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Draft Report

OPERATING EXPERIENCE ASSESSMENT
OBSERVATIONS REGARDING ACCIDENT SEQUENCE PRECURSOR EVENTS
FY 1993–2000

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Abstract

SECY-02-0041, "Status of Accident Sequence Precursor and Spar Model Development Programs," March 8, 2002, includes a table which lists accident sequence precursor (ASP) events from 1993 to 2000 not typically modeled in probabilistic risk assessments (PRAs) or individual plant examinations. Based on that table, the CCDP values of all ASP events in the ranges E-3, E-4, and E-5 for the years 1993 to 2000 were grouped and summed; the same was done for those events listed as not typically modeled in PRAs. Although only 5 of 54 were not modeled, 42 percent of the cumulative CCDP was due to the events not typically modeled in PRAs. The groupings according CCDP values also showed that the fraction not typically modeled in PRAs increased for the group with higher CCDPs. These results show that reliance on regulatory tools developed from current PRAs could miss a significant fraction of the actual risk and that defense in depth design and plant oversight activities which go beyond risk-based tools need to be maintained.

Introduction

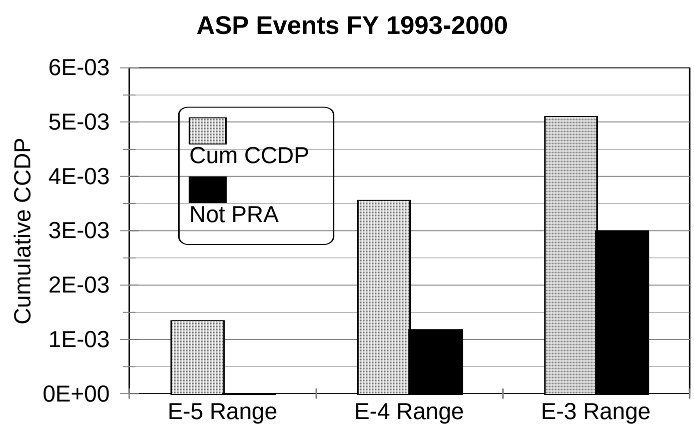
As part of the Office of Nuclear Regulatory Research activity associated with regulatory effectiveness and independent review of operating experience, Regulatory Effectiveness Assessment and Human Factors Branch staff developed some observations based on the information in SECY-02-0041, "Status of Accident Sequence Precursor and Spar Model Development Programs," March 8, 2002. Table 6 of SECY-02-0041 lists the precursor events not typically modeled in probabilistic risk assessments (PRAs) or individual plant examinations (IPEs). Table 6 covers accident sequence precursor (ASP) events for fiscal years 1993 to 2000 and identifies 8 events or conditions, some affecting multiple units, for a total number of 14. Five of the 14 were calculated to have conditional core damage probabilities (CCDPs) > E-5.

Accident Sequence Precursor Data Plots

We collected the CCDP values of all ASP events with CCDP > E-5 for FY 1993 to 2000 in Table 1. We grouped the events within CCDP ranges of E-5, E-4, and E-3, and added the CCDPs of events in each group. We did the same for all events not typically modeled in PRAs. The results are presented in Figure 1.

We also plotted the CCDP values in rank order in Figure 2, distinguishing between PRA and non-PRA events. The results show that a few high CCDPs contribute more of the cumulative CCDP than the sum of a large number of lower CCDP events. Figure 2 also shows that five ASP events or conditions not typically modeled in PRAs are among the top CCDPs. The ASP program looks at many events which are not quantified – this discussion only addresses those with CCDP > E-5.

Figure 1 - **CCDP Not Typically Modeled in PRA**



The results show that for ASP events from 1993 to 2000 with CCDP > E-5:

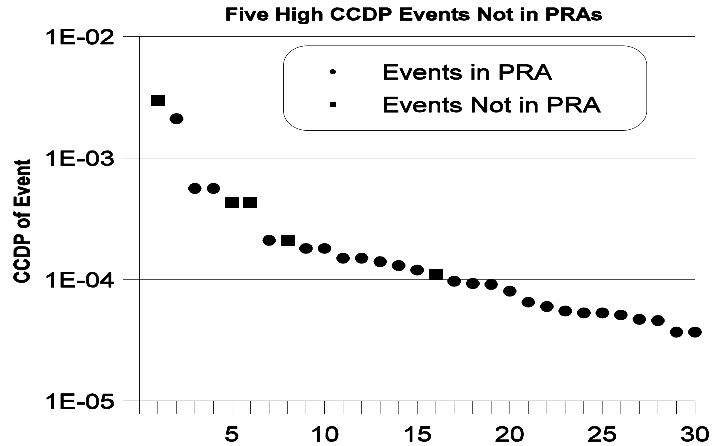
(1) approximately 42 percent of the cumulative CCDP from ASP events is not typically modeled in PRAs, despite the fact that they represent only 9 percent (5/54) of the number of ASP events with CCDP > E-5; and

(2) over 59 percent of the cumulative CCDP in the E-3 range was not typically modeled in PRAs - the fraction of the cumulative CCDP not modeled in typical PRAs increased for the higher CCDP ranges.

