam line had flooded sufficiently to cause water rather than steam to be released. At 1121, the A RCP was started. The RCP flow cooled and collapsed any remaining steam bubbles in the reactor upper vessel head and the B steam generator. This addition of flow lead to another cycle of the B steam generator safety valve. Safety injection was stopped, but the valve continued to leak water at approximately 100 gpm.

At 1152, the pressurizer level returned on scale and a steam bubble was re-established. At 1202, normal letdown from the RCS to the chemical and volume control system was re-established. Due to the B steam generator safety valve leak, the RCS continued to leak through the tube rupture in steam generator B. Operators re-started one safety injection pump at 1212 in response to the continued decrease in pressurizer level. The pump was intermittently operated until 1235. The safety relief valve on steam generator B stopped leaking at approximately 1225.

At 1227, the RCS and B steam generator pressures equalized. RCS pressure was maintained at 25 psia below steam generator B pressure. At 1840, B steam generator water level returned on scale. Feed and bleed was then used to cool steam generator B.

At 0700, on January 26, 1982, the residual heat removal system was placed in operation, and the plant was declared to be in cold shutdown.

B.1.3 Additional Event-Related Information

The ruptured B steam generator tube was located at row 42, column 55 on the hot-leg side of the steam generator. The rupture was approximately 4 inches long and 0.7 inches wide at its center. The rupture was fish-mouth shaped and pointed outward along the tube column. The tube appeared ballooned at the rupture location and had a wall thickness of less than 5% of the nominal thickness. Markings on the exterior of the tube had the appearance of fretting wear. Damage to sixteen additional tubes that had been plugged in steam generator B was identified. Foreign objects and tube fragments were found in the steam generator. An examination of steam generator A revealed the existence of some small foreign objects as well. The most probable cause of damage was due to a piece of metal that was left in the steam generator during a 1975 repair when a large ring was removed from the steam generator to increase the efficiency of the recirculation flow. The ring was cut into pieces to be removed, but one piece was left inside the steam generator.

Additional information on this event is included in the *Report to Congress on Abnormal Occurrences, January - March 1982*, NUREG-0090, Vol. 5, No. 1.

B.1.4 Modeling Assumptions

This event was modeled as a steam generator tube rupture initiating event. Since a second pressurizer PORV was available and other approaches could be used for depressurization during this event, the model was not revised to reflect the stuck open pressurizer PORV. Since the flow which continued to leak from the steam generator safety relief valve would not have depleted the RWST during the mission time and the valve eventually shut, the model was not revised to reflect the stuck-open, leaking steam generator safety relief valve. However, a sensitivity study was performed assuming that the steam generator tube rupture occurred, the safety relief valve stuck open and the steam generator could not be isolated (SG.ISO.AND.RCS.COOLDOWN set to failed and non-recoverable).

B.1.5 Analysis Results

The estimated conditional core damage probability for this event is 3.0×10^{-4}. The dominant sequence highlighted on the event tree in Figure B.1.1 (to be provided in the final report) involved the successful operation of auxiliary feedwater and the failure of high pressure injection. The estimated conditional core damage probability for the sensitivity case is 8.8×10^{-3}. The dominant sequence involved succussful operation of AFW and HPI, the failure to isolate the steam generator, and the failure to cool RCS below RHR initiation pressure.

