U.S. Nuclear Regulatory Commission Information and Analyses

Edited by D. E. Hickman

1994 Accident Sequence Precursor Program Results

By R. J. Belles,[a] J. W. Cletcher,[a] D. A. Copinger,[a] B. W. Dolan,[b] J. W. Minarick,[b] and P. D. O'Reilly[c]

Abstract: *The Accident Sequence Precursor (ASP) Program involves the systematic review and evaluation of operational events that have occurred at light-water reactors to identify and categorize precursors to potential severe core damage accident sequences. The results of the ASP Program are published in an annual report. The most recent report, which contains the analyses of the precursors for 1994, is NUREG/CR-4674, Vols. 21 and 22,* Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report, *published in December 1995. This article provides an overview of the ASP review and evaluation process and a summary of the results for 1994.*

INTRODUCTION

The Accident Sequence Precursor (ASP) Program involves the systematic review and evaluation of operational events or conditions that have occurred at licensed U.S. commercial light water reactors (LWRs). The principal objectives of the program are to quantify and rank the safety significance of operating reactor events, to determine their generic implications, to characterize risk insights, and to document and disseminate the evaluations for feedback to plant operators to promote learning from experience. Further details about the ASP Program and the 1994 precursors may be found in Refs. 1 and 2.

An accident sequence precursor is an operational event or a plant condition that is an important element of a postulated accident sequence associated with inadequate core cooling, a sequence that would be expected to result in core damage. The ASP methodology is used to evaluate disparate elements of operational experience, with random failure probabilities used for other branches of the event tree models. The figure of merit for ASP analyses is the conditional core damage probability (CCDP). The CCDP is the conditional probability of core damage given the failures observed in the event. Events with CCDPs greater than 1.0×10^{-6} are considered accident sequence precursors.

The results of the ASP analyses are considered indications of the level of risk associated with operating nuclear power plants based on direct assessment of actual operating experience. The precursor events from the ASP Program comprise a unique database of historical system failures, multiple losses of redundancy, and infrequent core damage initiators.

Licensee Event Reports (LERs) serve as the chief source of operational experience data for the ASP Program. The reporting requirements for LERs are described in NUREG-1022, *Licensee Event Report System, Description of System and Guidelines for Reporting*.[3]

[a]Oak Ridge National Laboratory.
[b]Science Applications International Corp.
[c]U.S. Nuclear Regulatory Commission.

SELECTION CRITERIA AND QUANTIFICATION

Precursor Selection Criteria

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. ASP Program staff examine LERs to determine the impact of operational events on potential core damage sequences. Previous experience has shown that most ASP events can be directly or indirectly associated with the following four types of initiators: reactor 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 (SG) tube rupture.

Screening

This section describes the steps used to identify 1994 operational events for quantification. Figure 1 illustrates the process.

A computerized search of the Sequence Coding and Search System (SCSS) database 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 selected LERs potentially involving failures in plant systems that provide protective functions for the plant for core-damage-related initiating events. A review of the 1984–1987 precursor evaluations and all 1990 LERs determined that this computerized search successfully identifies almost all precursors within a subset of approximately one-third to one-half of all LERs.

Engineering Review

LERs were also selected for review if an Augmented Inspection Team (AIT) or Incident Investigation Team (IIT) report was written regarding the event. In addition, the Nuclear Regulatory Commission (NRC) staff designated other events for inclusion in the review process.

The selected events were independently reviewed by two NOAC staff members. Each LER was reviewed to determine whether the reported event should be examined in greater detail. This initial review was a bounding review that was meant to capture events that in any way appeared to deserve detailed analysis and to eliminate events that were clearly unimportant. Accordingly, events that satisfied predefined criteria for rejection were eliminated, and all others were accepted as potentially significant and requiring analysis.

LERs were eliminated from further consideration as precursors if they involved 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 an LOFW
- An event with no appreciable impact on safety systems
- An event involving only post-core-damage impacts

Events identified for further consideration typically included the following:

- Core damage initiators (LOOP and small-break LOCA)
- Events in which reactor trip was demanded and a safety-related component failed
- 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
- Any event that, on the basis of the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage

Detailed Analysis

Events determined to be potentially significant as a result of this 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 analyses were not limited to the LERs; they also used final safety analysis reports and their amendments, individual plant

