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Changes in the U.S. NRC Accident Sequence Precursor program methods since the initial program evaluations of 1969-81 operational events are described, along with insights from the review of 1984-85 events. For 1984-85, (1) the number of significant precursors was consistent with the number observed in 1980-81, (2) dominant sequences associated with significant events were reasonably consistent with PRA estimates for BWRs, but lacked the contribution due to small-break LOCAs previously observed and predicted in PWRs, and (3) the frequency of initiating events and non-recoverable system failures exhibited some reduction compared to 1980-81. Operational events which provide information concerning additional PRA modeling needs are also described.

Keywords: Operational Event, Licensee Event Report, Severe Core Damage Sequence, Precursor.

Introduction

The Accident Sequence Precursor (ASP) program involves the review of Licensee Event Reports (LERs) of operational events that have occurred at U.S. light-water power reactors to identify and rank precursors to severe core damage (inadequate core cooling) accidents. Operational events are selected as precursors if they meet one of the following requirements:

- o The event involved the failure of at least one system required to mitigate a core damage initiator [i.e., a loss of feedwater (LOFW), loss of offsite power (LOOP), small-break loss of coolant accident (LOCA) or steam-line break];
- o The event involved the degradation of more than one system required to mitigate one of the above initiating events; or
- o The event involved an actual initiating event which required safety system response.

Events typically not addressed due to low significance and programmatic constraints include uncomplicated reactor trips, losses of feedwater without additional failures, and single failures in mitigating systems. With the exception of initiating events, precursors typically involve events not considered when applying the single failure criterion used in the design of safety-related systems in the United States.

Initiating event frequency and system failure probability estimates are used to calculate a conditional probability associated with each precursor. The conditional probabilities are estimated by mapping observed failures onto event trees depicting potential paths to severe core damage. This probability is an estimate of the chance of subsequent severe core damage, given that the failures observed during the event occurred as stated, and thus can be considered a measure of the residual protection available during the event. The conditional probabilities associated with each precursor are used to rank precursors as to significance. This ranking is then used to identify the more serious events, which in turn are used to identify the more likely sequences to severe core damage and unusual events not typically addressed in probabilistic risk assessement. An example probability estimate is shown in Figure 1.

