Entergy

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October 5, 2009

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Enforcement Discretion – Technical Specification 3.7.8 Required Action A.1

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Dear Sir or Madam:

The letter confirms the results of the teleconference that was conducted between Entergy Nuclear Operations, Inc. (ENO) and the Nuclear Regulatory Commission (NRC) at 1500 EDT, on October 1, 2009, in which ENO requested the NRC to exercise enforcement discretion from compliance with the requirements of Technical Specification (TS) 3.7.8 Required Action A.1 for Palisades Nuclear Plant (PNP). TS 3.7.8, "Service Water System (SWS)," Limiting Condition for Operation 3.7.8 requires two operable SWS trains. TS 3.7.8 Required Action A.1 requires that, with one or more SWS trains inoperable, restore the inoperable trains to operable status within 72 hours. TS 3.7.8, Required Actions B.1 and B.2 require that, if the required action and associated completion time of Condition A is not met, be in Mode 3 within six hours and in Mode 5 within 36 hours.

At the time of the teleconference on October 1, 2009, PNP was operating at approximately 100% power. On September 29, 2009, at approximately 0908 EDT, three control room alarms unexpectedly annunciated, indicating that standby service water pump P-7B had started, and the critical and non-critical service water header pressures were low. Service water pump P-7C was operating with abnormally low amperage and exhibiting signs of duress, with the pump shaft visibly vibrating and no pump discharge pressure. The pump was immediately secured and PNP entered Technical Specification 3.7.8 Condition A.

A001
NRR

ENO requested enforcement discretion for a period not to exceed 24 hours to complete repairs and post-maintenance testing of service water pump P-7C. The approval of the requested enforcement discretion was effective at 0908 EDT on October 2, 2009, and would expire at 0908 EDT on October 3, 2009.

This request was verbally transmitted to members of the NRC staff on October 1, 2009, at 1500 EDT. The NRC verbally granted the request on October 1, 2009, at 1900 EDT. Subsequently, at 0822 EDT on October 2, 2009, service water pump P-7C maintenance activities were completed and the pump was declared operable. PNP exited Required Action A.2 in TS 3.7.8, and the enforcement discretion was no longer needed.

Attachment 1 provides information documenting ENO's verbal request for enforcement discretion. It provides the information specified in NRC Regulatory Issue Summary 2005-01, "Changes to Notice of Enforcement Discretion (NOED) Process and Staff Guidance," dated February 7, 2005. Attachment 2 provides a risk evaluation of the requested enforcement discretion.

This letter contains no revisions to existing commitments and makes one new commitment:

> Ensure risk management actions provided in section four of Attachment 1 are continued for the duration of this enforcement discretion.

A copy of this request has been provided to the designated representative of the State of Michigan.

Sincerely,

cjs/jse

Attachment(s): 1. Request for Enforcement Discretion
2. Risk Evaluation of Service Water Pump LCO Extension

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades USNRC

## ATTACHMENT 1
## REQUEST FOR ENFORCEMENT DISCRETION

### 1. TECHNICAL SPECIFICATION OR OTHER LICENSE CONDITIONS THAT WILL BE VIOLATED

Palisades Nuclear Plant (PNP) Technical Specification (TS) 3.7.8, "Service Water System (SWS)," Limiting Condition for Operation 3.7.8 requires two operable SWS trains.

TS 3.7.8 Required Action A.1 requires that, with one or more SWS trains inoperable, restore the inoperable trains to operable status within 72 hours.

TS 3.7.8, Required Actions B.1 and B.2 require that, if the required action and associated completion time of Condition A is not met, be in Mode 3 within six hours and in Mode 5 within 36 hours.

### 2. CIRCUMSTANCES SURROUNDING THE SITUATION, INCLUDING LIKELY CAUSES, THE NEED FOR PROMPT ACTION, ACTION TAKEN IN AN ATTEMPT TO AVOID THE NEED FOR A NOTICE OF ENFORCEMENT DISCRETION (NOED), AND RELEVANT HISTORICAL EVENTS

**Circumstances Surrounding the Situation**

At 0908 hours on September 29, 2009, with PNP operating at approximately 100% power and service water pumps P-7A and P-7C in service, the Control Room received alarms for low service water pressure and standby service water pump auto start. An Auxiliary Operator (AO) was dispatched to investigate. The AO found all three service water pumps running, the discharge pressure of P-7C at 0 psig, and significant vibration of the packing shaft of P-7C. The AO recommended that Control Room operators stop P-7C, and the pump was immediately secured. Three operable service water pumps are required per TS 3.7.8. TS 3.7.8 Condition A was entered at 0908 hours on September 29, 2009.

**Likely Causes**

Immediately after service water pump P-7C was stopped, the pump was visually inspected. This inspection included all shafts, couplings, pump casings, spiders, bearing areas, suction bell and rotating elements. It was found that the packing gland nuts were not in place on the studs for the packing gland follower and there appeared to be damage to the packing shaft above the packing follower. The upper impeller was inspected with a boroscope with no damage or signs of failure observed. The pump was able to rotate freely. No other damage or signs of failure were immediately apparent.

A plan was developed for disassembly of the P-7C motor and pump for failure determination and repairs. Upon disassembly, the coupling between the packing shaft and

the top line shaft was found broken into two pieces. The material of the coupling is 416 stainless steel heat treated to a specified 28-32 Rc (Rockwell Hardness). The failed coupling was sent to an independent metallurgy laboratory for analysis. Per the metallurgists at the laboratory, the fracture surfaces were consistent with brittle fracture due to overload. Per ASTM Standard A582/A 582M – 95b "Standard Specification for Free-Machining Stainless Steel Bars," the hardness of material should be between 24 and 32 Rc (248 to 302 HB (Brinell Hardness)) for an intermediate temper condition. The laboratory found the hardness to be 37 Rc throughout the material. The material was also cut longitudinally and examined under an electron microscope. This examination found precipitates at the grain boundaries, which is not expected for this material. The hardness testing results and the precipitates are indicative of a problem in the heat treat process which caused the material to be susceptible to brittle failure. A review of the Certified Material Test Reports from the pump supplier, Hydro-Aire, shows that the final hardness of the couplings delivered with the pump were within specification. This conflicts with the results from the laboratory.

The catastrophic failure of the coupling was most likely due to brittle fracture in an overload condition. The overload was most likely caused by the stopping and starting of the pump to clear the basket strainers. Approximately 1-1/2 hours before the failure of the pump, service water pump P-7C was stopped and re-started, as were the other two pumps.

All couplings on service water pump P-7C will be replaced to address the material issues. Additionally, the pump packing shaft and motor shaft will be replaced due to excessive runout that was caused by the event.

An inspection of the service water bay was completed and no significant debris was observed that could cause failure of the pump.

