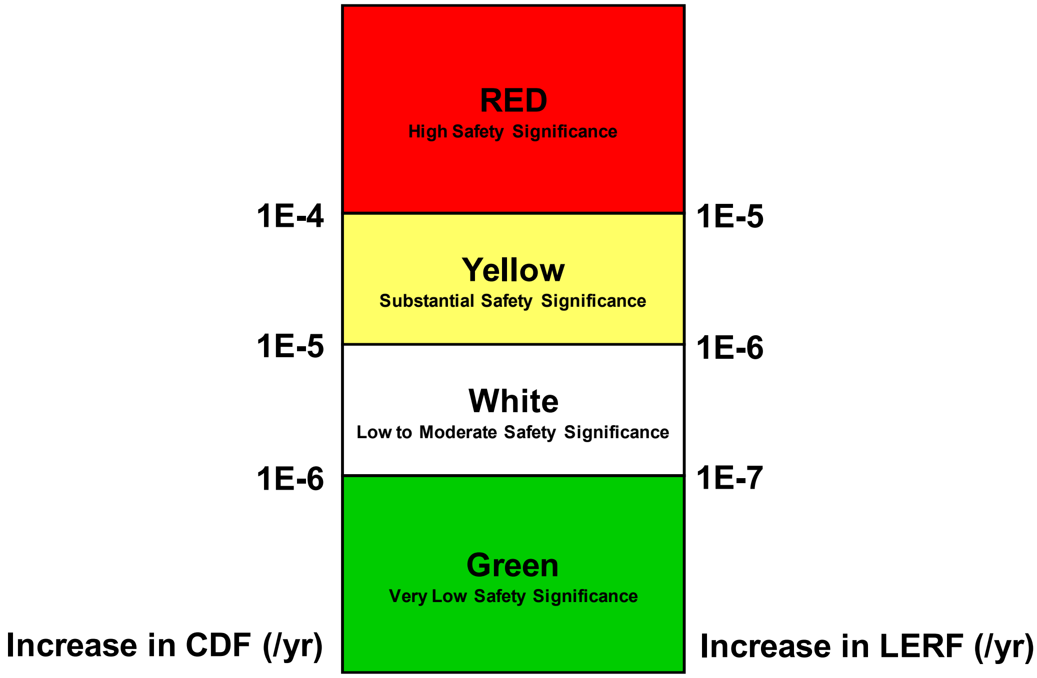
# Examples and Results for ROP Tabletop

*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)

   ***Results–*** The results presented are for internal events only (excluding internal fires and flooding).

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| **ABWR** | HPCF pump fails. | 23 days | 1.4E-8 | **SPAR Model–** Pump B FTS basic event set to TRUE. |
| | | 1 year | 2.2E-7 | |
| | Both HPCF pumps fail due to common-cause. | 23 days | 4.8E-8 | **SPAR Model–** Pumps CCF FTS basic event set to TRUE. |
| | | 1 year | 7.7E-7 | |

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)

***Results–*** The results presented are for internal events, including internal fires and flooding, where indicated.

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| US-APWR | One TDEFW pump fails. | 1 year | 2.2E-5 | **SPAR Model–** Pump A FTS basic event set to TRUE.[1] |
| | | 1 year | 3.4E-6 | **Importance Measure– PRA** [RAW = 4.4E+0]; internal events (excluding internal fire and floods). |
| | | 1 year | 3.4E-6 | **MHI PRA (Internal Fire and Flooding)–** Basic event representing failure to start of EFW Pump A (for fire)/Pump D (for flood) is set to TRUE. **Fire ΔCDF =** 2.2E-6 and **Flooding ΔCDF =** 1.2E-6. |
| | Both TDEFW pumps fail due to common-cause. | 1 year | 4.4E-4 | **SPAR Model–** Pump A and D CCF FTS basic event set to TRUE. |
| | | 1 year | 3.4E-5 | **Importance Measure– PRA** [RAW = 3.5+1]; internal events (excluding internal fire and floods). |
| | | 1 year | 8.8E-6 | **MHI PRA (Internal Fire and Flooding)–** Basic events representing failure to of EFW pumps A and D are set to TRUE. [**Fire ΔCDF =** 7.1E-6 and **Flooding ΔCDF** = 1.7E-6]. |

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 unavailability of both RCIC and HPCF.*

***Applicable Plant(s)–*** ABWR

***Results–*** The results presented are for internal events only (excluding internal fires and flooding).

[1] The US-APWR SPAR model results are preliminary. The model has not undergone the full set of quality assurance checks that are completed for all SPAR models.