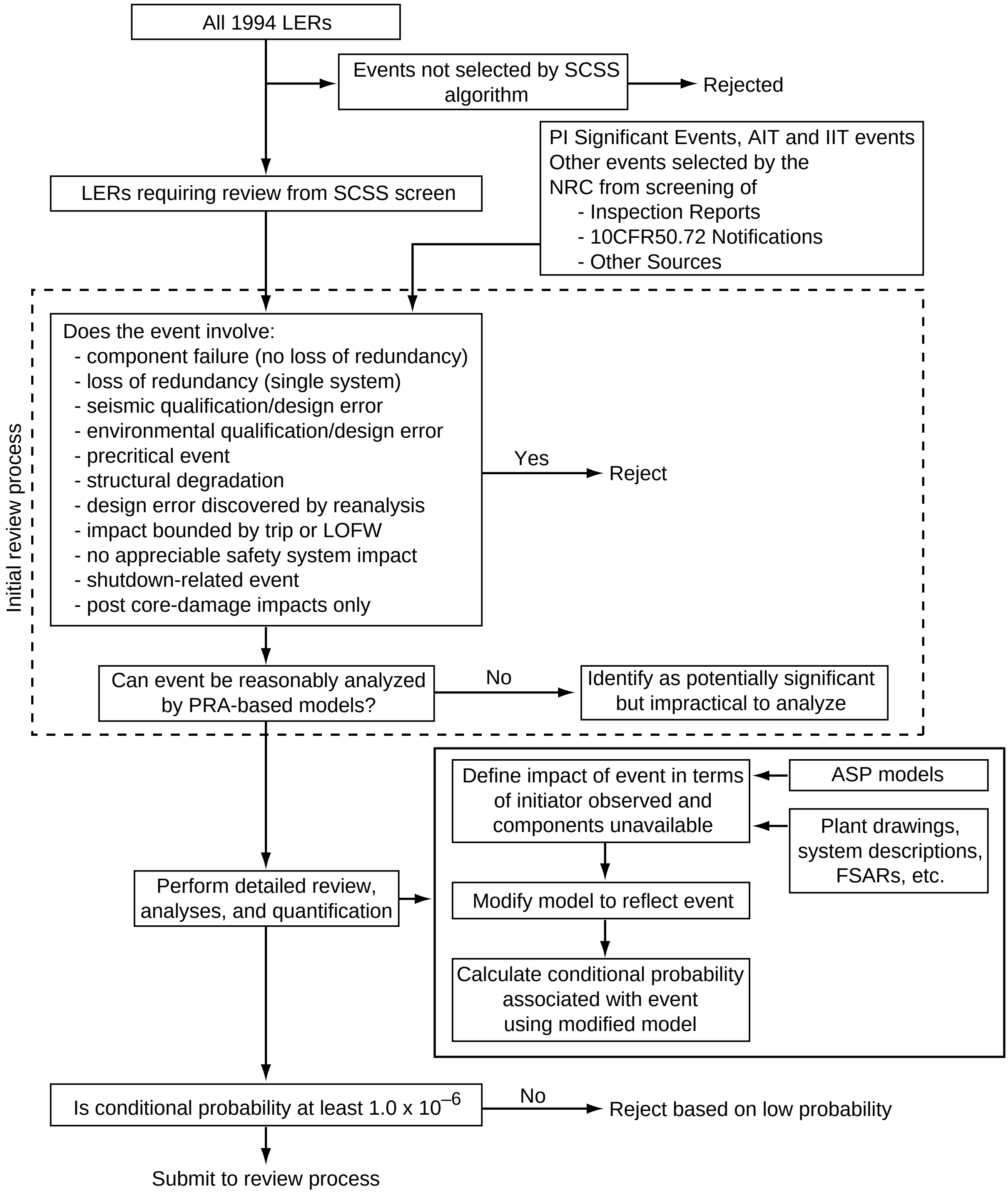

Fig. 1 ASP analysis process. (LER is Licensee Event Report, SCSS is Sequence Coding and Search System, PI is NRC's Performance Indicator Program, AIT is Augmented Inspection Team, IIT is Incident Investigation Team, LOFW is loss of feedwater, PRA is probabilistic risk assessment, ASP is accident sequence precursor, and FSAR is Final Safety Analysis Report.)

examinations, and other available information related to the event of interest.

The detailed analysis of each event considered the immediate impact of an initiating event or the potential impact of equipment failures or operator errors on the 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 period of failure.

3. If the event or failure was identified while the plant was not at power, then the event was first assessed to determine whether it could have impacted at-power operation. If the event could have impacted at-power operation, that impact was assessed. If the event could only occur at cold shutdown or refueling shutdown, then its impact on continued decay heat removal during shutdown 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 mitigation systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents if the conditional probability of subsequent core damage was at least 1.0×10^{-6}. Events of low significance were thus excluded, which allowed attention to be focused on the more important events.

Other Event Categories

In addition to precursors, three other categories of events are identified in the ASP Program review process: containment-related events, interesting events, and potentially significant events considered impractical to analyze.

1. Containment-related events involve loss of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control.

2. Interesting events provide insight into unusual failure modes with the potential to compromise continued core cooling but are not considered precursors.

3. Potentially significant events are considered impractical to analyze because of lack of information or inability to model the event reasonably within a PRA framework, considering the level of detail typically available in PRA models and the resources available to the ASP Program. Such events are thought to be capable of impacting core damage sequences; however, these 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.