The service water pump P-7C check valve was determined to not be a cause of the pump failure based on review by the check valve program engineer. The P-7C check valve is monitored quarterly and check valve operation was found acceptable on its most recent test date of July 23, 2009.

Packing gland bolts on service water pumps P-7A and B were inspected and found to be in satisfactory condition and fully engaged.

Service water pump P-7A is identical to P-7C. Service water pump P-7A was refurbished during the 2009 refueling outage. The P-7A refurbishment included the same stainless steel shafting, coupling, and impeller components as P-7C. The couplings in service water pump P-7A were fabricated in April 2008, thirteen months prior to those fabricated for P-7C. The heat numbers for the two batches of manufactured couplings are different; therefore, it is not credible that the heat treatment problem that caused the failure of the P-7C coupling is related to the couplings installed in P-7A. Service water pump P-7A has operated since May 2009. Service water pump

P-7C was refurbished in June 2009. Therefore, P-7A has more operating time than P-7C and has not exhibited any signs of degraded performance.

Service water pump P-7B is a pump of a different vendor than that of P-7A and P-7C. It has similar shafting and coupling dimensional arrangements as P-7A and P-7B, however the shafting and coupling material is carbon steel. P-7B was refurbished in September 2007 during the refueling outage. This refurbishment used carbon steel shafting and couplings (P-7B has not yet been refurbished with the stainless steel components that have been implemented in P-7A and P-7C). Therefore, the couplings for P-7B are not subject to the same failure mode as the P-7C pump coupling. P-7B has been operating since the end of the September 2007 refueling outage and has not exhibited any signs of degraded performance.

**Need for Prompt Action**

If operability of service water pump P-7C cannot be restored by 0908 on October 2, 2009, PNP is required to shut down. The expected duration of the outage based on the current schedule, will restore service water pump P-7C within 24 hours of the required completion time for TS 3.7.8 Action A.1. Time is needed to manufacture, transport, and install new couplings. Pump reassembly and post-maintenance testing are also required. The motor was sent to the vendor test facility to verify that it is not damaged. New couplings are being independently tested prior to installation.

**Action Taken in an Attempt to Avoid the Need for an NOED**

Service water pump P-7C was declared inoperable on September 29, 2009, at 0908 EDT. PNP entered into augmented, around the clock staffing for multiple departments to establish and execute an expedited repair schedule. Work activities were initiated promptly to determine the cause of the problem, extent of condition, and required repairs. Visual inspections of pump components were performed, which included inspections of shafts, couplings, pump casings, spiders, bearing areas, suction bell and rotating elements. Actions were taken to identify parts and other contingencies. PNP is working with vendors to secure parts and to restore the pump. Discrepancies discovered in the hardness of the material of the failed coupling has resulted in independent verification of hardness testing or replacement couplings. Management is stationed at the vendor facility overseeing part fabrication. Pump reassembly has started and will be completed upon receipt of parts from the vendor. The remaining work activities have been identified and scheduled. The PNP event response includes frequent alignment meetings to ensure the actions are progressing as planned and that additional support is provided when needed. Management oversight ensures proper priorities are established and resources are provided. Management is providing around the clock oversight of maintenance activities. These actions ensure the pump will be restored expeditiously.

**Relevant Historical Events**

At Indian Point, on August 10, 1993, and on August 9, 1993, a shaft coupling sheared due to impact from a foreign object; the coupling material was ASTM A276 Type 410 SST. At Indian Point, on September 22, 1993, a shaft coupling sheared due to high loads during start caused by a leaking discharge check valve; the coupling material was ASTM A276 Type 410 SST. Both Indian Point events involved pumps from a different vendor.

Service water pump P-7A was replaced in April 2009 during a refueling outage as routine, periodic replacement. Service water pump P-7C was replaced on-line in June 2009 due to degraded performance as a result of ingested foreign material. Neither of these pumps replacements involved failed couplings. During the replacements, the pumps were inspected in accordance with plant procedures.

## 3. INFORMATION TO SHOW THAT CAUSE AND PROPOSED PATH TO RESOLVE THE SITUATION ARE UNDERSTOOD, SUCH THAT THERE IS A HIGH LIKELIHOOD THAT PLANNED ACTIONS TO RESOLVE THE SITUATION CAN BE COMPLETED WITHIN THE PROPOSED NOED TIME FRAME

The hardness testing results and the precipitates are indicative of a problem in the heat treat process which caused the material to be susceptible to brittle failure.

The likely cause of improper heat treatment causing the coupling material to be susceptible to brittle failure was determined by a failure modes and effects evaluation. PNP is confident that the cause is understood based on on-site visual inspections of the pump and metallurgy laboratory examinations and analyses of the failed coupling.

The planned repairs to the pump are limited and consistent with normal work practices. The repairs are expected to resolve the situation because the identified deficiencies will be corrected during the reinstallation. Management is stationed at the vendor facility overseeing part fabrication. Additional supervisory oversight of the reinstallation will provide greater assurance that the repairs are performed correctly. The schedule for repairs and subsequent post-maintenance testing was established based on previous experience with similar repairs. The replacement of the damaged coupling can be completed with normal work practices and parts. New couplings are being independently tested prior to installation. Pump shafts and other components are being inspected as well. The motor was sent to the motor vendor test facility to verify that is not damaged.

Based on the information above, the proposed NOED time frame of 24 hours provides sufficient time to complete the planned actions.

## 4. SAFETY BASIS FOR THE REQUEST, INCLUDING AN EVALUATION OF THE SAFETY SIGNIFICANCE AND POTENTIAL CONSEQUENCES OF THE PROPOSED COURSE OF ACTION

a. Risk Assessment Using the Zero Maintenance Model

PNP has evaluated the request for enforcement discretion from a probabilistic risk standpoint (Attachment 2). This assessment considered the expected plant configuration during the period of enforcement discretion and determined that it does not involve an unacceptable increase in risk. The risk of continued PNP operation with an inoperable service water pump during a 24 hour period of noncompliance beyond the TS 72-hour completion time, as measured by the incremental conditional core damage probability (ICCDP) is 8.16E-8 for a plant internal event. This is below the guidance threshold of less than or equal to 5E-07 identified in NRC Inspection Manual Part 9900. The ICCDP for seismic, fire, and flood external events is bounded by the ICCDP for internal events, and, therefore, also meets the guidance threshold. The results bound the proposed 24 hour period of noncompliance.

At PNP, core damage sequences involving a large, early release generally are those that bypass containment (i.e., those that involve steam generator tube rupture (SGTR) and intersystem loss-of-coolant accident initiating events). The incremental conditional large early release probability (ICLERP) was determined to be 7.7E-12. This is below the guidance threshold of less than or equal to 5E-08 identified in NRC Inspection Manual Part 9900.

b. Discussion of the Dominant Risk Contributors

A review of the change to the cutsets contributing to core damage as a result of the changes made to represent the removal of service water pump P-7C from service determined that there were no changes to the top 100 cutsets. A review of the changed cutsets contributing to core damage as a result of the changes made to the initiating event frequency for a loss of service water initiating event are discussed below.