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| ABWR | RCIC pump unavailable. | 1 year | 4.1E-7 | **SPAR Model–** RCIC Pump FTS basic event set to TRUE. |
| | RCIC pump and both HPCF pumps unavailable. | 1 year | 1.6E-6 | **SPAR Model–** RCIC Pump FTS and HPCF Pumps CCF FTS basic events set to TRUE. |

4. **Suction source failure of EFW system leading to common-mode failure of EFW pump(s).**

   ***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 U.S. EPR

   ***Results–*** The results presented are for internal events, including internal fires and flooding, where indicated.

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| US-APWR | One MDEFW pump and one TDEFW pump unavailable due to lost suction source. | 1 year | 1.3E-4 | **SPAR Model–** Basic event representing the plugging of the EFW Pit B discharge line is set to TRUE and basic events representing start failures of EFW Pump C (motor-driven) and EFW Pump D (turbine-driven) are set to TRUE. |
| | | 1 year | 7.7E-5 | **MHI PRA (Internal Fire and Flooding)–** Basic event representing plugging of the EFW Pit B Discharge Line Isolation Valves is set to TRUE and basic events representing start failures of EFW Pump C (motor-driven) and EFW Pump D (turbine-driven) are set to TRUE. [**Fire ΔCDF =** 3.6E-5 and **Flooding ΔCDF =** 4.1E-5]. |
| U.S. EPR | One train of EFW unavailable due to lost suction source. | 1 year | 7.7E-7 | **Areva PRA–** RAW (internal events including internal fires and floods) = 2.47. |

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 1 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

***Results–*** The results presented are for internal events only (excluding internal fires and flooding).

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| **AP1000** | PXS-V121A fails to remain open | 295 days | 9.0E-5 | **SPAR Model–** PXS-V121A basic event set to TRUE (i.e., failure to remain open). |
| | | 1 year | 1.1E-4 | |

**6. Boric acid corrosion leads to reactor vessel head degradation.**

***Related SDP Cases–*** At Davis-Besse, a cavity (caused by boric acid corrosion) had formed around the a control rod drive mechanism (CRDM) nozzle in the 6.63 inch-thick low-alloy steel portion of the reactor pressure vessel (RPV) head, leaving only the stainless steel clad material (measuring 0.202 to 0.314 inches-thick) as the reactor coolant pressure boundary over an area of approximately 20 square-inches. The through-wall axial crack in the nozzle had been leaking for a period of 6 to 8 years before detection in February 2002.

The initial Phase 3 SDP analysis modeling of the as-found cavity suggest that the risk due to potential rupture may be low (i.e., CDF <$10^{-6}$/RY), there were significant unanalyzed parameters in which insufficient knowledge was available at the time of the analysis to evaluate and quantify the risk. However, given the breadth and amount of unanalyzed parameters, there is great potential for the risk to be substantially higher than that quantified for the modeled cavity. Therefore, the results of the Phase 2 analysis were used to determine the SDP result. The Phase 2 process calls for the use of either one or two orders of magnitude increase in the LOCA frequency. Increasing the MLOCA and LLOCA frequency by two orders of magnitude, through the SDP counting rule, and was determined to be appropriate given the significant unanalyzed parameters. Therefore, using the insights of the Phase 3 assessment and the results of the Phase 2 worksheets, the RPV head cavity risk was determined to be in the RED range ( CDF >$10^{-4}$/RY).

**Note:** The assessment below for the US-APWR and AP1000 design represents one of the three examples presented before the Commission during the October 14, 2010 briefing on SECY-10-0121.