1994 RESULTS

This section summarizes the results of the review and evaluation of 1994 operational events. The primary result of the ASP Program is the identification of operational events and conditions with CCDPs greater than 1.0×10^{-6} that satisfy at least one of the following four precursor selection criteria: (1) a core damage initiator requiring safety system response, (2) the failure of a complete system required to mitigate the consequences of a core damage initiator, (3) degradation of more than one system required for mitigation, or (4) a trip or loss of feedwater with a degraded mitigating system. In 1994, there were nine ASP events. Eight of these were analyzed as at-power events and one was analyzed as a shutdown event. These analyses are documented in Appendices C and D of Ref. 1.

Direct comparison of the 1994 results with those of earlier years is not possible without substantial effort to reconcile differences in analytical methods. Therefore only limited observations are provided here. The 1986 precursor report[4] includes a discussion of observations gleaned from the results for 1984–1986, and the 1987–1993 reports[5–11] include a similar discussion of the results for those years.

TABULATION OF 1994 PRECURSOR EVENTS

The 1994 ASP events are listed in Tables 1 to 3. The following information is included in each table:

- Conditional probability of potential core damage associated with the event (CCDP)
- Name of the plant where the event occurred (Plant)
- Plant type
- LER or inspection report (IR) number associated with the event (Event identifier)
- A brief description of the event (Description)
- Event date
- Initiator of the event or unavailability if no initiator was involved (Event type)

The tables are sorted as follows:

- Table 1: At-power precursors involving unavailabilities sorted by CCDP
- Table 2: At-power precursors involving initiating events sorted by CCDP
- Table 3: Shutdown precursors involving initiating events

Table 1 At-Power Precursors Involving Unavailabilities Sorted by CCDP

CCDP	Plant	Plant type	Event identifier	Description	Event date	Event type
1.4×10^{-4}	Haddam Neck	PWR	LERs 213/94-004, -005, -007, -013 IR 213/94-03	Power-operated relief valves and vital 480-V ac bus degraded	2/16/94	Unavailability
2.3×10^{-5}	Zion 2	PWR	LER 304/94-002	Unavailability of turbine-driven auxiliary feedwater pump and emergency diesel generator	3/7/94	Unavailability
1.2×10^{-5}	Point Beach 1 and 2	PWR	LER 266/94-002	Both diesel generators inoperable	2/8/94	Unavailability
6.1×10^{-6}	Dresden 2	BWR	LER 237/94-018	Motor control center trips due to improper breaker settings	6/8/94	Unavailability
3.1×10^{-6}	Dresden 2	BWR	LER 237/94-021	Long-term unavailability of high-pressure coolant injection	8/4/94	Unavailability
1.8×10^{-6}	Turkey Point 3 and 4	PWR	LER 250/94-005	Load sequencers periodically inoperable	11/3/94	Unavailability

Table 2 At-Power Precursors Involving Initiating Events Sorted by CCDP

CCDP	Plant	Plant type	Event identifier	Description	Event date	Event type
1.8×10^{-5}	River Bend	BWR	LER 458/94-023	Scram, main turbine-generator fails to trip, reactor core isolation cooling and control rod drive systems unavailable	9/8/94	Reactor trip
1.3×10^{-5}	Calvert Cliffs 2	PWR	LER 318/94-001	Trip, loss of 13.8-kV bus, and short-term saltwater cooling system unavailable	1/12/94	Reactor trip

Table 3 Shutdown Precursors Involving Initiating Events

CCDP	Plant	Plant type	Event identifier	Description	Event date	Event type
3.0×10^{-3}	Wolf Creek	PWR	IR 482/94-018	Reactor coolant system blows down to refueling water storage tank during hot shutdown	9/17/94	Interfacing systems LOCA

Containment-Related Events

One containment-related event was found in 1994: a design error discovered at Millstone 2 that could result in an untreated release to the atmosphere from the enclosure building.

Interesting Events

Nine "interesting" events were found in 1994. One particularly interesting event occurred at Salem 1. Following an unexpected reactor trip, two safety injections (SIs) were automatically initiated. The first was caused by a main steam pressure pulse and resulted in the pressurizer filling completely with water. This condition is sometimes referred to as a "solid" condition. The second SI was caused by a rapid decrease in reactor system pressure when a secondary-side safety valve opened with the pressurizer "solid." The pressurizer power-operated relief valves (PORVs) actuated over 300 times during the event. Complete descriptions of this event and other interesting events are documented in Appendix G of Ref. 1.

Potentially Significant Events That Were Impractical to Analyze

Twelve potentially significant events in 1994 were considered impractical to analyze. These events are documented in Appendix E of Ref. 1.

IMPORTANT PRECURSORS

Two 1994 precursors that had CCDPs greater than 10^{-4} were identified. Events with conditional core damage probabilities of that magnitude are considered important in the ASP Program. These events are described in the following sections.