The top 100 cutsets represent approximately 83% of the increased core damage probability. Thirteen cutsets showing an increased contribution to core damage are described below. The top 100 cutsets are listed in Attachment 1. Eighty seven out of one hundred of the listed cutsets did not change.

<u>Cutset 1 Loss of Service Water (Sequence 22-2)</u>

Cutset 1 is the same cutset as the baseline (0 maintenance) case with an increased contribution to core damage as it is the result of a loss of service water initiating event. The cutset represents a loss of primary coolant pump seal cooling and the failure to trip the primary coolant pump(s) in time to prevent seal failure that results

in a loss of coolant accident. The loss of service water fails injection pumps due to loss of cooling and containment heat removal.

Cutset 4 Loss of Service Water (Sequence 21-5)

This cutset represents a loss of service water. The loss of service water results a in loss of primary coolant pump seal cooling, and a consequential seal LOCA due to failure to trip the primary coolant pumps. The loss of service water fails injection pumps due to loss of cooling and containment heat removal.

Cutset 9 Loss of Service Water (Sequence 17)

This cutset represents a loss of service water event with a failure of secondary heat removal via the steam generator, successful initiation of once through cooling and failure of the containment heat removal, failure of main feedwater and low pressure feed (feeding steam generators with condensate pumps) due to loss of condenser vacuum, and failure of containment sprays and containment air coolers as a result of the loss of service water cooling to remove heat from the systems.

Cutset 19 Loss of Service Water (Sequence 17)

This cutset is similar to cutset 9 above, with the difference being the failure of auxiliary feedwater due to common cause failure of all the pump discharge check valves. The remainder of the cutset is the same as cutset 9.

Cutset 21 Loss of Service Water (Sequence 5)

Cutset 21 is also similar to cutsets 9 and 19. The difference in this cutset is that the failure of auxiliary feedwater is a long term failure to provide an alternate suction source to the auxiliary feedwater pumps. Failure of normal makeup to the condensate storage tank (T-2) is due to failure of the demineralized water transfer pump (P-936) to provide makeup from demineralized water storage tank (T-939). Operators would be aware of the failure of normal makeup when a low level alarm occurs at 73% level in the condensate storage tank. The operator would then have several hours to align an alternate source to the auxiliary feedwater pumps. This cutset includes failure of an operator action to align service water to pumps to auxiliary feedwater P-8A or P-8B OR fire protection water to auxiliary feedwater pump P-8C. This cutset does not credit the availability of water from primary system makeup storage tank (T-81) via pumped or gravity feed, which would provide additional time to align other water sources.

Cutset 23 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above, with the difference being the failure of auxiliary feedwater due to common cause failure of all the check valves in the

flow headers from the pump trains to the steam generators. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 28 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above, with the difference being the failure of auxiliary feedwater due to common cause failure of all four flow control valves in the flow headers from the pump trains to the steam generators. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 35 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above, with the difference being the failure of auxiliary feedwater due to spurious low suction trips of auxiliary feedwater pumps P-8A and P-8C, and failure of the turbine-driven auxiliary feedwater pump P-8B. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 36 Loss of Service Water (Sequence 5)

Cutset 36 is similar to cutset 21 above. Loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the demineralized water transfer pump (P-936). The difference between this cutset and cutset 21, is that the long term failure is the failure of another operator action related to the alignment of an alternate suction source to the auxiliary feedwater pumps after the contents of the condensate storage tank (T-2) have been depleted.

Cutset 37 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above, with the difference being the failure of auxiliary feedwater due to common cause failure of all three auxiliary feedwater pumps to run for the mission time (24 hours). The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 49 Loss of Service Water (Sequence 5)

Cutset 49 is similar to cutsets 21 and 36 above. Loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the condensate storage tank makeup CV-2010 to automatically open, and loss of flow from the demineralized water storage tank (T-939) to the condensate storage tank (T-2). Additionally, the cutset includes failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 60 Loss of Service Water (Sequence 5)

Cutset 60 is also similar to cutsets 21 and 36 above. In this cutset, the loss of normal makeup from the demineralized water storage tank (T-939) is due to loss of

the air supply (filter plugging) to the condensate storage tank makeup valve CV-2010. The cutset includes the failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 62 Loss of Service Water (Sequence 5)

Cutset 62 is also similar to cutsets 21 and 36 above. In this cutset, the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The cutset includes the failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 69 Loss of Service Water (Sequence 5)

Cutset 69 is also similar to cutsets 21 and 36 above. In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The long term failure of the alignment of service water or fire protection water to the auxiliary feedwater pump suction is due to failure of one of the manual valves MV-FW775 required to align fire protection water to pump P-8C (service water to pumps P-8A and P-8B is failed by the initiator).

Cutset 70 Loss of Service Water (Sequence 5)

Cutset 70 is also similar to cutsets 21 and 36 above. In this cutset, the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The long term failure of the alignment of service water or fire protection water to the auxiliary feedwater pump suction is due to failure of one of the manual valves (MV-FW774) required to align fire protection water to pump P-8C (service water to pumps P-8A and P-8B is failed by the initiator).

Cutset 75 Loss of Service Water (Sequence 5)

Cutset 75 is also similar to cutset 21 and 36 (see above). In this cutset, the loss of normal makeup from the demineralized water storage tank (T-939) is due to loss of the air supply (filter plugging) to the control valve (CV-2010). The cutset includes the failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

c. Discussion of the Compensatory Measures Implemented to Address the Dominant Risk Factors

In order to minimize risk during the period of noncompliance, PNP has identified additional controls to increase operator awareness of critical equipment, provide assurance that assumptions in the risk model are maintained, and minimize the

likelihood of a plant transient. PNP proposes the following actions during the period of enforcement discretion to manage risk:

1) No non-essential work will be allowed that could potentially jeopardize stable plant operation.

2) PNP has designated the following equipment as "Protected Equipment" and control the protected equipment in accordance with the applicable procedure during the extended TS action completion time:

    - service water pump P-7A
    - service water pump P-7B
    - electric fire water pump P-9A
    - diesel fire water pump P-9B
    - diesel fire water pump P-41
    - containment spray pumps P-54B and C
    - emergency diesel generator 1-1
    - component cooling water pump P-52A
    - component cooling water pump P-52B
    - 2400 VAC safeguards bus 1C
    - auxiliary feedwater pumps P-8A and B
    - screen wash pump P-4
    - traveling screens F-4B and C
    - traveling screens control panel
    - supplemental emergency diesel generator 1-3
    - Safeguards bus room
    - traveling screen F-4C breaker 52-563
    - traveling screen F-4B breaker 52-561
    - screen wash pump P-4 breaker 52-1406
    - switchyard

3) PNP is conducting hourly monitoring of critical service water header pressure, service water pump amperage, and lake (ultimate heat sink) temperature.