***Applicable Plant(s)–*** US-APWR and AP1000

***Results–*** The results presented are for internal events only (excluding internal fires and flooding).

| Plant | Example | Exposure Period | ΔCDF | Model Used/Assumptions |
|---|---|---|---|---|
| **US-APWR** | Increased likelihood of medium and large loss-of-coolant accidents due to corrosion of reactor vessel head. | 1 year | 1.4E-7 | **SPAR Model–** Increased then MLOCA and LLOCA initiating event frequencies by two orders of magnitude. |
| **AP1000** | Increased likelihood of medium and large loss-of-coolant accidents due to corrosion of reactor vessel head. | 1 year | 1.2E-6 | **SPAR Model–** Increased then MLOCA and LLOCA initiating event frequencies by two orders of magnitude. |

***Management Directive (MD) 8.3 Examples for New Reactors***

| Estimated Conditional Core Damage Probability (CCDP) | | | | |
|---|---|---|---|---|
| CCDP < 1E-6 | 1E-6 – 1E-5 | 1E-5 – 1E-4 | 1E-4 – 1E-3 | CCDP > 1E-3 |
| No Additional Inspection | | | | |
| | Special Inspection (SIT) | | | |
| | | Augmented Inspection (AIT) | | |
| | | | Incident Investigation (IIT) | |

**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

| Plant | Example | CCDP | Notes |
|---|---|---|---|
| **ABWR** | Loss offsite power with EDG failure. | 4.7E-6 | Offsite power was available to be restored immediately after reactor trip. Offsite power recovery curves used for all non-recovery probabilities (basis for the curves is provided by NUREG/CR-6890): 30 min, 2 hrs, 8 hrs, 15 hrs, and 24 hrs. |
| | Loss offsite power with stuck-open SRV and EDG failure. | 1.1E-5 | |

**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: UA-APWR and EPR. For the US-APWR we would apply the EDG failure as a failure of the combustion turbine generator (CTG).

| Plant | Example | CCDP | Notes |
|---|---|---|---|
| **US-APWR** | Case 1– Loss offsite power. | 6.2E-6 | Offsite power was available for recovery in ~1 hour. Offsite power recovery curves used for all non-recovery probabilities (basis for the curves is provided by NUREG/CR-6890): 1 hr, 2 hrs, and 4 hrs. |
| | Case 2– Loss of offsite power with a CTG failure. | 7.4E-5 | |
| **AP1000** | Case 1– Loss of offsite power. | 4.6E-9 | Offsite power was available for recovery in ~1 hour. Therefore, the 30 minute recovery event for the AP1000 was modeled as failed. |
| **U.S. EPR** | Case 1– Loss of offsite power. | 1.5E-7 | **Areva–** LOOP Frequency = 1.91E-02 /yr; Recovery of offsite power within 1 hour = 5.3E-01; Recovery of offsite power within 2 hours =3.2E-01. |
| | Case 2– Loss of offsite power with an EDG failure. | 4.6E-7 | |

**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.

**Note:** The assessment below for the AP1000 design represents one of the three examples presented before the Commission during the October 14, 2010 briefing on SECY-10-0121.

***Applicable Plant(s)–*** AP1000

| Plant | Example | CCDP | Notes |
|---|---|---|---|
| **AP1000** | Steam generator tube rupture initiating event. | 1.3E-5 | None. |

***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.

   **Note:** The assessment below for the ABWR is representative of one of the three examples presented before the Commission during the October 14, 2010 briefing on SECY-10-0121.