Wolf Creek—RCS Blows Down to Refueling Water Storage Tank During Hot Shutdown

At 4:00 a.m. on September 17, 1994, Wolf Creek was in Mode 4 preparing to begin a refueling outage with the reactor coolant system (RCS) at 350 psig and 300 °F. Two reactor coolant pumps (RCPs) were in service, the SGs were filled, and the condenser and condensate systems were secured. The SI pumps and one of two centrifugal charging pumps were out of service with breakers open to prevent low-temperature overpressurization. Residual heat removal (RHR) train A was in service to provide shutdown cooling, and maintenance was being performed on RHR valve 8716A, the A RHR-to-SI system hot leg recirculation isolation valve (Fig. 2). RHR train B was being lined up for recirculation to the refueling water storage tank (RWST) to raise boron concentration before placing the train in service. This required opening valve 8717, a manual valve in the 8-in. common line from the RHR pump discharge headers to the RWST emergency core cooling systems (ECCSs) pump suction header.

A nuclear station operator (NSO) was dispatched to open valve 8717. The control room operators then received a call from a plant electrician requesting that valve 8716A be stroked (opened and reclosed) in support of postmaintenance testing. Meanwhile, the NSO had arrived at valve 8717 and prepared to open it. Approximately 3 ft from the NSO, the electrician was working on valve 8716A, but neither he nor the NSO recognized the significance of opening valves 8717 and 8716A simultaneously. When opened together, these valves provide a direct pathway from the RHR pump discharge to the RWST via the ECCS suction header. When the control room operator opened valve 8716A from the control room, the operator stationed at valve 8717 apparently had only begun opening it. Water flowed from the RCS to the RWST until valve 8716A was reclosed; during this time the pressurizer level dropped about 2%, although this was not noted until the event was reviewed later.

The control room operator waited about 30 s and then reopened valve 8716A. Valve 8717 was fully open by this time, and reactor coolant inventory began rapidly flowing to the RWST. The operator stationed at valve 8717 observed loud flow and water hammer noises and called the control room to report them. At the same time, control room personnel received a high RWST level alarm, the pressurizer high-level annunciator cleared, and the pressurizer level instrumentation indicated low.

Operators responded by tripping the RCPs, increasing charging flow, and manually isolating letdown. A relief crew supervising operator who was present in the control room determined that both valves 8716A and 8717 were open and informed the control room operator that valve 8716A should be closed. This was done and the flow path was isolated about 66 s into the event.

While the blowdown was in progress, about 9200 gal flowed from the RCS to the RWST, causing the RWST to overflow. Approximately 650 gal overflowed from the RWST to the waste holdup tank. The