4) PNP is monitoring the following components every two hours:

    a. service water pump P-7A and B
    b. traveling screens F-4B and C
    c. screen wash pump P-4
    d. fire water pumps P-9A and B
    e. diesel fire water pump P-41
    f. service water pumps P-7A and B basket strainer differential pressure

5) The plant operations crews have been briefed on these risk management measures.

6) Guidance was developed for cycling service water pump P-7A and B in the event of increasing basket strainer differential pressure to reduce this pressure.

7) Operators have been briefed on a loss of service water (Off Normal Procedure 6.1, "Loss of Service Water").

8) Operators have been briefed on service water leak and increased flow scenarios.

9) Fire tours have been established in the screen house and the 1C switchgear room.

d. Demonstration of how the Proposed Compensatory Measures are Accounted for in the PRA

The benefit of the compensatory actions in general is in protecting equipment and not allowing test and maintenance activities on those components during the period of enforcement discretion. Since the process requires that the analyses be completed using a zero maintenance condition for the baseline risk, the benefits of protecting equipment is not quantifiable. A separate analysis was conducted of the change in risk for the cases analyzed to support the NOED using the normal maintenance baseline of the model. Using this baseline for risk and changing the events in the model for out of service conditions for protected components to zero (FALSE) demonstrated that, for the case of removing pump P-7C from service, the risk was returned to nominal baseline risk. That is, implementation of the contingencies offset the risk increase of the pump out of service. For the case of increasing the initiating event frequency for a loss of service water event, it was assumed that the compensatory measures would result in a smaller increase in the initiating event frequency. An increase of a factor of two versus the order of magnitude increase was used. This resulted in an overall reduction of the change in risk by approximately 53% (changed from 8.0E-08 to 4.24E-08) for an extension of 24 hours. The same factor would be applicable to any extended period.

e. Extent of Condition

Subsequent sensitivity analyses were performed to assess the potential for an increase in the probability of failure of the service water pump P-7A or B as a result of a potential common cause contributor. The sensitivity analysis was completed by increasing the probability of the common cause failure of the operating service water pumps by a factor ten. The results of the sensitivity analysis indicated that increasing the probability of failure of the pump resulted in no significant change to core damage frequency calculated for any of the cases analyzed. The incremental core damage probability increased from 8.0E-08 to 8.16E-08. The event contribution of the increased common cause term is 0.04%. The contribution of the

increased initiating event frequency for loss of service water increased to 87.8%. Clearly, the increased loss of service water initiator dominates the risk increase.

f. External Event Risk

Seismic Events

In the PNP IPEEE (Individual Plant Examination of External Events) (References 2.2.17 and 2.2.18) a seismic risk assessment was used to assess risks due to seismic events. The risk assessment was a hybrid of the conventional PSA and seismic margins analysis.

The service water system modeling used in the external events analysis is the same model used for internal events analysis. The same system success criteria were also used. The component random failure rates that were used in the IPE (Individual Plant Examination of Internal Events) (Reference 2.2.16) were also used in the SPRA (Seismic Probabilistic Risk Assessment). No adjustments to these probabilities were made. The seismic impact on these components was assessed by including seismic basic events and fragilities. The component fragilities that were identified in Section 3.5.2 of the IPEEE reports were used in the SPRA. The fragilities were input as a median capacity with a lognormal standard deviation (beta), which defined a lognormal fragility curve.

In addition to the seismic basic events, the seismic fault trees were modified to include seismically induced initiating events. The four seismic event tree headings that are seismically induced initiating events are: TBFR (Turbine Building Fire); TBFL (Turbine Building Flood); LOOP (Loss of Offsite Power); and SBL (Small Break Loss of Coolant Accident). All events that are affected by a turbine building fire have an associated basic event of TBFR. All basic events that are affected by a turbine building flood have an associated basic event of TBFL. The affected off site power related equipment received an associated basic event of LOOP. The initiating event SBLOCA (Small Break Loss of Coolant Accident) was given to all sequences that were quantified by the SBLOCA event tree and was not included in the fault tree as a basic event.

The seismic analysis has not been updated since originally developed for the Individual Plant Examination of External Events (IPEEE) submittal. A review of the results of the IPEEE submittal indicated that the core damage frequency was 8.88E-06 with a high confidence low probability of failure (HCLPF) of 0.217g PGA (peak ground acceleration). There were no specific seismic events identified as dominant contributors to the core damage frequency. Important seismic induced failures identified were: the fire protection system, main steam isolation valves, diesel generator fuel oil supply, and an undervoltage relay for 2400 volt ac Bus 1D. Several important random failures were identified in the report as important because of their contribution in combination with seismically induced failures. The important random failures (not seismically induced) identified in the report were: emergency

diesel generator 1-2, auxiliary feedwater (AFW) pump P-8C, and the atmospheric dump valves.

The service water system was determined to be seismically rugged and there were no significant contributions of the service water system to core damage resulting from seismically induced failures. Random failures of the service water system were identified as important contributors as a consequence of seismically induced failure of other system components as discussed below.

As noted, the fire protection system is an important contributor to seismic analysis due to the probability of seismically induced failure of fire protection system components and the condensate storage tank (CST). Seismically induced failure of the CST results in an earlier need for alignment of an alternate suction source for the operating AFW pump. The fire protection system provides an alternate suction source to AFW pumps P-8A and P-8B. The seismically induced failures of the fire protection system result in long term failure of AFW pumps P-8A and P-8B due to the unavailability of a suction source. These failures result in an increased importance of the random failures of the pump P-8C train to provide successful heat removal after depletion of the CST inventory. The same conditions result in an increased importance of the random probability of failure of the service water system to support operation of pump P-8C.

Auxiliary feedwater pump P-8C is important to long term makeup to the steam generators should the fire system become unavailable following a seismic event (as discussed in the results for Accident Classes IA & IB, Section 3.6.5.3.1 of the IPEEE report). The fire protection system has a low fragility and is a significant contributor to seismic risk once the contents of the condensate storage tank (T-2) are depleted and a long term suction source is required for continued operation of the AFW pumps. The seismically induced failure of the fire protection system represents a higher probability of failure of the long term suction to motor-driven AFW pump P-8A and turbine-driven AFW pump P-8B after the depletion of the available tank T-2 inventory. This increased probability of failure of heat removal via the AFW P-8A and P-8B pump trains results in an increased importance of motor-driven auxiliary feedwater pump P-8C. The importance of pump P-8C is a consequence of the fact that service water (a much more seismically rugged system) is more likely to remain available as a long term suction source to pump P-8C.