   **ABWR Inputs and Assumptions**
   - **ABWR:** SPAR (Toshiba)
   - CDF (internal events at-power, no flooding) = 3.97E-7 /yr
   - Assumed critical hours (3-year) = 25,000
   - 3 EDGs included, non-safety combustion gas turbine not included
   - EDG mission time = 24 hr
   - No load/run basic event modeled in SPAR, URI contribution ignored
   - CCF multiplier = 2.0
   - Assumed EDG starts (3-year) = 50/EDG, total = 150 (pooled)
   - Assumed EDG run-hours (3-year) = 200/EDG, total = 600 (pooled)
   - CCF-adjusted Birnbaums range from 3E-7/yr to 3E-6/yr
   - Assumed UAI = 0

   **ABWR Results**
   **Base Case:** no failures, **MSPI = -6.67E-8**
   **Case 1A:** 3 EDG start failures, **MSPI = 2.47E-9**
   **Case 1B**: 3 EDG run failures, **MSPI = 2.42E-7**
   **Case 1C**: # start failures to MSPI > 1E-6 is **greater than 25**
   **Case 1D**: # run failures to MSPI > 1E-6 **is 11**
   **Case 1E**: PLE for **10 failures**

   **Note:** risk cap never applied because delta URI for single failures less than 5E-7.

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.

   **U.S EPR Inputs and Assumptions**
   - **U.S. EPR:** FSAR Rev 2, Chapter 19
   - CDF (internal events at-power, no flooding) = 2.8E-7 /yr (Section 19.1.4.1.2.1 of FSAR)
   - Assumed critical hours (3-year) = 25,000
   - 4 EDGs included, non-safety SBO DGs not included
   - EDG mission time = 12 hr (Page 19.1-25 of FSAR)

- URI contribution from load/run ignored
- CCF multiplier = 3.0
- Assumed EDG starts (3-year) = 50/EDG, total = 200 (pooled)
- Assumed EDG run-hours (3-year) = 200/EDG, total = 800 (pooled)
- Performance data and importances from Table 19.1-14 of FSAR and RAI response to Question 19-126 (some EDG data are estimated)
- Assume UA (planned) = 0
- Approximately 42.5 hr per train (total 170 hr) of unplanned unavailability gives UAI=0
- Use 2701 hr unplanned unavailability beyond 170 hr for simulation (i.e., total of 2701 + 170 hr)

**U.S. EPR Results**

**Base Case**: no failures, no extended EDG outage, assumed UAI=0, **MSPI = -3.23E-8**

**Case 2E1:** no failures, additional 2701 hr unplanned unavailability, **UAI=9.83E-9**, URI= -3.23E-8, **MSPI= -2.24E-8**

**Case 2E2:** no failures, one EDG assumed always to be in an unplanned outage, **UAI=9.08E-8,** URI= -3.23E-8, **MSPI= 5.86E-8**

**Case 2E3:** # start failures to MSPI > 1E-6 is **greater than 25**

**Case 2E4:** # run failures to MSPI > 1E-6 is **greater than 25**

**Case 2E4a:** # run failures to MSPI > 1E-6 (24 hr mission time) is **16**

**Case 2E5:** PLE for **12 failures**

**Case 2E6:** sensitivity study using actual EDG data provided by Areva on 9/28/11 rather than estimated values from FSAR, **base MSPI = -2.40E-8**, # run failures to MSPI>1E-6 **is greater than 25**

**Note:** risk cap never applied because delta URI for single failures less than 5E-7.

**US-APWR Inputs and Assumptions**

- **US-APWR:** DCD Rev 3, Tier 2, Chapter 19 and PRA Rev. 3
- CDF (internal events at-power, no flooding) = 1.0E-6 /yr (Section 19.1.4.1.2 of DCD)
- Assumed critical hours (3-year) = 25,000
- 4 CTGs included, non-safety SBO CTGs not included
- Use EDG industry average failure rates, train unavailability, and prior distribution parameters
- CTG mission time = 24 hr (Chapter 6, Attachment 6A of PRA)
- Load/run included in PRA and included in URI
- CCF multiplier = 3.0
- Assumed CTG starts (3-year) = 50/CTG, total = 200 (pooled)
- Assumed CTG load/run (3-year) = 50/CTG, total = 200 (pooled)
- Assumed CTG run-hours (3-year) = 200/EDG, total = 800 (pooled)
- Performance data and importance values from PRA Table 18.2-1 of PRA
- Assume UA (planned) = 0
- Approximately 42.5 hr per train (total 170 hr) of unplanned unavailability gives UAI=0
- Use 2701 hr unplanned unavailability beyond 170 hr for simulation (i.e., total of 2701 + 170 hr)