As seen in Figure 2, cumulative CCDP is dominated by a small number of high CCDP events. The E-3 group includes only 2 events – the cumulative CCDP of those 2 events exceeds the cumulative CCDP of the 14 events in the E-4 group.

The highest CCDP is the Wolf Creek blow-down event. During that event, while in hot shutdown, operator errors resulted in a loss of primary coolant via a flow path which by-passed containment and could have disabled the emergency core cooling system (ECCS) pumps which are used to restore reactor coolant. That event was a shutdown event, and PRAs do not typically model shutdown events.

Figure 2 - ASP Events Rank Ordered by CCDP



NRC Review Activities

One other observation regarding the Wolf Creek event is pertinent. For 2 months following the event, neither the licensee nor the NRC recognized the significance of the event – the licensee did not plan to issue an licensee event report. The region issued a morning report which provided a basic description of the event. The potential significance was recognized by a manager in headquarters with the authority to initiate a more thorough assessment of the event, which resulted in recognition of the event's significance. Wolf Creek involved a loss-of-coolant accident, potentially inoperable ECCS, containment bypass, human error, and limited time for operator response. If this precursor had not been identified, about one-third of the cumulative CCDP for the period 1993 to 2000 would have been missed.

This highlights the importance of system reviews, inspection, and oversight in ensuring that potentially significant events and conditions (even if not counted as such in risk models) are recognized and addressed. Precursors such as Wolf Creek, while not included in current PRAs, can and should be quantified when the details of the event are understood. Failure to identify such precursors can result in significant underestimation of risk and unwarranted confidence regarding nuclear safety performance.

The recent Davis-Besse reactor vessel head corrosion event is another example of a risk significant condition not considered in current PRAs. Boric acid ate away carbon steel leaving only the thin layer of stainless cladding intact.

The recent condition at Point Beach involving failure of the auxiliary feedwater (AFW) system on loss of air which existed for the life of the plant was also not modeled in the plant PRA. Although AFW failure modes are typically modeled, the plant-specific unrecognized system interaction with the air system failure leading to failure of both AFW trains was not modeled. The system dependency was undetected despite a review of the Point Beach AFW system by the licensee and the NRC following the Three Mile Island event.

The NRC has adopted a regulatory approach which focuses review and oversight activities on equipment in high risk sequences developed from existing plant PRAs. For situations where high risk event sequences have been identified in a PRA, licensee or regulatory action has been taken to limit the risk to a small fraction of the total risk – if the risk is recognized and measured, it will be addressed.

It is understandable that some events not represented in PRAs occur with high CCDPs. Nuclear plants are complex machines which are difficult to model; uncertainties exist in PRA results due to modeling, equipment, and operator performance inaccuracies. Sequences which are not recognized or are eliminated do not show a risk contribution. The PRA is a normative tool for those sequences it includes, but has no impact on the risk profile of those not included. Or stated differently: if the risk is not measured, it is less likely to be addressed than if it is measured.

As a consequence, NRC system reviewers and inspectors must continue to address events or conditions which: reduce defense in depth; manifest previously unrecognized common mode failure mechanisms or system interactions; invalidate the assumptions of the plant PRA; or are not modeled in the plant PRA.

Overall Observations

We believe the ASP results suggest the following general observations:

(1) the ASP program is of unique value because it provides a quantitative measure (CCDP) of the significance of events not typically modeled in current PRAs or IPEs; over the past 8 years, those events accounted for over 40 percent of the cumulative CCDP;

(2) the current NRC policy of risk-informed, performance-based (rather than risk-based) needs to account for the fact that a significant fraction of CCDP is not typically modeled in PRAs;

(3) defense in depth continues to be an important concept in nuclear power safety to compensate for the potential incompleteness and uncertainties in PRAs; and

(4) lessons learned from events and conditions found in operating experience should be incorporated into PRAs and IPEs as part of updating of risk models and addressed in PRAs of new plants in the design and planning stages.