Auxiliary feedwater flow requirements in the PRA are 165 gpm to either steam generator. These flow requirements are a small fraction (<2%) of the total flow (8000 gpm) from a single service water pump. At the time of condensate storage tank depletion, the flow requirements will be lower. Therefore, the PRA model assumes no additional service water pumps are required to be placed in service to provide a suction source for the AFW pumps.

For the condition stated, either service water flow to the non-critical header or service water flow to the critical loads in containment not isolated and providing AFW suction, two SW pumps would be required.

The contribution to core damage from seismic events determined in the IPEEE was 8.88E-06. This represents approximately 13% of the total core damage frequency from the current internal events analysis (2.49E-05), fire (3.3.1E-05), flooding (~<2E-07) and seismic (8.88E-06). Therefore the expected seismic contribution is bounded by the internal events core damage assessment.

Fire Events

The PNP fire analysis used an approach that combined the deterministic evaluation techniques from the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology with classical PRA techniques. The FIVE methodology was used to establish fire boundaries and to evaluate the probability and the timing of damage to components located in a fire area/zone involved in a fire. Based on the results from implementing the FIVE methodology, PRA techniques were then employed to determine the probability of core damage associated with fires within the identified fire areas/zones. Fire areas identified by the fire protection program were used as the basis of the fire areas evaluated by the fire risk analysis. These fire areas were evaluated for further division based on combustible loading and fire-spread potential to identify fire zones within fire areas. The fire areas/zones identified were evaluated and quantified using the fault trees and transient event tree from the IPE. The fault and event trees were modified to accurately reflect the fire analysis.

The core damage frequency contribution from internal fires for PNP is 3.31E-05/yr. The dominant contribution to the fire CDF (>89%) is related to five fire areas: cable spreading room (33.5%); main control room (24.4%); 1D switchgear room (14.7%); turbine building (9.3%); and 1C switchgear room (7.6%).

The principle finding of the fire analysis was that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures (unrelated to damage caused by the fire) or human errors, core damage will not occur. As a result, the study concluded that there are no major vulnerabilities due to fire events at PNP. This is primarily due to the fact that the damage in the important fire areas was to support systems (e.g. ac power or dc power) that resulted in the loss of one division of equipment with adequate equipment unaffected on the other division. During the service water pump P-7C repair, an operable service water pump will remain available on each division.

Flooding and Other Events

Other external events (high winds, external floods, transportation, etc.) were screened by demonstrating conformance to the 1975 Standard Review Plan using

prior evaluations completed during the Systematic Evaluation Program (SEP) or demonstrating low hazard frequency for aircraft hazards. There were no significant contributors to core damage frequency from other external events (other than seismic and fire) identified.

g. Forecasted Weather Conditions

Based on information obtained by operations there were no significant adverse weather conditions forecasted for the proposed period of this NOED. Therefore, there were no plant vulnerabilities indentified related to weather conditions. Compensatory measures identified to protect equipment during the period of the NOED are considered adequate based on the anticipated weather conditions.

Based on the risk analysis and the proposed compensatory measures, PNP concludes there is no increase in radiological risk to the public.

## 5. JUSTIFICATION FOR THE DURATION OF THE NONCOMPLIANCE

PNP requests that the NRC exercise discretion to not enforce compliance with TS 3.7.8, Required Action A.1, to allow for restoration of the service water pump P-7C to operable status. The duration of the noncompliance is limited to the time required to complete the necessary restoration activities. The restoration activities include:

- Manufacturing of replacement couplings and transportation of couplings to PLP.
- Completing maintenance activities to place service water pump P-7C back in service.
- Performing post-maintenance testing.
- Completing the operability review.

The enforcement discretion would be in effect until service water pump P-7C is restored to operable status or the 24-hour noncompliance period ends, whichever occurs first.

## 6. CONDITION AND OPERATIONS STATUS OF THE PLANT, INCLUDING SAFETY-RELATED EQUIPMENT THAT IS OUT OF SERVICE OR OTHERWISE INOPERABLE

PNP is currently at 100% power. Equipment out of service includes:

- service water pump P-7C
- RIA-2320 steam generator vent monitor
- E/U-294 ultra-violet smoke detector

The service water pump is safety-related. The other equipment is not safety-related. The information presented in section four reflects the unavailability of service water pump P-7C.

## 7. STATUS AND POTENTIAL CHALLENGES TO OFF-SITE AND ON-SITE POWER SOURCES

Diesel generators 1-1 and 1-2 are operable and available to the safeguards busses and two qualified circuits between the offsite network and the onsite Class 1-E AC electrical power distribution system are operable. Supplemental emergency diesel generator 1-3 is operable and available.

Electrical system stability was verified by the following: the 345 kV bus voltages are normal and stable; system frequency is normal and stable; and all 345 kV system line currents are normal. This will continue to be monitored.

## 8. BASIS FOR DETERMINING THAT THE NONCOMPLIANCE WILL NOT BE OF POTENTIAL DETRIMENT TO THE PUBLIC HEALTH AND SAFETY

The proposed period of noncompliance will not be detrimental to public health and safety. PNP has evaluated the risk and determined it is sufficiently low. A summary of the evaluation is provided as part of item four, above. To further protect health and safety of the public, a number of risk management actions have been taken to increase operator awareness of critical equipment, to provide assurance that assumptions in the risk model are maintained, and to minimize the likelihood of a transient for the duration of the noncompliance.

## 9. BASIS FOR CONCLUDING THAT THE NONCOMPLIANCE WILL NOT INVOLVE ADVERSE CONSEQUENCES TO THE ENVIRONMENT

Although the proposed action involves noncompliance with a requirement of the TS,

1. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite. The proposed action does not affect the

generation of any radioactive effluent, nor does it affect any of the permitted release paths; and

2. There is no significant increase in individual or cumulative occupational radiation exposure. The proposed action would not significantly affect plant radiation levels, and, therefore, would not significantly affect dose rates and occupational exposure; and

3. There are no significant nonradiological environmental consequences.

Therefore, PNP has concluded that the proposed action will not involve adverse consequences to the environment.

## 10. ONSITE SAFETY COMMITTEE REVIEW COMMITTEE REVIEW APPROVAL

This request was approved by the onsite safety review committee.

## 11. WHICH NOED CRITERION FOR APPROPRIATE PLANT CONDITIONS IS SATISFIED AND HOW IT IS SATISFIED

PNP has evaluated the requested enforcement discretion against the criteria specified in section B of NRC Inspection Manual, Part 9900: "Operations – Notices of Enforcement Discretion [NOED]," issued February 7, 2005, and in NRC Regulatory Issue Summary 2005-01, "Changes to Notice of Enforcement Discretion (NOED) Process and Staff Guidance," also dated February 7, 2005.