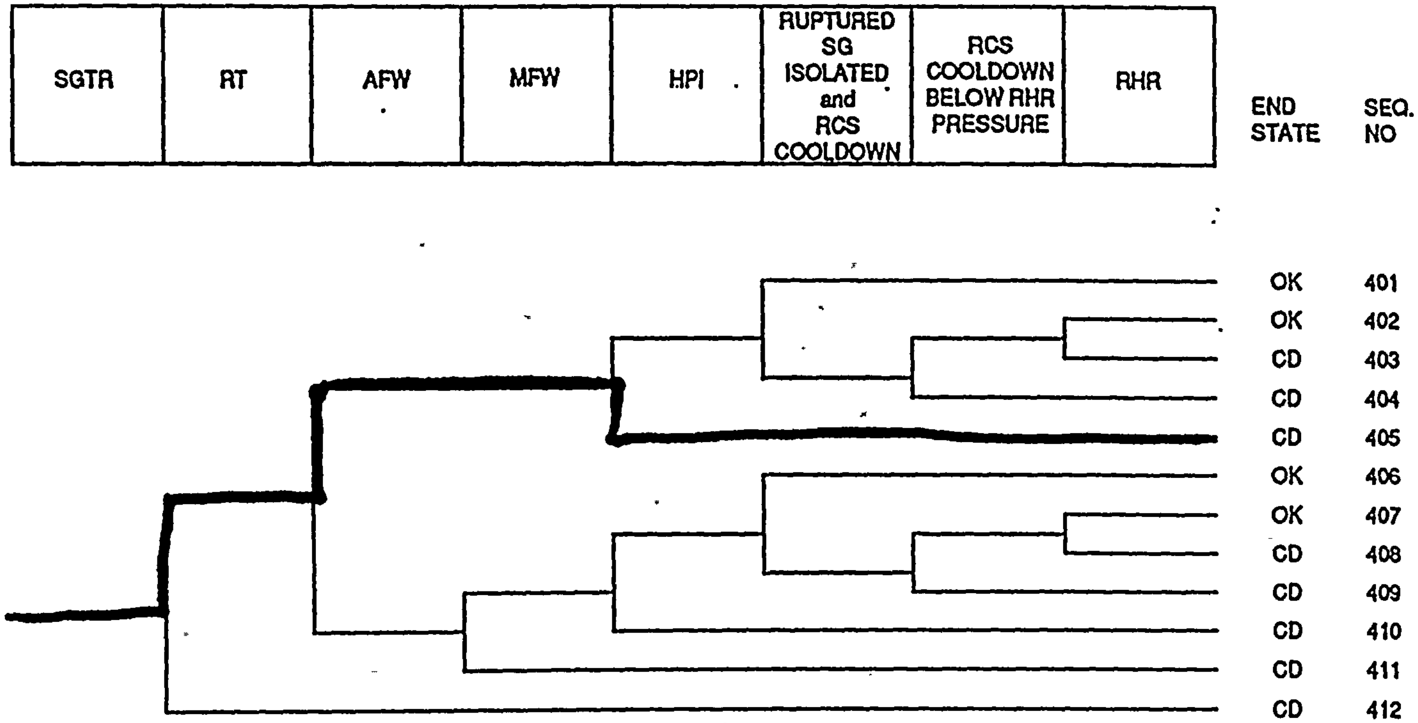
SGTR
RT
AFW
MFW
HPI
RUPTURED SG ISOLATED and RCS COOLDOWN
RCS COOLDOWN BELOW RHR PRESSURE
RHR
END STATE
SEQ. NO
OK 401
OK 402
CD 403
CD 404
CD 405
OK 406
OK 407
CD 408
CD 409
CD 410
CD 411
CD 412

Fig B.1.1

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 244/82-003 and -005
Event Description: Steam Generator Tube Rupture
Event Date: January 25, 1982
Plant: Ginna

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SGTR	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
SGTR	3.0E-04
Total	3.0E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01
412	sgtr rt	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01
412	sgtr rt	CD	2.8E-05	1.0E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\ginna.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.6E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	
rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	1.6E-05	4.5E-01	
afw/atws	4.3E-03	1.0E+00	

afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	6.0E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	8.2E-03	1.0E+00	
sg.iso.and.rcs.cooldown	1.0E-02	1.0E-01	
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

* branch model file
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11-06-1995
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CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 244/82-003 and -005
Event Description: Steam Generator Tube Rupture with SG relief valve stuck open
Event Date: January 25, 1982
Plant: Ginna

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SGTR	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
SGTR	8.8E-03
Total	8.8E-03

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
404	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN rcs.cool.below.rhr	CD	6.0E-03	1.0E+00
403	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN -rcs.cool.below.rhr rhr	CD	2.5E-03	7.0E-02
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
403	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN -rcs.cool.below.rhr rhr	CD	2.5E-03	7.0E-02
404	sgtr -rt -afw -hpi SG.ISO.AND.RCS.COOLDOWN rcs.cool.below.rhr	CD	6.0E-03	1.0E+00
405	sgtr -rt -afw hpi	CD	2.7E-04	8.9E-01

