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Proposed Examples for ROP Tabletop
Significance Determination Process (SDP) Examples for New Reactors

1. **Failure of high-pressure core flood (HPCF) pump along with a bounding case of common-cause failure (CCF) of both HPCF pumps.**

   Related SDP Case – Perry experienced a high-pressure core spray (HPCS) pump failure in 2002 due to the failure of contacts in pump circuit breaker. This resulted in a WHITE finding (EA-03-007; Internal Events ΔCDF = 5E-6). The HPCS system was unavailable from August 28 to October 23, 2002, the time from last successful surveillance until time of discovery. However, the plant was in an outage during this period (September 23 through October 3). The Phase 3 analysis used a T/2 approach and considered the HPCS system to be unavailable for the total time period minus the plant outage time divided by 2; therefore, the exposure period = [(56 days - 10 days) / 2] = 23 days.

   Applicable Plant(s) – Advanced Boiling-Water Reactor (ABWR)

   SPAR Model Used– ABWR

2. **Failure of the turbine-driven emergency feedwater (EFW) pump along with a bounding case of CCF of both turbine-driven EFW pumps.**

   Related SDP Case – Calvert Cliffs, Unit 1 experienced a turbine-driven auxiliary feedwater (AFW) pump failure in 2001 due to the failure of pump bearing (excessive sealant). This resulted in a YELLOW finding (EA-01-206; Internal Events ΔCDF = 8E-6, the licensee's external events contribution increased the ΔCDF to >1E-5). By SDP rules, the exposure period was limited to 1 year.

   Applicable Plant(s) – Advanced Pressurized-Water Reactor (US-APWR)

   SPAR Model Used– US-APWR (completion expected in September 2011). Request Mitsubishi Heavy Industries, LTD (MHI) to run this case with their external events models. A contingency would be for the staff to perform a hand calculation using RAW values from the US-APWR PRA (e.g., Tables 19.1-30 to 37).

3. **Failure of suction source for reactor core isolation cooling (RCIC) and/or HPCF systems.**

   Related SDP Case– At Clinton in 2005, NRC inspectors determined that HPCS and RCIC may fail due to air entrainment during the switch-over from the RCIC tank to the suppression pool (due to the setpoint for switch-over was set too low). It was later determined that only the HPCS system would be negatively impacted. This resulted in a WHITE finding (EA-06-291; Internal Events ΔCDF = 6E-6). By SDP rules, the exposure period was limited to 1 year.

   * *Despite only causing the unavailability of HPCS is this SDP case, this test case will assume that RCIC would have been the system affected. We will also perform a bounding case of loss of suction source for both RCIC and HPCF.*

   Applicable Plant(s) – ABWR

SPAR Model Used – ABWR

4. **Suction source failure of EFW system leading to common-mode failure of all EFW pumps.**

   Related SDP Cases – (1) Callaway experienced a condensate storage tank (CST) diaphragm degradation that caused on demand failure of an AFW pump with the potential to cause failure of the other AFW pumps. This resulted in a WHITE finding (EA-02-046; Internal Events ΔCDF = 4E-6).

   (2) Comanche Peak, Unit 1 experienced a degraded CST bladder that could have led to the potential unavailability of the AFW pump(s) in 2010. This resulted in a GREEN finding (EA-10-144; Internal Events ΔCDF = 9.8E-7).

   Applicable Plant(s) – US-APWR and Evolutionary Pressurized-Water Reactor (EPR)

   SPAR Model Used– US-APWR (completion expected in September 2011). Request Areva (EPR) and MHI (US-APWR) to perform analysis including external events based on a similar (although not necessarily identical CCF clogging of one EFW tank/pit. A contingency would be for the staff to use importance measures from the DCDs Chapter 19 to do a hand calculation of the impact.

5. **Disk-stem motor-operated valve (MOV) failure on the in-containment refueling water storage tank (IRWST) drain line isolation MOV (PXS-V121A or B).**

   Related SDP Case– This is a hypothetical case. However, disk-stem MOV failures have occurred at NPPs. For example, Browns Ferry, Unit experienced a low-pressure coolant injection (LPCI) suction MOV failure that occurred in 2010. This resulted in a RED finding [EA-11-018; Internal Events ΔCDF = 1E-6). The final SDP result was greatly influenced by self-induced SBO fire procedures. The exposure period used in the Phase 3 analysis is 295 days [T/2 of 589 days (March 13, 2009 to October 23, 2010)].

   Applicable Plant(s)– AP1000

   SPAR Model Used– AP1000. Request that Westinghouse identify realistic fault exposure time based on expected operational cycle & IST that might identify this failure mode.

Management Directive (MD) 8.3 Examples for New Reactors

1. **Loss of offsite power with a demand failure of an emergency diesel generator (EDG).**

   ***Related MD 8.3 Case*–** On September 15, 2003, Peach Bottom, Units 2 and 3 experienced a brief loss of offsite power (LOOP) to the emergency buses. The loss of offsite power resulted in the loss of power to the reactor protection system (RPS) motor generator sets which automatically shut down Unit 2 and 3 and automatically initiated Primary Containment Isolation System (PCIS) Group I isolation causing the main steam isolation valves (MSIVs) to shut. All four EDGs automatically started; however, EDG E2 tripped on low jacket water coolant pressure approximately 1 hour after the LOOP occurred. In addition, Safety Relief Valve (SRV) D initially failed to reclose after lifting. The valve closed 15 minutes later with no operator action.