Table 1 – ASP Events 1993-2000, CCDP > E-5, Ranked by CCDP Value (some titles truncated)

Event Date	Plant Name	CCDP/CDP	NO PRA	Precursor Title
09/17/94	Wolf Creek 1	3.0e-03	x	RCS blows down to RWST (9,200 gal) during hot shutdown
02/06/96	Catawba 2	2.1e-03		LOOP with EDG B unavail
05/19/96	Arkansas 1	5.6e-04		Reactor trip and subsequent steam generator dryout
06/24/98	Davis-Besse	5.6e-04		A tornado touchdown causes reactor trip and LOOP
10/22/99	Cook 1	4.3e-04	x	Potential high-energy line break conditions affecting the operability of
10/22/99	Cook 2	4.3e-04	x	Potential high-energy line break conditions affecting the operability of
10/19/92	Oconee 2	2.1e-04		Loop with failure of both Keowee units
01/30/96	Wolf Creek 1	2.1e-04	x	Reactor trip with a loss of train a of essential service water and
12/31/92	Sequoyah 1	1.8e-04		Loss of offsite power
12/31/92	Sequoyah 2	1.8e-04		Loss of offsite power
02/25/93	Catawba 1	1.5e-04		Essential service water potentially unavail
02/25/93	Catawba 2	1.5e-04		Essential service water potentially unavail
02/16/94	Haddam	1.4e-04		Both pressurizer PORVs and vital 480 vac bus degraded
09/14/93	LaSalle 1	1.3e-04		Scram and LOOP
03/26/93	Perry 1	1.2e-04		Clogged suppression pool strainers and service water flood
08/01/96	Haddam	1.1e-04	x	Potentially inadequate RHR pump NPSH following a large-or medium break
05/15/00	Diablo	9.7e-05		Extended loss of offsite power to safety-related buses due to 12-kV bus
01/09/95	St Lucie 1	9.3e-05		Failed pressurizer PORVs, RCP seal failure, relief valve failure, plus other
12/27/93	McGuire 2	9.3e-05		Loop and failure of an MSIV to close
02/15/00	Indian Point 2	8.0e-05		Manual reactor trip following a steam generator tube failure
06/27/93	Haddam	6.5e-05		Degradation of MCC-5, pressurizer PORV, both EDGs
04/22/93	Quad Cities 2	6.0e-05		Degradation of both EDGs
10/12/93	Beaver Valley	5.5e-05		Dual-unit LOOP
06/29/96	Prairie Island	5.3e-05		Loop to safeguards buses on both units
06/29/96	Prairie Island	5.3e-05		Loop to safeguards buses on both units
09/30/93	Arkansas 1	5.1e-05		Both trains of recirculation inoperable for 14 h
03/14/93	Palo Verde 2	4.7e-05		Steam generator tube rupture (240 gpm)
05/21/96	Seabrook 1	4.6e-05		Turbine-driven EFW unavail
12/30/99	Cook 1	3.7e-05		Valves required to operate post-accident could fail to open due to
12/30/99	Cook 2	3.7e-05		Valves required to operate post-accident could fail to open due to
06/11/99	Cook 1	3.2e-05		Lack of capability to operate emergency service water following a seismic
12/02/92	Oconee 3	3.2e-05		Both Keowee units potentially unavailable
12/02/92	Oconee 2	3.2e-05		Both Keowee units potentially unavailable
06/11/99	Cook 2	3.2e-05		Lack of capability to operate emergency service water following a seismic
12/02/92	Oconee 1	3.2e-05		Both Keowee units potentially unavailable
01/25/95	Millstone Pt 2	3.1e-05		Containment sump isolation valves potentially unavailable due to pressure
06/11/95	Comanche Pk	2.9e-05		Reactor trip, AFW pump trip, second AFW pump unavail
03/07/94	Zion 2	2.3e-05		Unavail of turbine-driven AFW pump and an EDG
04/21/97	Oconee 2	2.2e-05		Unisolable RCS leak
04/20/95	Arkansas 1	2.0e-05		Reactor trip with one EFW train unavailable
09/08/94	River Bend 1	1.8e-05		Scram, main turbine-generator fails to trip, RCIC and control rod drive
06/10/95	Waterford 3	1.7e-05		Reactor trip, breaker failure and fire, and degraded shutdown cooling
11/02/97	St Lucie 1	1.7e-05		Non-conservative recirculation actuation setpoint
05/23/96	Byron 1	1.7e-05		Transformer bus fault causes a LOOP during shutdown
10/14/98	Davis-Besse	1.4e-05		Manual reactor trip due to component cooling system leak and
01/12/94	Calvert Cliff 2	1.3e-05		Trip, loss of 13.8-kv bus, and salt water cooling system unavail for 2 min
10/17/92	Callaway	1.3e-05		Loss of main control room annunciators
11/20/95	St Lucie 2	1.3e-05		Failure of one EDG with CCF implications
01/22/97	Maine	1.3e-05		RCS hot-leg recirc vlv subject to pressure lock
01/22/93	South Texas	1.2e-05		Unavail of one emergency diesel generator and the turbine driven auxiliary
02/08/94	Point Beach 1	1.2e-05		Both EDGs inoperable for 47 hrs
02/08/94	Point Beach 2	1.2e-05		Both EDGs inoperable for 47 hrs
01/19/95	Arkansas 2	1.1e-05		Loss of dc bus could fail both EFW trains
07/14/98	Big Rock	1.1e-05		Long-term unavailability of liquid poison control system