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\pwrb8283.cmp
BRANCH MODEL: c:\aspcode\models\ginna.82
PROBABILITY FILE: c:\aspcode\models\pwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	2.6E-04	1.0E+00	
loop	1.6E-05	3.6E-01	
loca	2.4E-06	5.4E-01	
sgtr	1.6E-06	1.0E+00	

rt	2.8E-04	1.0E-01	
rt(loop)	0.0E+00	1.0E+00	
afw	1.6E-05	4.5E-01	
afw/atws	4.3E-03	1.0E+00	
afw/ep	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	1.0E-03
porv.chall	4.0E-02	1.0E+00	
porv.chall/afw	1.0E+00	1.0E+00	
porv.chall/loop	1.0E-01	1.0E+00	
porv.chall/sbo	1.0E+00	1.0E+00	
porv.reseat	2.0E-02	1.1E-02	
porv.reseat/ep	2.0E-02	1.0E+00	
srv.reseat(atws)	1.0E-01	1.0E+00	
hpi	3.0E-04	8.9E-01	
feed.bleed	2.0E-02	1.0E+00	1.0E-02
emrg.boration	0.0E+00	1.0E+00	1.0E-02
recov.sec.cool	2.0E-01	1.0E+00	
recov.sec.cool/offsite.pwr	3.4E-01	1.0E+00	
rcs.cooldown	3.0E-03	1.0E+00	1.0E-03
rhr	2.2E-02	7.0E-02	1.0E-03
rhr.and.hpr	1.0E-03	1.0E+00	1.0E-03
hpr	4.0E-03	1.0E+00	1.0E-03
ep	2.9E-03	8.9E-01	
seal.loca	2.3E-01	1.0E+00	
offsite.pwr.rec/-ep.and.-afw	2.1E-01	1.0E+00	
offsite.pwr.rec/-ep.and.afw	9.9E-02	1.0E+00	
offsite.pwr.rec/seal.loca	6.0E-01	1.0E+00	
offsite.pwr.rec/-seal.loca	8.2E-03	1.0E+00	
SG.ISO.AND.RCS.COOLDOWN	1.0E-02 > 1.0E+00	1.0E-01 > 1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E-02 > Failed		
rcs.cool.below.rhr	3.0E-03	1.0E+00	3.0E-03
prim.press.limited	8.8E-03	1.0E+00	

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** forced

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2.0 Selection Criteria and Quantification

2.1 Accident Sequence Precursor Selection Criteria

The Accident Sequence Precursor (ASP) Program identifies and documents potentially important operational events that have involved portions of core damage sequences and quantifies the core damage probability associated with those sequences.

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. Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with four initiators: trip [which includes loss of main feedwater (LOFW) within its sequences], loss-of-offsite power (LOOP), small-break loss-of-coolant accident (LOCA), and steam generator tube ruptures (SGTR) (PWRs only). These four initiators are primarily associated with loss of core cooling. ASP Program staff members examine licensee event reports (LERs) and other event documentation to determine the impact that operational events have on potential core damage sequences.

2.1.1 Precursors

This section describes the steps used to identify events for quantification. Figure 2.1 illustrates this process.

A computerized search of the SCSS data base at the Nuclear Operations Analysis Center (NOAC) of the Oak Ridge National Laboratory was conducted to identify LERs that met minimum selection criteria for precursors. This computerized search identified LERs potentially involving failures in plant systems that provide protective functions for the plant and those potentially involving core damage-related initiating events. Based on a review of the 1984–1987 precursor evaluations and all 1990 LERs, this computerized search successfully identifies almost all precursors and the resulting subset is approximately one-third to one-half of the total LERs. It should be noted, however, that the computerized search scheme has not been tested on the LER database for the years prior to 1984. Since the LER reporting requirements for 1982-83 were different than for 1984 and later, the possibility exists that some 1982-83 precursor events were not included in the selected subset. Events described in NUREG -0900[20] and in issues of *Nuclear Safety* that potentially impacted core damage sequences were also selected for review.

Those events selected for review by the computerized search of the SCSS data base underwent at least two independent reviews by different staff members. The independent reviews of each LER were performed to determine if the reported event should be examined in greater 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 process involved eliminating events that satisfied predefined criteria for rejection and accepting all others as either potentially significant and requiring analysis, or potentially significant but impractical to analyze. All events identified as impractical to analyze at any point in the study are documented in Appendix E. Events were also 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—for example, short-term single failures in redundant systems, uncomplicated reactor trips, and LOFW events.

Enclosure 2

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

- a component failure with no loss of redundancy,
- a short-term 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,
- a design error discovered by reanalysis,
- an event 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.

Events identified for further consideration typically included the following:

- unexpected core damage initiators (LOOP, SGTR, and small-break LOCA);
- all events in which a 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 in which 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.

Events determined to be potentially significant as a result of this initial review were then subjected to a thorough, detailed analysis. This extensive analysis was intended 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) and their amendments, individual plant examinations (IPEs), and other information related to the event of interest.