   The preliminary risk assessment in accordance with MD 8.3 was conducted for Unit 3 using the Peach Bottom Unit 2 and 3 SPAR Model, Revision 3.02, dated January 2003, with no test and maintenance unavailability included. The assessment assumed a Unit 3 plant-centered LOOP, MSIV closure, a single stuck open relief valve, and EDG E2 failure to run. This resulted in conditional core damage probability (CCDP) of low E-3. The Unit 2 CCDP for a Plant Centered LOOP, MSIV closure, and E2 EDG failure to run was low E-4. Based on these results and using the guidance in MD 8.3, the NRC determined that an Augmented Inspection Team (AIT) should be sent to the site.

   ***Applicable Plant(s)*–** ABWR

   ***SPAR Model Used*–** ABWR

2. **Loss of offsite power.**

   ***Related MD 8.3 Cases*–** (1) On May 20, 2006, an electrical fault in the Catawba 230kV switchyard caused several power circuit breakers to open resulting in a LOOP and a subsequent reactor trip of both units.er. Following the LOOP, the four EDGs started and supplied power to the 4.16kV vital busses.

   In accordance with MD 8.3, the NRC concluded that the circumstances of the event met the MD 8.3 deterministic criteria due to an apparent single electrical failure causing a loss of offsite power to both operating units and reactor trips. The risk review indicated the CCDP (1.8E-4) for the event met the criterion for an AIT.

   (2) On June 14, 2004, at Palo Verde Nuclear Generating Station, a ground-fault occurred on a 230 kV transmission line approximately 47 miles from the site. A failure in the protective relaying resulted in the ground fault not isolating from the local grid for approximately 38 seconds. This uninterrupted fault cascaded into the protective tripping of a number of 230 kV and 500 kV transmission lines, a nearly concurrent trip of all three units within approximately 30 seconds of fault initiation. The Unit 2 Train ‘A’ EDG started, but failed early in the load sequence. This resulted in the Train ‘A’ Safety Buses de-energizing.

   The NRC evaluated this event using the SPAR Model for Palo Verde 1, 2, and 3, Revision 3, and modified appropriate basic events to include updated LOOP curves published in NUREG CR-5496, "Evaluation of Loss of offsite power Events at Nuclear Power Plants:

1980 - 1996." The staff evaluated the risk associated with the Unit 2 reactor because it represented the dominant risk of the event. For the, the team established that a LOOP had occurred and that the event may have been recovered at a rate equivalent to the industry average. Both EDG A and Charging Pump E were determined to have failed and assumed to be unrecoverable. In addition, the team ignored all sequences that included a failure of operators to trip reactor coolant pumps, because all pumps trip automatically on a LOOP. The conditional core damage probability was estimated to be 7E-4 indicating that the event was of substantial risk significance and warranted an AIT.

Applicable Plant(s) – Case 1– US-APWR and AP1000. Case 2– EPR and US-APWR. For the US-APWR we would apply the EDG failure as a failure of the combustion turbine generator (CTG). Request Areva to perform the analysis for the EPR.

SPAR Model Used – US-APWR (completion expected in September 2011)

3. **Steam generator tube failure/rupture.**

Related MD 8.3 Case – In early February 2000, at Indian Point, Unit 2, primary-to-secondary tube leakage (ranging from one to four gallons per day) was detected in Steam Generator (SG) 24. On February 15, 2000, while the unit was operating at 99% power, SG leakage rapidly increased to greater than 75 gallons per minute (gpm). The reactor was manually tripped 13 minutes later, and the faulted steam generator was isolated one hour after the reactor trip. In addition to shutting down the reactor and isolating the affected steam generator, the plant operators also took appropriate action to cool down and depressurize the reactor coolant system to prevent leakage into the faulted steam generator. The highest leak rate which was observed during the event (about 146 gpm) occurred prior to the reactor trip.

The NRC used Revision 2QA SPAR Model for Indian Point, Unit 2. The preliminary event CCDP was calculated to be 3.3E-04. The licensee's initial analysis indicated a CCDP of 7.7E-05. Based on the initial risk estimates and deterministic criteria in MD 8.3, an AIT was performed.

Applicable Plant(s) – AP1000

SPAR Model Used – AP1000

Mitigating Systems Performance Index (MSPI) Examples for New Reactors

1. **Brunswick, Unit 1, Emergency AC Power (EAC) WHITE (2006, 2nd Quarter).** It appears that two EDG failures caused the 1E-6 GREEN/WHITE threshold to be crossed, and a third failure caused the MSPI to reach as high as 5.1E-6; for simplicity, this will be modeled as three EDG start failures using the ABWR SPAR model. The unavailability index (UAI) will be set to zero for simplicity. The number of start failures will be increased until the PLE is reached and the 1E-6 threshold is crossed, to understand the margin.

2. **South Texas, Unit 2, EAC WHITE (2006, 2nd Quarter).** 2701 hours of EDG unavailability contributed to the MSPI exceeding 1E-6. This will be modeled in the EPR. It will be assumed that this is entirely unplanned unavailability and that planned unavailability contributes zero. Separately, to understand the margin, the number of failures to MSPI > 1E-6 and to reach the PLE limit with UAI=0 will be assessed. If time permits, similar calculations will be performed on the emergency CTGs for the US-APWR.

3. **Ginna, Heat Removal WHITE (2009, 3rd Quarter).** Three turbine-driven AFW pump failures will be used on the US-APWR. UAI will be set to zero. To understand the margin, the number of failures to MSPI >1E-6 and to reach the PLE will be calculated.