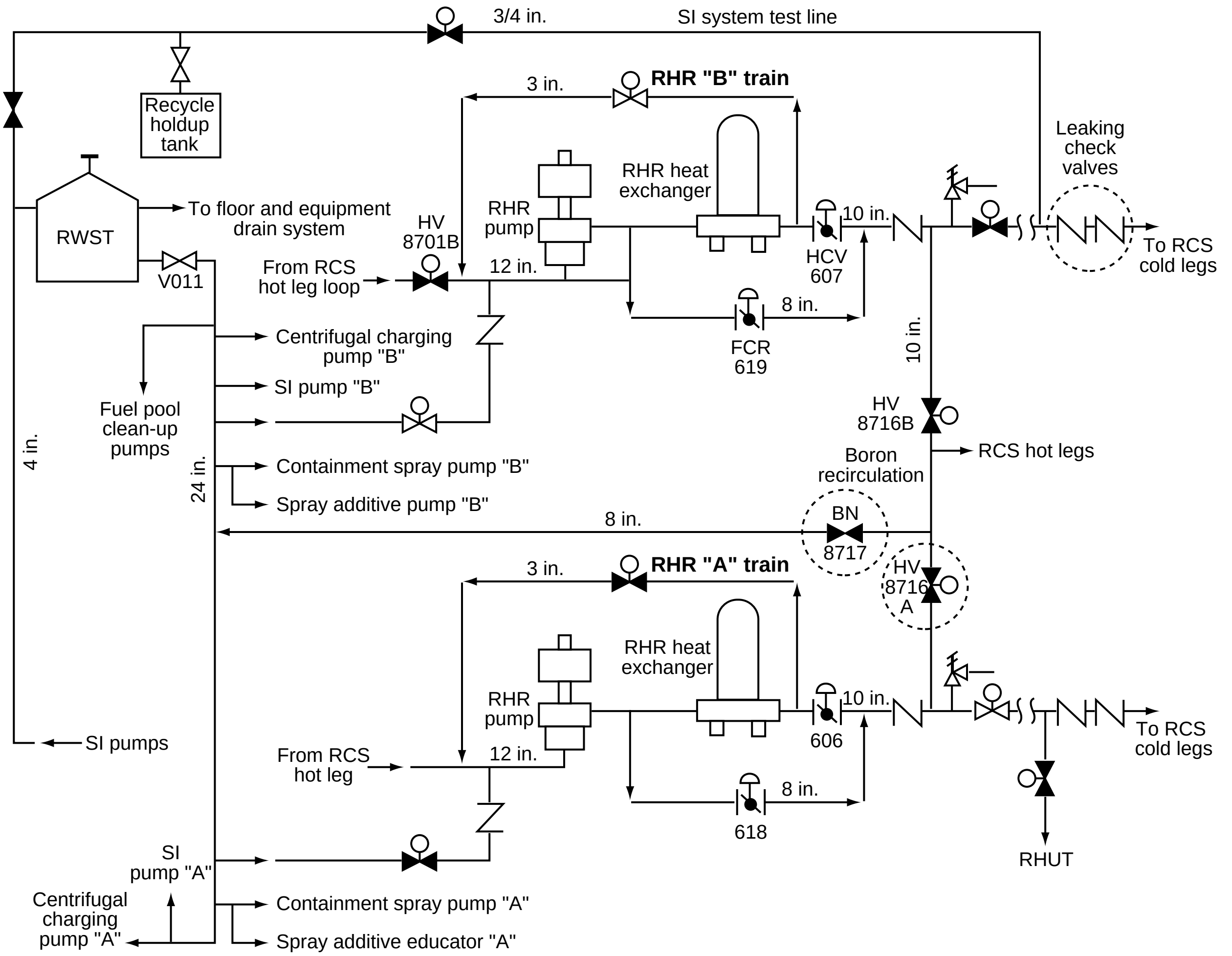

Fig. 2 Valve lineup before event. (RCS is reactor coolant system, SI is safety injection, RWST is refueling water storage tank, RHR is residual heat removal, and RHUT is radioactive waste holdup tank.)

RHR and charging systems remained in service, and RCS level was gradually restored.

A subsequent Westinghouse analysis of this event determined that, had the blowdown not been quickly isolated, the primary system could have drained down to the RCS loop elevation in as little as 3 min. The RWST ECCS suction header could have filled with steam shortly thereafter. Unisolated, the blowdown could have led to core uncovery in as little as 30 min.

Westinghouse further determined that an operating RHR pump could have been damaged by as little as 0.5 min of operation after the primary system drained down to the RCS loop elevation. The analysis also indicates that, once the RWST ECCS suction header filled with steam, operation of the multistage SI pumps could have resulted in their failure. Isolation of the blowdown path would have allowed water to flow from the RWST back into the suction header; however, there was no assurance that the ECCS pumps could fulfill their functions while drawing water from the RWST following such an event. The study also indicates that, with the suction header filled with steam, recovery of the RHR pumps would be problematic even if they were shut off in time. In less than the time required to fill, vent, and restart an RHR pump, reactor pressure could exceed the RHR reactor high-pressure shutoff point.

The ASP evaluation of this event is strongly influenced by assumptions regarding human reliability, the time and degree of effort required to recover ECCS systems, and the viability of the reflux cooling method wherein steam from a boiling core may be condensed in the SG tubes with the condensate draining back to the reactor. Substantial uncertainty is associated with each of these assumptions.

Approximately 3 min were available for the operators to diagnose and isolate the blowdown before all RHR and ECCS pumps could have been rendered inoperable. Although procedures did not address the response to this condition, one operator's understanding of the existing system alignment allowed him to diagnose and correct the problem within 66 s. To estimate the likelihood that operators would fail to isolate the blowdown before uncovering the RCS loops, the time reliability correlation (TRC) models from *Human Reliability Analysis*[12] were employed. Operator response within the first 3 min was assumed to be rule-based and without hesitancy. This is considered appropriate on the basis of the indications available to the operators at the time. Setting the median response time to the response time observed in this event (~60 s) and using Table 10-8 of Ref. 12 results in an estimated crew error probability of 0.06.

Had the operators failed to isolate the blowdown path within 3 min, a direct vent path to atmosphere would have been established from the RCS through the RWST. Analyses were performed showing that core damage could have occurred as early as 27 min later. After the RCS loops voided at 3 min, the ECCS common suction header would have begun to void. Recovery of the event at this point would require more difficult operator actions. These actions were considered recovery (general diagnosis that must be used in the absence of rules) with hesitancy (because of conflict, burden, and uncertainty) within the context of the TRC model. On the basis of Table 10-11 in Ref. 12, a crew failure probability of 0.05 was estimated for the 27-min period.