The detailed review of each event considered the immediate impact of an initiating event or 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 assessed to determine whether it impacted at-power or hot shutdown operation. If the event could only occur at cold shutdown or refueling shutdown, or the conditions clearly did not impact at-power operation, then its impact on continued decay heat removal during shutdown was assessed; otherwise it was analyzed as if the plant were at power. (Although no cold shutdown events were analyzed in the present study, some potentially significant shutdown-related events are described in Appendix D).

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER or multiple LERs, 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 the conditional probability of subsequent core damage was at least 1.0×10^{-6} (see section 2.2). Events of low significance are thus excluded, allowing attention to be focused on the more important events. This approach is consistent with the approach used to define 1988–1993 precursors, but differs from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria regardless of conditional core damage probability.

As noted above, 115 operational events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ were identified as accident sequence precursors.

2.1.2 Potentially Significant Shutdown-Related Events

No cold shutdown events were analyzed in this study because the lack of information concerning plant status at the time of the event (e.g., systems unavailable, decay heat loads, RCS heat-up rates, etc.) prevented development of models for such events. However, cold shutdown events such as a prolonged loss of RHR cooling during conditions of high decay heat can be risk significant. Sixteen shutdown-related events which may have potential risk significance are described in Appendix D.

2.1.3 Potentially Significant Events Considered Impractical to Analyze

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 and the resources available to the ASP Program.

Forty-three events (some involving more than a single LER) identified as potentially significant were considered impractical to analyze. It is thought that such events are capable of impacting core damage sequences. However, the events usually involve component degradations in which the extent of the degradation could not be determined or the impact of the degradation on plant response could not be ascertained.

For many events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a 1-year period (as would be done using a bounding analysis) would result in the conclusion that a very significant condition existed. This conclusion would not be supported by the specifics of the event as reported in the LER(s) or by the limited engineering evaluation performed in the ASP Program. Descriptions of events considered impractical to analyze are provided in Appendix E.

Selection Criteria and Quantification

2.1.4 Containment-Related Events

In addition to accident sequence precursors, events involving loss of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control, identified in the reviews of 1982-83 LERs are documented in Appendix F. It should be noted that the SCSS search algorithm does not specifically search for containment related events. These events, if identified for other reasons during the search, are then examined and documented.

2.1.5 "Interesting" Events

Otner events that provided insight into unusual failure modes with the potential to compromise continued core cooling but that were determined not to be precursors were also identified. These are documented as "interesting" events in Appendix G.

2.2 Precursor Quantification

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 the ASP models, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event trees and system models 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. Quantification results in a revised probability of core damage failure, given the operational event. 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. Details of the event modeling process and calculational results can be found in Appendix A of this report.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the light-water reactor (LWR) population for the 1982-86 time period, even though they are applied to sequences that are plant-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. Appendix A documents the accident sequence models used in the 1982-83 precursor analyses, and provides examples of the probability values used in the calculations.

The evaluation of precursors in this report considered equipment and recovery procedures believed to have been available at the various plants in the 1982-83 time frame. This includes features addressed in the current (1994) ASP models that were not considered in the analysis of 1984-91 events, and only partially in the analysis of 1992-93 events. These features include the potential use of the residual heat removal system for long-term decay heat removal following a small-break LOCA in PWRs, the potential use of the reactor core isolation cooling system to supply makeup following a small-break LOCA in BWRs, and core damage sequences associated with failure to trip the reactor (this condition was previously designated "ATWS," and not developed). In addition, the potential long-term recovery of the power conversion system for BWR decay heat removal has been addressed in the models.

Because of these differences in the models, and the need to assume in the analysis of 1982-83 events that equipment reported as failed near the time of a reactor trip could have impacted post-trip response (equipment response following a reactor trip was required to be reported beginning in 1984), the evaluations for these years may not be directly comparable to the results for other years.

Another difference between earlier and the most recent (1994) precursor analyses involves the documentation of the significance of precursors involving unavailable equipment without initiating events. These events are termed unavailabilities in this report, but are also referred to as condition assessments. The 1994 analyses distinguish a precursor conditional core damage probability (CCDP), which addresses the risk impact of the failed equipment as well as all other nominally functioning equipment during the unavailability period, and an importance measure defined as the difference between the CCDP and the nominal core damage probability (CDP) over the same time period. This importance measure, which estimates the increase in core damage probability because of the failures, was referred to as the CCDP in pre-1994 reports, and was used to rank unavailabilities.