Section B of NRC Inspection Manual, Part 9900, states, "for an operating plant, the NOED is intended to (a) avoid unnecessary transients as a result of compliance with the license condition and, thus, minimize the potential safety consequences and operational risks, or (b) avoid testing, inspection, or system realignment that is inappropriate for the particular plant conditions."

The NOED criteria in section 2.1.1(a) for an operating plant are satisfied. PNP is operating at approximately 100% power. Compliance with TS 3.7.8 would initiate an unnecessary transient by requiring the plant to initiate a shutdown on October 2, 2009. The proposed action would allow continued plant operation to perform the required repair and testing. Granting the NOED will preclude the operational risk associated with a transient during the shutdown. No corresponding health and safety benefit is gained by requiring a plant shutdown. Based on the above, the criteria are satisfied.

## 12. FOLLOW-UP LICENSE AMENDMENT

A follow-up license amendment will not be submitted.

## 13. SEVERE WEATHER OR OTHER NATURAL PHENOMENA

The proposed enforcement discretion does not involve severe weather or other natural events.

## 14. OTHER INFORMATION

The service water system (SWS) provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident (DBA) or transient. During normal operation or a normal shutdown, the SWS also provides this function for various safety related and non-safety related components.

PNP has three service water pumps, which are designated as P-7A, P-7B and P-7C. The service water pumps are 50-percent capacity, electric motor driven pumps, connected in parallel. The service water pumps take suction from a common intake structure supplied by Lake Michigan. The motors for P-7A and P-7C are connected to one 2.4 kV bus and the motor for P-7B is connected to a separate 2.4 kV bus. The discharge of the pumps flows into a common header before splitting into three headers, two critical headers for safety-related equipment and one non-critical header for non-safety related equipment.

There are two SWS trains, each associated with a safeguards electrical train. The SWS train associated with the left safeguards train consists of one service water pump, P-7B, associated piping, valves, and controls for the equipment to perform their safety function. The SWS train associated with the right safeguards train consists of two service water pumps, P-7A and P-7C, associated piping, valves, and controls for the equipment to perform its safety function.

# ATTACHMENT 2

# RISK EVALUATION OF SERVICE WATER PUMP LCO EXTENSION

88 Pages Follow

LTR-PSA-09-04
October 1, 2009
Rev 1

To: Bob VanWagner

Subject: Evaluation of Service Water Pump (P-7C) LCO Extension Request

Prepared By: Frank Yanik and Brian Brogan

Reviewed By: Brian Brogan Brian Brogan 10/2/09

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INTRODUCTION/OBJECTIVE

The purpose of this assessment is to evaluate the safety significance of extending the service water pump P-7C Allowed Outage Time (AOT). As noted in the Palisades Technical Specifications (TS) the limiting condition for operation (LCO) for P-7C is 72 hours. Revision 1 of this evaluation addresses a common cause sensitivity analysis between P-7A and P-7B.

The objective of the PRA analysis is to provide the safety basis for an NOED request, which includes an evaluation of the safety significance and potential consequences of the proposed course of action. The results from this evaluation are an input to an NOED which is prepared by the site's Regulatory Affairs department.

## TABLE OF CONTENTS

## 1.0 BACKGROUND

### 1.1 CONDITION REPORT (CR-PLP-2009-04519)

At 09:08 hours on 09/29/2009 (all times local) the following alarms annunciated in the control room.

EK-1149, SERVICE WATER PUMPS STANDBY PUMP RUNNING,
EK-1163, CRITICAL SERV WATER HEADER 'B' LO PRESSURE,
EK-1164, CRITICAL SERV WATER HEADER 'A' LO PRESSURE, and
EK-1165, NONCRITICAL SERVICE WATER LOW PRESSURE.

Service water pump P-7B started in Standby. Service water pump P-7C was operating with 31amps (normally greater than 80) and local indication of duress (shaft visibly vibrating with no discharge pressure). Pump P-7C was secured. At the beginning of shift, service water pumps P-7A and P-7C were in service with basket strainer differential pressures at 6 psid and 5 psid respectively. The operating crew rotated the operating service water pumps leaving pumps P-7A and P-7C in-service. Final basket strainer differential pressures were: P-7A at 4 psid, P-7B at 3.5 psid and P-7C at 3.5 psid. Service Water Header Pressure rose 2.3 psi.

The operating crew entered ONP-6.1, "Loss of Service Water" and Technical Specification LCO 3.7.8 (72 hours). There was no Emergency Plan impact and the event was not reportable.

### 1.2 Introduction/OBJECTIVE

The purpose of this assessment is to evaluate the safety significance of extending the service water pump P-7C Allowed Outage Time (AOT). As noted in the Palisades Technical Specifications (TS) the limiting condition for operation (LCO) for service water pump P-7C is 72 hours.

The objective of this PRA analysis is to provide the safety basis for an NOED request, which includes an evaluation of the safety significance and potential consequences of the proposed course of action. The results from this evaluation are an input to an NOED which is prepared by the site's Regulatory Affairs department.

## 2.0 ANALYSIS INPUT/REFERENCES

### 2.1 INPUT

2.1.1 SAPHIRE Codes - executables (*.exe files) can be found in the "J:\Engineering\Eng_prgm\Rel_Eng\PSA\SAPHIRE" folder on the Palisades intranet. Table 2.1-1 lists the file specifics.

**Table 2.1.1 (Reference 2.2.5)**

| Filename | Date | Time | Size |
|---|---|---|---|
| SAPHIRE-7-26-866621894.exe | 10/24/2005 | 3:45p | 14,079 KB |

2.1.2 Table 2.1.2 below lists the baseline CAFTA files. This baseline CAFTA model (Reference 2.2.1) serves as the starting point of the core damage fault tree model update documented in this analysis.

| Table 2.1.2 | | | | |
|---|---|---|---|---|
| **Filename** | **Description** | **Date** | **Time** | **Size - KB** |
| PSAR2c.be | PSAR2c CAFTA Basic Event File | 6/26/2006 | 1:42p | 1,248 |
| PSAR2c.caf | PSAR2c CAFTA Fault Tree File | 6/26/2006 | 1:36p | 449 |
| PSAR2c.gt | PSAR2c CAFTA Gate Type File | 6/24/2006 | 1:31p | 1,024 |
| PSAR2c.tc | PSAR2c CAFTA Type Code File | 5/27/2004 | 9:03a | 30 |
| PSAR2c CAFTA Files.zip | PSAR2c CAFTA zip file | 6/29/2006 | 8:47a | 289 |