If the blowdown had been isolated after the loops voided (after 3 min, but before 30 min), substantial time and effort would have been required to refill and vent the RWST ECCS suction header and the ECCS pump suction lines that are aligned to it. Without extensive venting and priming, the high-pressure pumps would be expected to fail. An analysis performed by Westinghouse indicates that significant voids entrained in the suction supply (5 to 20%) would guarantee a loss of ECCS prime, and other analyses have shown that operation in that condition for more than 1 or 2 min would cause pump failure. The high-pressure ECCS pumps therefore were assumed in this analysis to be unavailable once the RWST ECCS suction header voided.

A conservative analysis (without consideration of the SG secondary-side inventory that existed during the event) showed that without some form of decay heat removal pressure in the RCS could exceed the RHR pump shutoff head within 15 min. This is less than the time that would likely be required to restore the RHR system to service. Because the PORVs were found to be inoperable subsequent to the event, it was assumed that depressurization of the RCS would have been difficult to achieve. The RHR pumps were therefore assumed to be inoperable once the RWST ECCS suction header voided. The only remaining decay-heat removal path would have been reflux cooling via the SGs. The SGs were available during the event, and reflux cooling was considered a viable core cooling method. In the short term, the water inventory in the SG could have provided decay-heat removal. Eventually, SG makeup and the opening of atmospheric vent valves would have been required for continued heat removal via this method. Reflux cooling is assumed to require two SGs for success. Assuming both motor-driven auxiliary feedwater pumps and all four SGs and their atmospheric dump valves were available, a failure probability of $\sim 7.0 \times 10^{-4}$ was estimated for reflux cooling on the basis of component failure probabilities used in the IRRAS-based ASP models for Wolf Creek. This estimate addresses equipment availability only and not the uncertainty in the viability of the reflux cooling method. Because consideration of such uncertainty is beyond the scope of this analysis, the potential impact of reflux cooling being unavailable or ineffective was addressed in a sensitivity analysis.

The CCDP for this event is estimated to be 3.0×10^{-3}. This estimate may be pessimistic because all ECCS pumps are assumed to be unavailable once significant voiding occurs in the ECCS common suction header. Assumptions concerning the viability of reflux cooling play an important role in the CCDP estimated for this event; for example, an assumed failure of ~0.05 for reflux cooling raises the estimated CCDP by a factor of 2, to 6.0×10^{-3}.

Although this event was terminated quickly, several important lessons were learned relating to an extended blowdown that might have occurred under other circumstances. These lessons included (1) the previously unrecognized design vulnerability in the piping arrangement that connects the discharge of both trains of RHR with the RWST header, (2) the failure to adequately control work activities that resulted in the initiation of the event, and (3) the uncertainties in the ability of the operators to mitigate an extended blowdown of this type.[a]

Haddam Neck—PORVs and Vital 480-V ac Bus Degraded

During testing on February 16, 1994, it was discovered that one of two feed breakers to motor control center-5 (MCC-5) jammed and failed to close when demanded. MCC-5 is powered from both safety-related trains through an automatic bus transfer (ABT) scheme and supplies power to a number of vital components in both safety-system trains. During testing on February 19, 1994, it was discovered that air operators for the pressurizer PORVs were experiencing control air leaks and that the PORVs could not be operated properly from their safety-grade control air supply. Investigation revealed that repairs to fix a prior PORV failure were made during the previous refueling outage. The PORV diaphragms were not seated correctly and were coated with a lubricant rather than a required sealant. A substantial air leak resulted, and the PORVs could not be opened more than 50%. In addition, the PORVs could not have been operated successfully from their emergency accumulator air supplies. The CCDP estimated for these events is 1.4×10^{-4}.

Surveillance testing of the PORVs in May 1993 identified one valve that was experiencing leakage from its diaphragm assembly. This leak, in conjunction with failure of the associated air pressure regulator, resulted in excessive air consumption. Had the system been demanded, operator action to isolate the leaking PORV would have been required to ensure an adequate long-term supply of control air to the other PORV.

Repairs to the system, including replacement of the PORV diaphragms, were completed before the end of the 1993 refueling outage. The design of the replacement diaphragms differed somewhat from the original ones, which may have contributed to difficulties experienced during the replacement process. Errors made during replacement included incorrect installation and the use of a lubricant instead of a sealant around the diaphragms' bolt circle. This allowed the diaphragms to extrude out between the sections of their housing, which damaged the diaphragms and created a pathway for air leakage from some unknown time after they were replaced until the condition was discovered on February 19, 1994. An NRC inspection team report indicates that both valves could only be opened about 50% during testing. The LER for the event indicates that the two safety functions were potentially compromised by the PORV failures: feed-and-bleed cooling and high-pressure safety injection (HPSI) makeup during certain small-break LOCAs.