For most unavailabilities that meet the ASP selection criteria, observed failures significantly impact the core damage model. In these cases, there is little difference between the CCDP and the importance measure. For some events, however, nominal plant response dominates the risk. In these cases, the CCDP can be considerably higher than the importance measure. For 1994 unavailabilities, the CCDP, CDP, and importance are all provided to better characterize the significance of an event. This is facilitated by the computer code used to evaluate 1994 events (the GEM module in SAPHIRE), which reports these three values.

The analyses of 1982-83 events, however, were performed using the event evaluation code (EVENTEVL) used in the assessment of 1984-93 precursors. Because this code only reports the importance measure for unavailabilities, that value was used as a measure of event significance in this report. In the documentation of each unavailability, the importance measure value is referred to as the increase in core damage probability over the period of the unavailability, which is what it represents. An example of the difference between a conditional probability calculation and an importance calculation is provided in Appendix A.

2.3 Review of Precursor Documentation

With completion of the initial analyses of the precursors and reviews by team members, this draft report containing the analyses is being transmitted to an NRC contractor, Oak Ridge National Laboratories (ORNL), for an independent review. The review is intended to (1) provide an independent quality check of the analyses, (2) ensure consistency with the ASP analysis guidelines and with other ASP analyses for the same event type, and (3) verify the adequacy of the modeling approach and appropriateness of the assumptions used in the analyses. In addition, the draft report is being sent to the pertinent nuclear plant licensees for review and to the NRC staff for review. Comments received from the licensees within 30 days will be considered during resolution of comments received from ORNL and NRC staff.

2.4 Precursor Documentation Format

The 1982-83 precursors are documented in Appendices B and C. The at-power events with conditional core damage probabilities (CCDPs) $\geq 1.0 \times 10^{-5}$ are contained in Appendix B and those with CCDPs between 1.0×10^{-5} and 1.0×10^{-6} are summarized in Appendix C. For the events in Appendix B, a description of the event

is provided with additional information relevant to the assessment of the event, the ASP modeling assumptions and approach used in the analysis, and analysis results. The conditional core damage probability calculations are documented and the documentation includes probability summaries for end states, the conditional probabilities for the more important sequences and the branch probabilities used. A figure indicating the dominant core damage sequence postulated for each event will be included in the final report. Copies of the LERs are not provided with this draft report.

2.5 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 1982-83 LERs.* For 1969–1981 and 1984–1987, all LERs reported during the year were evaluated for precursors. For 1988–1994 and for the present ASP study of 1982-83 events, only a subset of the LERs were evaluated after a computerized search of the SCSS data base. While this subset is thought 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 resulted from the screening process. *Reports to Congress on Abnormal Occurrences*[20] (NUREG-0900 series) and operating experience articles in *Nuclear Safety* were also reviewed for events that may have been missed by the SCSS computerized screening.

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 event information.* The accuracy and completeness of the LERs and other event-related documentation in reflecting pertinent operational information for the 1982-83 events are questionable in some cases. Requirements associated with LER reporting at the time, plus the approach to event reporting practiced at particular plants, could have resulted in variation in the extent of events reported and report details among plants. 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. Finally, the lack of a requirement at the time to link plant trip information to reportable events required that certain assumptions be made in the analysis of certain kinds of 1982-83 events. Specifically, through use of the "Grey Books" (*Licensed Operating Reactors Status Report*, NUREG-0200)[19] it was possible to determine that system unavailabilities reported in LERs could have overlapped with plant trips if it was assumed that the component could have been out-of-service for ½ the test/surveillance period associated with that component. However, with the link between trips and events not being described in the LERs, it was often impossible to determine whether or not the component was actually unavailable during the trip or whether it was demanded

during the trip. Nevertheless, in order to avoid missing any important precursors for the time period, any reported component unavailability which overlapped a plant trip within ½ of the component's test/surveillance period, and which was believed not to have been demanded during the trip, was assumed to be unavailable concurrent with the trip. (If the component had been demanded and failed, the failure would have been reported; if it had been demanded and worked successfully, then the failure would have occurred after the trip). Since such assumptions may be conservative, these events are distinguished from the other precursors listed in Tables 3.1 - 3.6. As noted above, these events are termed "windowed" events to indicate that they were analyzed because the potential time window for their unavailability was assumed to have overlapped a plant trip.

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. The system models are structured to reflect the plant-specific systems, at least to the train level. While major differences between plants are represented in this way, the plant models utilized in the analysis may not adequately reflect all important differences. 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 (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 1982-83 precursor analysis for some plants. This information was not uniformly available; much of it was based on FSAR and 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 have been consistently analyzed.

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 during 1982-83 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, and others, 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.