2.1.3 Table 2.1.3 lists the PSAR2c SAPHIRE project file (Reference 2.2.1) used as the initial data set for this analysis.

| Table 2.1.3 | | | | |
|---|---|---|---|---|
| **Filename** | **Date** | **Time** | **Size - KB** | **Description** |
| Caf2Sap PSAR2c.txt | 6/29/2006 | 8:59a | 11 | Text rules file used by caf2sap.exe to create MAR-D files. |
| caf2sap.exe | 3/24/2003 | 8:16a | 28 | Visual basic application for creating SAPHIRE MAR-D fault tree files. |
| Creation of Rules File PSAR2c.xls | 6/26/2006 | 2:42p | 2,162 | EXCEL spreadsheet that creates the *.txt rules file for SAPHIRE MAR-D fault tree assembly. |
| PSAR2c FTree Logic.ftl | 6/29/2006 | 9:16a | 3,421 | MAR-D fault tree file created from the PSAR2c CAFTA master fault tree. |
| SAPHIRE v7.26 PSAR2c Ftree Files.zip | 6/29/2006 | 9:43a | 1,099 | Above listed supporting files. |

2.1.4 Table 2.1.4 defines the House Event configuration used in this evaluation:

| Table 2.1.4 | | | |
|---|---|---|---|
| **House Event** | | **House Event** | |
| A-HSE-CST-MAKEUP | F | I-HSE-M2LEFT-INS | T |
| C-HSE-P-52A-STBY | T | I-HSE-M2RGHT-INS | F |
| C-HSE-P-52B-STBY | T | M-HSE-P-2A-TRIP | T |
| C-HSE-P-52C-STBY | F | M-HSE-P-2B-TRIP | F |
| D-HSE-CHGR1-INS | T | M-HSE-SJAE1-INS | T |
| D-HSE-CHGR2-INS | T | M-HSE-SJAE2-INS | F |
| D-HSE-CHGR3-INS | F | U-HSE-P-7A-STBY | F |
| D-HSE-CHGR4-INS | F | U-HSE-P-7B-STBY | F |
| E-HSE-AIR-GT-75F | T | U-HSE-P-7C-STBY | T |
| E-HSE-AIR-LT-75F | F | X-HSE-2SG-BLDN | 1 |
| E-HSE-BYPASS-REG | T | X-HSE-2SG-BLDN-A | 1 |
| E-HSE-EDG11-DEM | T | X-HSE-2SG-BLDN-B | 1 |
| E-HSE-EDG11-RUN | T | X-HSE-SGA-BLDN | 1 |
| E-HSE-EDG12-DEM | T | X-HSE-SGB-BLDN | 1 |
| E-HSE-EDG12-RUN | T | Y-HSE-LOOP1A-BRK | T |
| I-HSE-C-2AC-INS | T | Y-HSE-LOOP1B-BRK | F |
| I-HSE-C-2B-INS | F | Y-HSE-LOOP2A-BRK | F |
| I-HSE-F-12A-INS | T | Y-HSE-LOOP2B-BRK | F |
| I-HSE-F-12B-INS | F | Y-HSE-RAS-POST | F |
| I-HSE-F-5A-INS | T | Y-HSE-RAS-PRE | F |
| I-HSE-F-5B-INS | F | X-HSE-DOOR-167B | T |
| X-HSE-DOOR-167 | T | | |

NOTE: The configuration for service water pumps in service was change to match the current plant configuration.

## 2.2 REFERENCES

2.2.1 EA-PSA-PSAR2c-06-10 r0, "Update of Palisades CDF Model - PSAR2b to PSAR2c".
2.2.2 EA-PSA-SAPHIRE-05-16 r0, "SAPHIRE v7.26 Validation and Verification".
2.2.3 SAPHIRE REFERENCE MANUAL, "SYSTEMS ANALYSIS PROGRAMS FOR HANDS ON INTEGRATED RELIABILITY EVALUATIONS (SAPHIRE) VERSION 6.0", Idaho National Engineering Laboratory, 1998.
2.2.4 SAPHIRE TECHNICAL REFERENCE, "Systems Analysis Program for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 6.0", Idaho National Engineering Laboratory, 1998.
2.2.5 EA-PSA-SAPHIRE-05-16 r0, "SAPHIRE v7.26 Validation and Verification". NUREG/CR-2300 volume 1, "PRA Procedures Guide".
2.2.6 NUREG-0492, "Fault Tree Handbook", 1981.
2.2.7 EA-PSA-CET-R1-04-21r0, "Conversion of IPE CET Models from CAFTA to SAPHIRE".
2.2.8 Nuclear Regulatory Commission (NRC) Inspection Manual, Part 9900: Technical Guidance, "Operations – Notices of Enforcement Discretion", February 7, 2005.
2.2.9 NUREG/CR-6850 (EPRI 1011989), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume2: Detailed Methodology".
2.2.10 Design Basis Document (DBD-7.08), "Plant Protection against Flooding", revision 5, 12/15/2004.
2.2.11 EA-PSA-PSAR2-04-02 r0, "Update of Palisades CDF Model - PSAR1B Modified w/HELB to PSAR2".
2.2.12 CPCo to NRC Letter, January 29, 1993, Palisades Plant Individual Plant Examination for Severe Accident Vulnerabilities (IPE), [F341/1523].
2.2.13 CPCo Letter to NRC, dated 6/30/95, Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events for Severe Accident Vulnerabilities (IPEEE), Final Report (G326/2290).
2.2.14 Palisades Letter for Submittal of the Revised Fire Analysis, dated May 31, 1996 (G700/0629).
2.2.15 EA-PSA-LERF-99-0020, "Re-Creation of Palisades IPE LERF Model".
2.2.16 Applicant's Environmental Report – Operating License Renewal Stage Palisades Nuclear Plant Nuclear Management Company, Docket No. 50-255, License No. DPR-20, March 2005.
2.2.17 Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, EPRI, Palo Alto, CA: 2008. 1016737.
2.2.18 ASME/American Nuclear Society, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa-2009, March 2009.
2.2.19 U.S. Regulatory Commission, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, NUREG-1855, Volume 1, Main Report, March 2009.
2.2.20 U.S. Nuclear Regulatory Commission, "CCF Parameter Estimations, 2007 Update", http://nrcoe.inl.gov/results/CCF/ParamEst2007/ccfparamest.htm, September 2008.

## 3.0 DEFINITIONS/ACRONYMNS

3.1 *AOT* – allowed outage time as defined in the Technical Specifications.
3.2 *CCDP* - conditional core damage probability, the core damage probability for a given plant initiating event for all potential accident initiating events in the PSA.
3.3 *containment bridge tree (CBT)* - containment system event tree that includes containment system fault trees such as containment air coolers, sprays, etc. The end states of the

containment bridge tree describe the state of various containment functions from the availability of sprays to the status of different PCS injection systems.