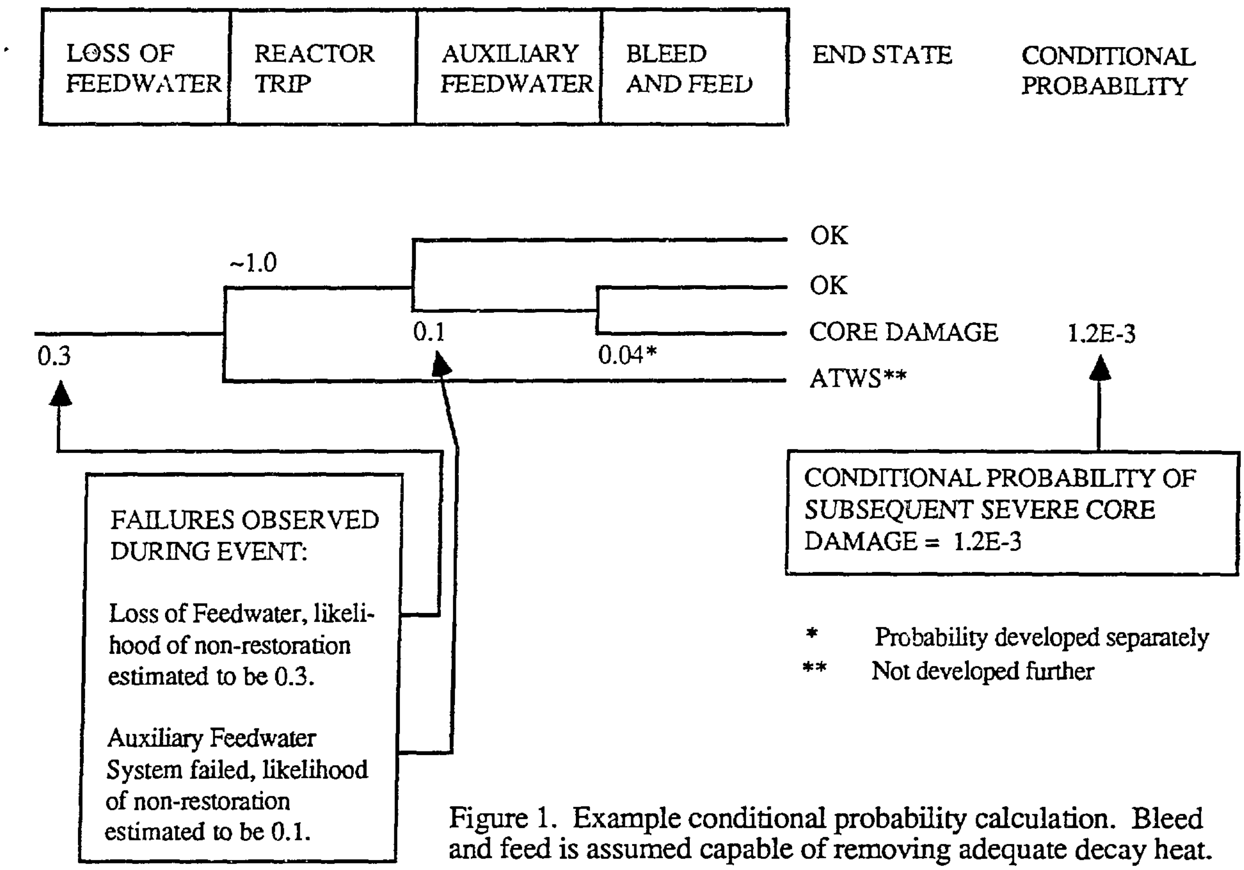

Figure 1. Example conditional probability calculation. Bleed and feed is assumed capable of removing adequate decay heat.

The ASP program is sponsored by the U.S. Nuclear Regulatory Commission and was initiated at Oak Ridge National Laboratory in 1979. Operational events which occurred in 1969-81 were initially reviewed. The results of this review were published in 1982 and 1984 (1-3). Approximately 230 precursors were identified in that time period, with conditional probabilities of core damage ranging from below 1E-9 up to approximately 0.5 for the Three Mile Island accident.

The event tree models used in the 1969-81 effort were recognized at the time to be less than ideal. Simplified, functionally-based event trees had been developed, with systems which served similar functions on different plants were grouped together. When these trees were used to describe events on a particular plant, their branches were tailored to more accurately reflect the plant. The standardized event trees did not completely reflect known differences in how plants respond to initiating events and did not clearly distinguish among less common designs. For example, feedwater coolant injection (FWCI) combined with isolation condensors on older BWRs were modeled under the assumption that they behaved similarly to high pressure coolant injection (HPCI) combined with reactor core isolation cooling (RCIC) in later BWRs.

Since this level of modeling did not distinguish differences in design, it treated all plants of a particular type equally. The potential impact of an observed event was thus evenly distributed across the entire plant data base.

Methods Improvements

In reality, all plants are not equivalent and all events do not equally impact all plants. In 1984, the ASP program, in cooperation with the NRC Accident Sequence Evaluation Program, initiated development of systemic event trees, applicable to plant classes, which would more accurately encompass core damage sequences applicable to all commercial LWRs in the United States (4, 5). Based on the structure of these event trees, four PWR and three BWR plant classes were defined, and computerized transient (including LOFW), LOOP and small-break LOCA event sequence models were developed for each class (6). To distinguish differences among system designs, system models based on redundant "trains" are used in conjunction with the plant-class event sequences to more accurately represent each plant. The system models include the potential for restoration of initial failures and required operator response.

The plant-specific system models are linked to the plant-class event sequence models in a way which preserves grouping of similar designs, as shown in Figure 2. These groupings are necessary because of conflicting modeling objectives necessitated by the limited amount of operational data available, combined with the variety of reactor plant designs that exist in the Unitied States. On the one hand, if modeling is too plant specific, not enough operational data will exist to evaluate events with confidence. The limited amount of operational data places limits on the degree to which one can divide the reactor population. On the other hand, if modeling is too generic, limitations will exist in regard to the appropriateness of what can be discerned from the data. Correct apportionment of operational data will not be possible in this case since data that may be applicable to only a few plants would be distributed across a large number of plants.