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. See reference 1 for a more comprehensive discussion of test interval assumptions.

Appendix A:
ASP MODELS

A.0 ASP Models

This appendix describes the methods and models used to estimate the significance of 1982-83 precursors. The modeling approach is similar to that used to evaluate 1984-91 operational events. Simplified train-based models are used, in conjunction with a simplified recovery model, to estimate system failure probabilities specific to an operational event. These probabilities are then used in event tree models that describe core damage sequences relevant to the event. The event trees have been expanded beyond those used in the analysis of 1984-91 events to address features of the ASP models used to assess 1994 operational events (Ref. 1) known to have existed in the 1982-83 time period.

A.1 Precursor Significance Estimation

The ASP program performs retrospective analyses of operating experience. These analyses require that certain methodological assumptions be made in order to estimate the risk significance of an event. If one assumes, following an operational event in which core cooling was successful, that components observed failed were "failed" with probability 1.0, and components that functioned successfully were "successful" with probability 1.0, then one can conclude that the risk of core damage was zero, and that the only potential sequence was the combination of events that occurred. In order to avoid such trivial results, the status of certain components must be considered latent. In the ASP program, this latency is associated with components that operated successfully—these components are considered to have been capable of failing during the operational event.

Quantification of precursor significance involves the determination of a conditional probability of subsequent core damage given the failures and other undesirable conditions (such as an initiating event or an unexpected relief valve challenge) observed during an operational event. The effect of a precursor on systems addressed in the core damage models is assessed by reviewing the operational event specifics against plant design and operating information, and translating the results of the review into a revised model for the plant that reflects the observed failures. The precursors's significance is estimated by calculating a conditional probability of core damage given the observed failures. The conditional probability calculated in this way 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 Types of Events Analyzed

Two different types of events are addressed in precursor quantitative analysis. In the first, an initiating event such as a loss of offsite power (LOOP) or small-break loss of coolant accident (LOCA) occurs as a part of the precursor. The probability of core damage for this type of event is calculated based on the required plant response to the particular initiating event and other failures that may have occurred at the same time. This type of event includes the "windowed" events subsetted for the 1982-83 ASP program and discussed in Section 2.2 of the main report.

The second type of event involves a failure condition that existed over a period of time during which an initiating event could have, but did not occur. The probability of core damage is calculated based on the required plant response to a set of postulated initiating events, considering the failures that were observed. Unlike an initiating event assessment, where a particular initiating event is assumed to occur with probability 1.0, each initiating event is assumed to occur with a probability based on the initiating event frequency and the failure duration.

A.1.2 Modification of System Failure Probabilities to Reflect Observed Failures

The ASP models used to evaluate 1982-83 operational events describe sequences to core damage in terms of combinations of mitigating systems success and failure following an initiating event. Each system model represents those combinations of train or component failures that will result in system failure. Failures observed during an operational event must be represented in terms of changes to one or more of the potential failures included in the system models.

If a failed component is included in one of the trains in the system model, the failure is reflected by setting the probability for the impacted train to 1.0. Redundant train failure probabilities are conditional, which allows potential common cause failures to be addressed. If the observed failure could have occurred in other similar components at the same time, then the system failure probability is increased to represent this. If the failure could not simultaneously occur in other components (for example, if a component was removed from service for preventive maintenance), then the system failure probability is also revised, but only to reflect the "removal" of the unavailable component from the model.

If a failed component is not specifically included as an event in a model, then the failure is addressed by setting elements impacted by the failure to failed. For example, support systems are not completely developed in the 1982-83 ASP models. A breaker failure that results in the loss of power to a group of components would be represented by setting the elements associated with each component in the group to failed.

Occasionally, a precursor occurs that cannot be modelled by modifying probabilities in existing system models. In such a case, the model is revised as necessary to address the event, typically by adding events to the system model or by addressing an unusual initiating event through the use of an additional event tree.

A.1.3 Recovery from Observed Failures

The models used to evaluated 1982-83 events address the potential for recovery of an entire system if the system fails. This is the same approach that was used in the analysis of most precursors through 1991.[1] In this approach, the potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes were used to describe the different types of short-term recovery that could be involved:

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[1] Later precursor analyses utilize Time-Reliability Correlations to estimate the probability of failing to recover a failed system when recovery is dominated by operator action.

Recovery Class	Likelihood of Non-Recovery[2]	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	0.55	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	0.10	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.01	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.