The HPSI pumps at Haddam Neck do not develop sufficient discharge head for feed-and-bleed cooling without the operators opening the PORVs. Air is supplied to the PORVs from the containment air compressors. The containment air compressors are located within the containment building and are not rated for the environmental conditions that could occur during feed-and-bleed cooling, so the compressors could be expected to fail under such conditions. The PORVs are also provided with safety-related control air accumulators that maintain a reserve supply of control air in the event of compressor failure, but these accumulators were inadequate to operate the PORVs during the time that the air-operator diaphragms were damaged.

During a period of time overlapping the PORV unavailability, the ABT circuit for MCC-5 failed when tested. At the time of the event, MCC-5 supplied many pieces of important equipment in both trains, including equipment that would have been required for successful operation of HPSI, low-pressure safety injection (LPSI), recirculation, long-term cooling, containment spray, RCS loop operation, one PORV block valve, emergency boration, feedwater isolation, RCP seal cooling, service water, control air, and the closed cooling water system. Subsequent to this event, modifications were made to reduce the dependency upon MCC-5.

MCC-5 can be supplied from either 480-V ac bus 5 (emergency train A) or bus 6 (emergency train B). Normally, it is aligned such that bus 5 is the preferred supply and bus 6 is the alternate supply. At the time of the event, if the preferred supply was lost, an ABT system would align MCC-5 to the alternate bus. If power was subsequently restored to the preferred bus, the ABT would realign MCC-5 back to the preferred bus. For the test of the ABT system, bus 5 was deenergized. As designed, the breaker supplying MCC-5 from bus 5 opened, and the supply breaker from bus 6 automatically closed to restore power. When bus 5 was reenergized, MCC-5 automatically realigned itself to bus 5. During the second part of the test, the preferred power source selector switch

[a]For a more detailed description of this event, see *Nuclear Safety*, 36(2): 335-343 (July–December 1995).

(PPSSS) for the ABT was moved to make bus 6 the preferred power supply and bus 5 the alternate. When the PPSSS was moved to the bus 6 position, the bus 5 supply breaker opened as expected, but the bus 6 supply breaker failed to automatically close and thus deenergized MCC-5.

Subsequent investigation revealed that a mechanical defect in the MCC-5 feeder breaker from bus 6 prevented it from closing, which caused the breaker to randomly fail. With bus 6 still energized and selected as the preferred power source to MCC-5, the bus 5 supply to MCC-5 was prevented from closing by the ABT system logic.

The event was modeled as an unavailability of the PORVs for feed-and-bleed cooling and of the bus 6 feeder breaker for MCC-5. The last successful operation of the PORVs was during an outage in May and June 1993 following installation of the new diaphragms. The probable cause of the PORV failure was incorrect installation of the air-operator diaphragms during the 1993 outage. It was therefore assumed that the PORVs were inoperable for feed-and-bleed cooling from July 1993 until the leakage was discovered on February 19, 1994.

The defect that led to the intermittent failure of the bus 6 feeder breaker was presumed to have existed from the time of the previous failure during the June 1993 refueling outage until the time of this event in February 1994. The interval analyzed was the period from July 21, 1993, until February 19, 1994, a period of 234 days (5616 h).

The analysis of this event is similar to that of LER 213/93-007 and AIT Report 213/93-80 provided in the 1993 ASP Program Annual Report.[11] That analysis also dealt with failures of PORV control air system components coincident with the inoperability of the MCC-5 ABT.

The estimated CCDP for this combined event is 1.4×10^{-4}. A postulated LOOP contributes ~78% to the CCDP. The dominant sequence, which contributes about 30% of the total, involves a postulated LOOP during the 5616-h period, emergency power success, recovery of ac power and MCC-5, and failure of AFW and feed-and-bleed cooling.

NUMBER OF PRECURSORS IDENTIFIED

Nine precursors with a CCDP greater than 1.0×10^{-6}, affecting 11 units, were identified in 1994. The distribution of precursors as a function of the conditional probability [p(cd)] is shown in Table 4. The distribution of 1988–1993 precursors is also shown for comparison purposes.

As described previously, differences in the ASP models and the analysis methods from year to year preclude a direct comparison between the number of events identified for different calendar years. In particular, the CCDPs estimated for the 1992–1994 events are lower than for equivalent events in earlier years because supplemental and plant-specific mitigating systems beyond those included in the pre-1992 ASP models were incorporated into the analyses as a result of licensee review and comments. In addition, new modeling techniques were adopted for the analysis of the 1994 events.

INSIGHTS

A review of the analyses for the nine precursors for 1994 revealed the following trends:

1. As shown in Tables 1 to 3, five of the six events with a CCDP greater that 10^{-5} are pressurized-water reactor (PWR) events. For all 1994 precursors, six

Table 4 Number of Precursors by Year

Year	$10^{-3} \le p(cd) < 1$	$10^{-4} \le p(cd) < 10^{-3}$	$10^{-5} \le p(cd) < 10^{-4}$	$10^{-6} \le p(cd) < 10^{-5}$	Total number of precursors
1988	0	7	14	11	32
1989	0	7	11	12	30
1990	0	6	11	11	28
1991	1	12	8	6	27
1992	0	7	7	13	27
1993	0	4	7	5	16
1994	1	1	4	3	9

were associated with PWRs and three with boiling-water reactors (BWRs).