**US-APWR Results**

**Base Case**: no failures, no extended CTG outage, assumed UAI=0, **MSPI = -5.74E-8**

**Case 2A1:** no failures, additional 2701 hr unplanned unavailability, **UAI=4.32E-8**, URI= -5.74-8, **MSPI= -1.42E-8**

**Case 2A2:** no failures, one CTG assumed always to be in an unplanned outage, **UAI=3.99E-7,** URI= -5.74E-8, **MSPI= 3.42E-7**

**Case 2A3:** # start failures to MSPI > 1E-6 is **greater than 25**

**Case 2A4:** load/run failures to MSPI > 1E-6 is **greater than 25**

**Case 2A5:** run failures to MSPI > 1E-6 is **18**

**Case 2A6:** PLE for **15 failures**

**Case 2A7:** A review of the preliminary US-APWR SPAR model shows nearly identical Birnbaum values for the CTGs when compared to those in the DCD PRA. The overall MSPI results would not change substantially.

**Note:** risk cap never applied because delta URI for single failures less than 5E-7.

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.

<u>**US-APWR Inputs and Assumptions**</u>

- **US-APWR:** DCD Rev 3, Tier 2, Chapter 19 and PRA Rev. 3
- CDF (internal events at-power, no flooding) = 1.0E-6 /yr (Section 19.1.4.1.2 of DCD)
- Assumed critical hours (3-year) = 25,000
- 2 MD-EFW pumps, 2 TD-EFW pumps, and 2 highest importance MOVs (103A & D) included
- Mission time = 24 hr
- CCF multiplier = 1.5 (TD pumps), 1.25 (MD pumps), and 2 for MOVs
- Assumed TD-pump starts (3-year) = 36/pump, total = 72 (pooled)
- Assumed TD-pump run-hours (3-year) = 54/pump, total = 108 (pooled)
- Assumed MD-pump starts (3-year) = 50/pump, total = 100 (pooled)
- Assumed MD-pump run-hours (3-year) = 100/pump, total = 200 (pooled)
- Assumed MOV demands (3-year) = 50/MOV, total = 100 (pooled)
- CCF-adjusted Birnbaums ranged from 6E-6/yr for MOV, to 5E-6/yr for TD-EFW pump, to 1E-6/yr for MD-EFW pump
- Reliability and importance data from DCD Table 19.1-30 and 31, and PRA Table 18.2-1
- Assumed UAI = 0

<u>**US-APWR Results**</u>

**Base Case**: no failures, assumed UAI=0, **MSPI = -4.50E-8**

**Case 3A:** 3 TDP run failures, **MSPI= 1.81E-7**

**Case 3B:** 3 TDP start failures, **MSPI= 1.48E-7**

**Case 3C:** # TDP run failures to MSPI > 1E-6 is **14**

**Case 3D:** # TDP start failures to MSPI> 1E-6 is **17**

**Case 3E:** # MDP start or run failures to MSPI > 1E-6 is **greater than 25**

**Case 3F:** # MOV failures to MSPI > 1E-6 is **greater than 25**

**Case 3G:** PLE for **8 failures (TDP), 6 failures (MDP), or 5 failures (MOV)** ***CORRECTED***

**Case 3H:** A review of the preliminary US-APWR SPAR model shows Birnbaum values for both the TDPs and MDPs that are about 3 to 5 times higher compared to those in the DCD PRA, while MOV 103 A & D values are comparable.

**Case 3H1:** Base case w SPAR model, no failures, MSPI = **-2.95E-7**

**Case 3H2:** SPAR model, 3 TDP start failures, **MSPI = 1.004E-6**

**Case 3H3:** SPAR model, 3 TDP run failures, **MSPI = 1.194E-6**

**Case 3H4:** SPAR model, **> 25 MDP start or run failures** to MSPI > 1E-6

**Note:** risk cap never applied because delta URI for single failures less than 5E-7.