Substantial time is usually available to recover a failed residual heat removal (RHR) or BWR power conversion system (PCS). For these systems, the nonrecovery probabilities listed above are overly conservative. Data in Refs. 2 and 3 was used to estimate the following nonrecovery probabilities for these systems:

System	p(nonrecovery)
BWR RHR system	0.016 (0.054 if failures involve service water)
BWR PCS	0.52 (0.017 for MSIV closure)
PWR RHR system	0.057

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.

A.1-4 Conditional Probability Associated with Each Precursor

As described earlier in this appendix, the calculation process for each precursor involves a determination of initiators that must be modeled, plus any modifications to system probabilities necessitated by failures observed

[2]These nonrecovery probabilities are consistent with values specified in M.B. Sattison *et al.*, "Methods Improvements Incorporated into the SAPHIRE ASP Models," *Proceedings of the U.S. Nuclear Regulatory Commission Twenty-Second Water Reactor Safety Information Meeting*, NUREG/CP-0140, Vol. 1, April 1995.

in an operational event. Once the probabilities that reflect the conditions of the precursor are established, the sequences leading to core damage are calculated to estimate the conditional probability for the precursor. This calculational process is summarized in Table A.1.

Several simplified examples that illustrate the basics of precursor calculational process follow. It is not the intent of the examples to describe a detailed precursor analysis, but instead to provide a basic understanding of the process.

The hypothetical core damage model for these examples, shown in Fig. A.1, consists of initiator I and four systems that provide protection against core damage: system A, B, C, and D. In Fig. A.1, the up branch represents success and the down branch failure for each of the systems. Three sequences result in core damage if completed: sequence 3 [I /A ("/" represents system success) B C], sequence 6 (I A /B C D) and sequence 7 (I A B). In a conventional PRA approach, the frequency of core damage would be calculated using the frequency of the initiating event I, $\lambda(I)$, and the failure probabilities for A, B, C, and D [p(A), p(B), p(C), and p(D)]. Assuming $\lambda(I) = 0.1\ yr^{-1}$ and $p(A|I) = 0.003$, $p(B|IA) = 0.01$, $p(C|I) = 0.05$, and $p(D|IC) = 0.1$,[3] the frequency of core damage is determined by calculating the frequency of each of the three core damage sequences and adding the frequencies:

$$0.1\ yr^{-1} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} +$$
$$0.1\ yr^{-1} \times 0.003 \times (1 - 0.01) \times 0.05 \times 0.1 \text{ (sequence 6)} +$$
$$0.1\ yr^{-1} \times 0.003 \times 0.01 \text{ (sequence 7)}$$

$$= 4.99 \times 10^{-4}\ yr^{-1} \text{ (sequence 3)} + 1.49 \times 10^{-6}\ yr^{-1} \text{ (sequence 6)} + 3.00 \times 10^{-6}\ yr^{-1} \text{ (sequence 7)}$$

$$= 5.03 \times 10^{-4}\ yr^{-1}.$$

In a nominal PRA, sequence 3 would be the dominant core damage sequence.

The ASP program calculates a conditional probability of core damage, given an initiating event or component failures. This probability is different than the frequency calculated above and cannot be directly compared with it.

<u>Example 1. Initiating Event Assessment</u>. Assume that a precursor involving initiating event I occurs. In response to I, systems A, B, and C start and operate correctly and system D is not demanded. In a precursor initiating event assessment, the <u>probability</u> of I is set to 1.0. Although systems A, B, and C were successful, nominal failure probabilities are assumed. Since system D was not demanded, a nominal failure probability is assumed for it as well. The conditional probability of core damage associated with precursor I is calculated by summing the conditional probabilities for the three sequences:

$$1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} +$$
$$1.0 \times 0.003 \times (1 - 0.010) \times 0.05 \times 0.1 \text{ (sequence 6)} +$$
$$1.0 \times 0.003 \times 0.01 \text{ (sequence 7)}$$

[3] The notation p(B | IA) means the probability that B fails, given I occurred and A failed.

$$= 5.03 \times 10^{-3}.$$

If, instead, B had failed when demanded, its probability would have been set to 1.0. The conditional core damage probability for precursor IB would be calculated as

$$1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 1.0 \times 0.003 \times 1.0 \text{ (sequence 7)} = 7.99 \times 10^{-3}.$$

Since B is failed sequence 6 cannot occur.