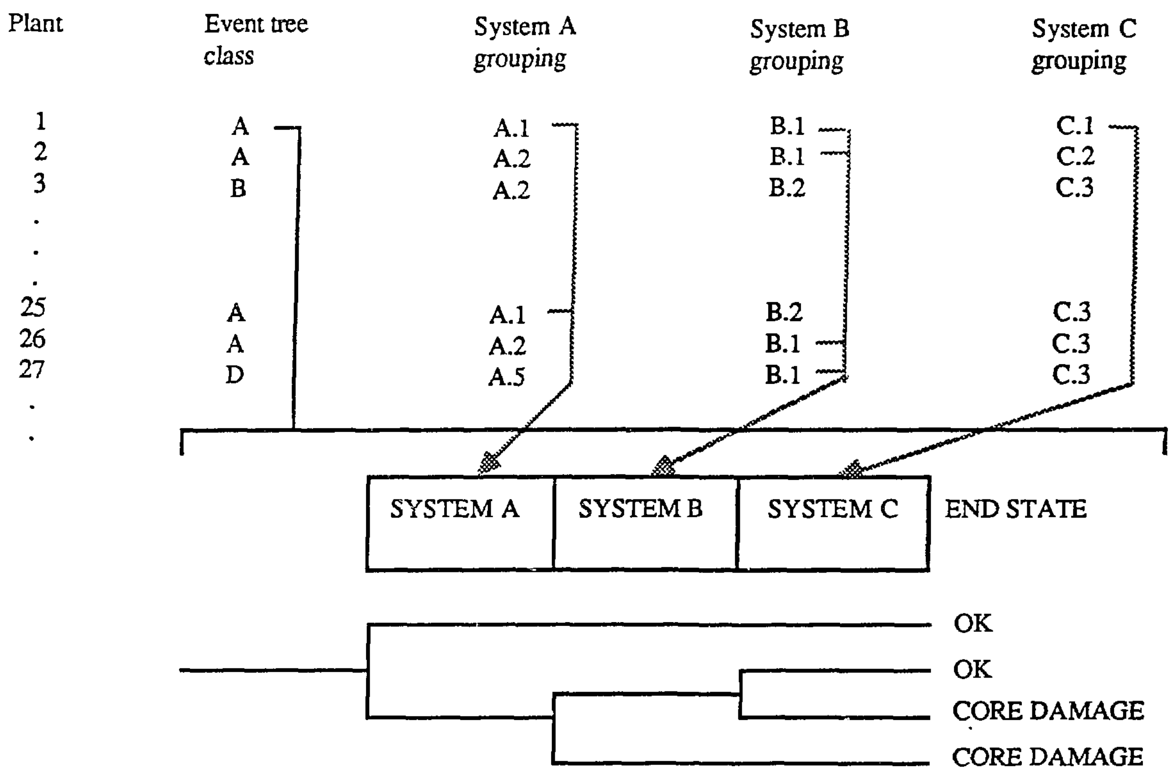

Figure 2. Grouped plant-class event tree and system models for plant 1.

The current plant-class specific event sequence and train-specific system models permit operational data to be apportioned among relevant plants while still permitting a reasonable estimate of precursor significance to be calculated with consideration of the plant design. Implementation of the revised, computer based models has been completed for all U.S. commercial reactor plants critical as of the end of 1985.

Current Results

The first use of the revised models has been in the identification and analysis of 1984-85 precursors. Preliminary insights drawn from these events concern precursor frequency, dominant sequences, initiating event frequencies, system failure probabilities, and the identification of events which are questionably addressed in contemporary PRAs.

Precursor Frequency

One hundred eleven precursors were identified (~0.6/reactor year) compared with 230 (~0.4/reactor year) in 1969-81. The number of LERs selected for detailed review for precursors was also substantially greater than in 1969-81, although fewer LERs were reported.

The revised LER rule, which went into effect in the United States in 1984, plus the current review of all reactor trip events by the ASP program are believed to be the primary causes for the increased precursor frequency. One requirement of the revised LER rule is the detailed reporting of all operational events involving reactor trip. Because of this, LOFW events are now reported, and single failures that occur following trip can be clearly tied to reactor trip events. Both of these types of events are now identified and analyzed. (LOFW without additional failures are quantified on a representative basis for the plant classes defined.)