2. Two events involved at-power initiators in 1994, down from eight in 1993. Six events involving at-power unavailabilities occurred in 1994, compared with eight in 1993.

3. Five of the precursors pertaining to at-power unavailabilities involved the degradation or unavailability of electrical equipment: (1) the degradation of the bus transfer scheme for MCC-5 Haddam Neck, (2) the degradation of the emergency load sequencers at Turkey Point Units 3 and 4, (3) improper breaker settings for an MCC at Dresden Unit 2, (4) the inoperability of both emergency diesel generators (EDGs) at Point Beach Units 1 and 2 (one was removed from service for maintenance; the other had a failed electrical fuel pump and exciter), and (5) zebra mussel shells found in the lube oil and jacket water coolers for one of the EDGs at Zion Unit 2.

4. Four of the six precursors involving unavailabilities occurred at PWRs. One of the precursors involving initiating events occurred at a BWR and the other occurred at a PWR.

5. Six of the nine events (67%) occurred at multiunit sites. This is about the same as the percentage of units at multiunit sites (71%). Two of the precursor events affected both units at a dual-unit site.

A review of the ASP reports for 1990–1994 reveals the following trends:

1. Long-term unavailabilities and LOOP initiators typically dominate the events with the highest CCDPs.

2. The events with the highest CCDPs are dominated by PWRs.

3. The number of precursors identified for 1994 is lower than for previous years. This is due in part to differences in the ASP models for 1994. In addition, the CCDPs estimated for the 1994 events are lower than equivalent events in earlier years because of consideration of supplemental and plant-specific mitigating systems beyond those used in the ASP models. Several events that would have met the precursor criteria for prior years were rejected on low probability following the incorporation of additional mitigating systems in the models.

REFERENCES

1. R. J. Belles, J. W. Cletcher, D. A. Copinger, B. W. Dolan, J. W. Minarick, and L. N. Vanden Heuvel, *Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report*, Main Report and Appendices A–H, Report NUREG/CR-4674 (ORNL/NOAC-232), Vol. 21, December 1995.
2. R. J. Belles, J. W. Cletcher, D. A. Copinger, B. W. Dolan, J. W. Minarick, and L. N. Vanden Heuvel, *Precursors to Potential Severe Core Damage Accidents: 1994, A Status Report*, Appendix I, Report NUREG/CR-4674 (ORNL/NOAC-232), Vol. 22, December 1995.
3. Nuclear Regulatory Commission, *Licensee Event Report System, Description of System and Guidelines for Reporting* (Second Draft Revision of Comment), Report NUREG-1022, February 1994.
4. J. W. Minarick, J. D. Harris, P. N. Austin, J. W. Cletcher, and E. W. Hagen, *Precursors to Severe Core Accidents: 1986—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendices A through C—Vol. 5, Appendices D through F—Vol. 6, May 1988.
5. J. W. Minarick, J. D. Harris, J. W. Cletcher, P. N. Austin, and A. A. Blake, *Precursors to Severe Core Accidents: 1987—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 7, Appendices B through D—Vol. 8, July 1989.
6. J. W. Minarick, J. W. Cletcher, and A. A. Blake, *Precursors to Severe Core Accidents: 1988—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 9, Appendices B and C—Vol. 10, February 1990.
7. J. W. Minarick, J. W. Cletcher, D. A. Copinger, and B. W. Dolan, *Precursors to Severe Core Accidents: 1989—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 11, Appendices B and C—Vol. 12, August 1990.
8. J. W. Minarick, J. W. Cletcher, D. A. Copinger, and B. W. Dolan, *Precursors to Severe Core Accidents: 1990—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 13, Appendices B and C—Vol. 14, August 1991.
9. J. W. Minarick, J. W. Cletcher, D. A. Copinger, and B. W. Dolan, *Precursors to Severe Core Accidents: 1991—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 15, Appendices B, C, and D—Vol. 16, September 1992.
10. D. A. Copinger et al., *Precursors to Severe Core Accidents: 1992—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendix A—Vol. 17, Appendices B–G—Vol. 18, December 1993.
11. L. N. Vanden Heuvel, J. W. Cletcher, D. A. Copinger, J. W. Minarick, and B. W. Dolan, *Precursors to Severe Core Accidents: 1993—A Status Report*, Report NUREG/CR-4674 (ORNL/NOAC-232), Main Report and Appendices A–D—Vol. 19, Appendices E and F—Vol. 20, September 1994.
12. E. M. Dougherty and J. R. Fragola, *Human Reliability Analysis*, John Wiley and Sons, New York, 1988.