<u>Example 2. Condition Assessment</u>. Assume that during a monthly test system B is found to be failed, and that the failure could have occurred at any time during the month. The best estimate for the duration of the failure is one half of the test period, or 360 h. To estimate the probability of initiating event I during the 360 h period, the yearly frequency of I must be converted to an hourly rate. If I can only occur at power, and the plant is at power for 70% of a year, then the frequency for I is estimated to be $0.1\ \text{yr}^{-1}/(8760\ \text{h/yr} \times 0.7) = 1.63 \times 10^{-5}\ \text{h}^{-1}$.

If, as in example 1, B is always demanded following I, the probability of I in the 360 h period is the probability that at least one I occurs (since the failure of B will then be discovered), or

$$1 - e^{-\lambda(I) \times \text{failure duration}} = 1 - e^{-1.63E\text{-}5 \times 360} = 5.85 \times 10^{-3}.$$

Using this value for the probability of I, and setting p(B) = 1.0, the conditional probability of core damage for precursor B is calculated by again summing the conditional probabilities for the core damage sequences in Fig. A.1:

$$5.85 \times 10^{-3} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 5.85 \times 10^{-3} \times 0.003 \times 1.0 \text{ (sequence 7)}$$

$$= 4.67 \times 10^{-5}.$$

As before, since B is failed, sequence 6 cannot occur. The conditional probability is the probability of core damage in the 360 h period, given the failure of B. Note that the dominant core damage sequence is sequence 3, with a conditional probability of 2.92×10^{-5}. This sequence is unrelated to the failure of B. The potential failure of systems C and D over the 360 h period still drive the core damage risk.

To understand the significance of the failure of system B, another calculation, an importance measure, is required. The importance measure that is used is equivalent to risk achievement worth on an interval scale (see Ref. 4). In this calculation, the increase in core damage probability over the 360 h period due to the failure of B is estimated: p(cd | B) - p(cd). For this example the value is $4.67 \times 10^{-5} - 2.94 \times 10^{5} = 1.73 \times 10^{5}$, where the second term on the left side of the equation is calculated using the previously developed probability of I in the 360 h period and nominal failure probabilities for A, B, C, and D.

For most conditions identified as precursors in the ASP program, the importance and the conditional core damage probability are numerically close, and either can be used as a significance measure for the precursor. However, for some events—typically those in which the components that are failed are not the primary mitigating plant features—the conditional core damage probability can be significantly higher than the importance. In such cases, it is important to note that the potential failure of other components, unrelated to the precursor, are still dominating the plant risk.

ASP MODELS

The importance measure for unavailabilities (condition assessments) like this example event were previously referred to as a "conditional core damage probability" in annual precursor reports before 1994, instead of as the increase in core damage probability over the duration of the unavailability. Because the computer code used to analyze 1982-83 events is the same as was used for 1984-93 evaluations, the results for 1982-83 conditions are also presented in the computer output in terms of "conditional probability," when in actuality the result is an importance.

A.2 Overview of 1982-83 ASP Models

Models used to rank 1982-83 precursors as to significance consist of system-based plant-class event trees and simplified plant-specific system models. These models describe mitigation sequences for the following initiating events: a nonspecific reactor trip [which includes loss of feedwater (LOFW) within the model], LOOP, small-break LOCA, and steam generator tube rupture [SGTR, pressurized water reactors (PWRs) only].

Plant classes were 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. ASP plant categorization is described in the following section.

The event trees consider two end states: success (OK), in which core cooling exists, and core damage (CD), in which adequate core cooling is believed not to exist. In the ASP models, core damage is assumed to occur following core uncovery. It is acknowledged that clad and fuel damage will occur at later times, depending on the criteria used to define "damage," and that time may be available to recover core cooling once core uncovery occurs but before the onset of core damage. However, this potential recovery is not addressed in the models. Each event tree describes combinations of system failures that will prevent core cooling, and makeup if required, in both the short and long term. Primary systems designed to provide these functions and alternate systems capable of also performing these functions are addressed.

The models used to evaluate 1982-83 events consider both additional systems that can provide core protection and initiating events not included in the plant-class models used in the assessment of 1984-91 events, and only partially included in the assessment of 1992-93 events. Response to a failure to trip the reactor is now addressed, as is an SGTR in PWRs. In PWRs, the potential use of the residual heat removal system following a small-break LOCA (to avoid sump recirculation) is addressed, as is the potential recovery of secondary-side cooling in the long term following the initiation of feed and bleed. In boiling water reactors (BWRs), the potential use of reactor core isolation cooling (RCIC) and the control rod drive (CRD) system for makeup if a single relief valve sticks open is addressed, as is the potential long-term recovery of the power conversion system (PCS) for decay heat removal in BWRs. These models better reflect the capabilities of plant systems in preventing core damage.