Twenty seven events with estimated conditional probabilities greater than 1E-4 were observed in 1984-85. On a per reactor year basis, this number of events is essentially the same as in 1980-81. However, the number of events with conditional probabilities greater than 1E-3 was only half that observed in 1980-81. This difference is not attributable to the use of revised event sequence models but results from the data itself. The number of precursors involving BWRs was substantially greater than expected based on the number of BWRs in the U.S. reactor population, if an equal likelihood of precursors per reactor year in BWRs and PWRs is assumed.

Likely Scenarios and Sequences

Precursors with estimated conditional probabilities greater than 1E-4 were reviewed to qualitatively identify likely core-damage sequences associated with these events. These sequences include the observed plant state plus the postulated failures, beyond the operational event, required for core damage. For 1984-85 precursors, these sequences can be categorized as:

- o Failure of secondary-side cooling and feed and bleed, plus failure to successfully initiate condensate cooling following steam generator depressurization (89% of PWR sequences),
- o Station blackout (11% of PWR sequences),
- o Failure of all high-pressure cooling and failure to depressurize following transients and small LOCAs (67% of BWR sequences),
- o Failure of long-term decay heat removal following a transient (17% of BWR sequence), and
- o Failure of high-pressure cooling following a LOOP plus unavailability of emergency power for low-pressure core cooling (11% of BWR sequences).

Comparing these sequences with the relative contribution of dominant sequences developed from PRAs (7,8,9), reasonable agreement appears to exist for BWRs. However, for PWRs, sequences involving small LOCAs that contributed substantially to the PRA estimates (and which were observed in 1969-81 precursors) were not seen in 1984-85. Sequences associated with loss of cooling water systems, which are important contributors to core damage in two of the five PRAs documented in Ref. 10, are infrequently observed in the precursors.

Initating Event Frequencies and System Failure Probabilities

Operational events involving LOOP, small LOCA, and certain system failures have been used in the ASP program to estimate average initiating event frequencies and system failure probabilities. This estimation recognizes that many failures are recoverable in a 20-30 minute time period, and attempts to account for this by assigning a non-recovery likelihood to each failure. The estimate of non-recovery likelihood is based on engineering judgement. The total number of non-recoverable failures is then used, along with an estimate of the observation period (for initiators) or number of demands (for systems) to calculate an average frequency or failure probability. The approximate number of such non-recoverable events observed in 1984-85 is shown in Table 1, along with the expected number of events based on 1969-81 data.

Table 1. Observed non-recoverable 1984-85 initiating events and demand-related system failures

Initiating event or system failure*	Expected events in 1984-85 based on 1969-81 data**	Observed number of events in 1984-85
BWR LOOP	1.1	1.6
BWR small-break LOCA	1.2	1.0
BWR failure of HPCI and RCIC	1.0	0.7
BWR failure of emergency power	1.6	0.3
BWR failure of automatic depressurization	0.4	0.0
BWR failure of reactor vessel isolation	0.2	1.0
BWR failure of long-term core cooling	0.1	2.3
PWR LOOP	3.1	3.2
PWR small-break LOCA	1.0	0.0
PWR failure of auxiliary feedwater	0.8	0.4
PWR failure of high pressure injection	0.8	2.0
PWR failure of long-term core cooling	0.5	0.0
PWR failure of emergency power	0.5	0.0
PWR failure of steam generator isolation	0.9	~0.0

* System failure probabilities independent of initiating event.
** Based on the total reactor years in 1984-85, consistent with NUREG/CR-3591 (2).

As can be seen in Table 1, most categories of initiators and system failures exhibit a smaller than expected number of event in 1984-85 compared to 1969-81. The number of categories exhibiting a decrease is not, however, sufficient to draw statistically-based conclusions concerning inprovement since the 1969-81 time period (11).

PRA Modeling Insights

Most precursor events or event combinations had likelihood and occurrence characteristics similar that are typically modeled or probabilistically estimated by PRAs. However, a small but important number of precursors occured in a manner that would not normally be evaluated or considered in PRAs. These precursors involved phenomenological or situation-specific scenarios including initiating events which were strongly influenced by operator error, a maintenance error which resulted in an overpressurization not usually modeled in BWR PRAs, and events involving system interactions.

- At Davis-Besse (LER 50-346/85-013), a loss of main feedwater and unavailability of auxiliary feedwater was initiated by a spurious control system trip. The event was complicated by an operator error, which resulted in AFW isolation. Independant of the AFW isolation, both AFW pumps (which are turbine driven on this plant), tripped on overspeed and both AFW isolation valves initially failed to reopen after being closed. The pressurizer power-operated relief valve (PORV) stuck open after several actuations and was manually isolated. The operators placed the startup feed pump in service and then restored the AFW system.
- At Susquehanna 2 (LER 50-388/84-013), a station blackout condition existed during a Loss of Turbine Generator and LOOP Startup Test. Following initiation of the test, the four diesel generators failed to start because of a common-mode error made when racking out the feeder breakers to the 4KV buses in preparation for the test. When initial attempts at reenergizing the 4KV buses did not succeed, the emergency plan was entered; power was restored to the buses through manual closure of the breakers to the offsite source. Discrepancies between in-plant switch labeling and the terminology used in the test procedure contributed to the error.
- At Hatch 1 (LER 321/85-018), the inadvertant actuation of a filter train fire deluge system caused by damage to an associated pressure gauge, combined with partially plugged drains in the filter

train, resulted in water in the control room ventilation ducting. This water dripped onto an electrical panel, and opened a safety relief valve, which, after three lifts, stuck open. RCIC was out of service for maintenance, and HPCI was inoperable during part of the event. Feedwater was available throughout the event.

- At San Onofre 1 (LER 50-206/85-017), following a loss of power to the safety-related buses, five failed check valves in the main feed system prevented AFW flow to the steam generators. Because of the failed valves, cold AFW water flowed toward the main feed pumps and caused a water hammer that damaged a feedwater line and a check valve, and resulted in a nonisolable leak. The event was recovered when the operators closed the main feedwater regulating valves as part of the normal post-trip response, isolating the main feedwater system from the steam generators.
- At Browns Ferry 1 (LER 50-259/84-032), low pressure core spray system piping was pressurized to near reactor pressure through a series of errors involving failure to secure power to a motor-operated valve and the backwards installation of the air operator on a testable check valve eight months earlier.

Notes

1. J.W. Minarick and C.A. Kukielka, Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, NUREG/CR-2497, June 1982.

2. W.B. Cottrell, J.W. Minarick, P.N. Austin, E.W. Hagen and J.D. Harris, Precursors to Potential Severe Core Damage Accidents: 1980-1981, A Status Report, NUREG/CR-3591, July 1984. This document provides a more detailed discussion of the modeling methods used in the earlier program efforts.

3. D.L. Phung, J.W. Minarick and J.D. Harris, A Review of Comments on the 1969-1979 Accident Sequence Precursor Report: NUREG/CR-2497, ORNL/NRC/LTR-85/14, June 1985. This document also provides a ranking of 1969-79 events on a consistent basis with 1980-81 events.

4. M. Modarres, E. Lois, and P. Amico, LWR Categorization Report, University of Maryland, internal ASP program report, November 13, 1984.

5. Accident Sequence Evaluation Program: Event Tree Development, Sandia National Laboratory, internal ASEP program report, August 1985.

6. The event sequences do not address containment response except as required for short-term core cooling. Except for emergency power following LOOP, support systems are also not specifically addressed. The impact of support system unavailabilities is addressed through their impact on front-line systems. Certain unique-design plants, such as San Onofre 1 and LaCross, are separately modeled.

7. J. Young and S. Asselin, Perceptions of LWR Risk for Decision Making, presented at the Eleventh Water Reactor Safety Research Meeting, Gaithersburg, Md., October 28, 1983.

8. V. Joksimovich, M.V. Frank, and D.R. Worledge, Dominant Accident Sequences Derived From Review of Five PRA Studies, presented at the American Nuclear Society Meeting on Anticipated and Abnormal Plant Transients in Light Water Reactors, Jackson, Wyoming, September 26-29, 1983.

9. Page 8-8 of NUREG/CR-3591 compares the results of the two previous reports to the dominant sequences observed in 1980-81.

10. Reactor Risk Reference Document, NUREG-1150, January 1987, Vol. 1 (draft), p. ES-7.

11. This conclusion can be drawn for PWR events in 1980-81 compared to 1969-79; see NUREG/CR-3591, pp 5-2 - 5-5 and 8-2 - 8-3.

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