3.4 *containment event tree (CET)* - the non-system challenges or phenomenological threats to the containment are characterized in the containment event tree logic. This logic represents various issues from steam explosions to direct containment heating and the likelihood of such events challenging the containment structurally integrity. The plant damage state frequencies are input to the CET's.

3.5 *CDF* - core damage frequency, the calculated probability of a core damage event for any given year for all potential accident initiating events in the PSA.

3.6 *CDP* - core damage probability, the core damage probability for a specified time (i.e., 4 hours or 3 months) for all potential accident initiating events in the PSA, equal to the CDF times the specified length of time.

3.7 *CLERP* - conditional large early release probability, the large early release probability for a given plant initiating event for all potential accident initiating events in the PSA.

3.8 *LERF* - large early release frequency, the calculated probability of a significant radiological release to the public prior to completing emergency plan evacuation procedures following a core damage event for any given year for all potential accident initiating events in the PSA.

3.9 *LERP* - large early release probability, the large early release probability for a specified time (i.e., 4 hours or 3 months) for all potential accident initiating events in the PSA, equal to the LERF times the specified length of time.

3.10 *Level I (1)* - PSA studies that deterministically evaluate internal events and only core damage.

3.11 *Level II (2)* - PSA studies that deterministically evaluate internal events and core damage as well as the containment response that includes the probability of containment failure.

3.12 *model* - an approximate mathematical representation that simulates the behavior of a process, item, or concept (such as failure rate). For example, the probability of a system is synthesized using models that relate system failures to component failures and human errors.

3.13 *Notice of Enforcement Discretion (NOED)* - Is a document issued by Nuclear Regulatory Commission (NRC) to exercise enforcement discretion with regard to limiting condition for operation (LCO) in power reactor Technical Specifications (TS) or other license conditions.

3.14 *plant damage state* - it is not practical to perform detailed analysis of each core damage sequence. Therefore, the core damage sequences are grouped into bins that pose similar containment system challenges and result in like fission product releases. These product or bins are referred to as plant damage states. The plant damage state frequencies are input to the containment event tree.

3.15 *probabilistic risk assessment (PRA) or probabilistic safety analysis (PSA)* - a quantitative assessment of the risk associated with plant operation and maintenance. Risk is measured in terms of the frequency of occurrence of different events, including core damage. In general the scope of a PRA is divided into three categories: Level 1, 2, and 3. A Level 1 maps from initiating events to plant damage states (PDSs), including their aggregate, core damage. Level 2 includes Level 1 mapping from initiating events to release categories. Level 3 includes Level 2 and uses the release categories of Level 2 to quantify consequences, the most common of which are health effects and property damage in terms of cost. Full scope PRA includes internal and external events.

3.16 *Safety Basis* - Information typically provided by PRA personnel to justify that a requested NOED has no significant increase in radiological risk to the public.

3.17 *truncation limits* - the cutoff value of probability or frequency of individual accident sequences below which they are no longer retained in quantitative PRA model results. A truncation value is primarily used for the purpose of managing the size of the analysis results.

## 4.0 ASSUMPTIONS

### 4.1 MAJOR ASSUMPTIONS

4.1.1 The plant is assumed to be in either mode's 1, 2 or 3 as the initial condition prior to an event.

4.1.2 Use the zero maintenance PRA model to establish the plant's baseline risk and the estimated risk increase associated with the period of enforcement discretion (Reference 2.2.8). For the plant-specific configuration the plant intends to operate in during the period of enforcement discretion, the incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) will be quantified and compared with guidance thresholds of less than or equal to an ICCDP of 5E-7 and an ICLERP of 5E-8. These numerical guidance values are not pass-fail criteria.

4.1.3 It is considered a common cause stressor does not exist between P-7A and P-7C (with respect to the 9/29/09 experienced failure mode) beyond the existing common cause contribution in the model, given that this failure occurred subsequent to the replacement of P-7C in June of 2009 and that such a failure has never been experienced during the life of the plant. Moreover, validation that the shaft heat treatment procedure is different than the coupling heat treatment process (i.e., different vendor, different oven, different procedures, different personnel, different location etc.) has been demonstrated.

Nevertheless, the common cause failure probability of P-7A and P-7B to run was increased by a factor of ten to 2.132E-05. The current analysis of record applies a value of 2.132E-06. To provide a perspective regarding the current baseline value of 2.132E-06, the data employing the latest MGL data is presented below:

| | |
|---|---|
| SWS pump Fail to Run CCCG Size of 2 | β |
| Table 2.1.8.1 (Reference 2.2.20) | 1.17E-02 |
| | 2 Group |
| Common cause probabilities | Pf * β |
| CCF factor | 1.17E-02 |
| CCF failure probability | 1.11E-06 |

This information shows that the current baseline analysis (2.2.1) service water pump "pair" failure probability is about a factor of "2" greater than the latest NRC data.

Moreover, applying a factor of ten increase, the new 'pair' failure to run probability value is about a factor of 20 greater than the latest NRC data. This value is used in a sensitivity analysis described later in this evaluation.

### 4.2 MINOR ASSUMPTIONS

4.2.1 The Level 1 analysis applied a 1E-10 truncation limit. The Level II analysis applied a 1E-09 truncation limit.

The Palisades Level II analysis is a detailed assessment of containment performance. It is considerably more rigorous than the Owners Group simplified LERF methodology. Consequently to solve some 60,000 plant damage state sequences, a truncation limit of 1E-09 is employed. This is considered appropriate given the detail in the Palisades

plant damage state and containment event tree models. Moreover, the plant damage states are not subsumed resulting in a conservative aggregated result.

Basis: The Palisades Level II sequences analysis results (methods described in References 2.2.7, 2.2.15, 2.2.12, 2.2.15 and 2.2.16) do not subsume the correlated containment bridge tree sequences to the assigned sequence endstates. This is because the interface between the Level 1 and the Level 2 analyses is controlled by the Plant Damage State (PDS) Containment Bridge Tree (CBT). The core damage event tree sequences are binned according to the available six distinct containment safeguard system states. The result of combining the internal event initiators to the six containment safeguard categories results in some 181 plant damage states. The 181 endstates are then mapped to 23 containment event trees. Given the unique identification of these bins, Boolean subsuming cannot occur. The outcome is a conservative answer as the resultant release categories are overestimated on the order of 20 to 40%, typically.

## 5.0 METHODOLOGY

The methods employed to address the impact of extending service water pump (P-7C) allowed outage time (AOT) are described and include the SAPHIRE software and users manual (References 2.2.3 and 2.2.4) as well as the Nuclear Regulatory Commission (NRC) Inspection Manual, Part 9900 (Reference 2.2.8).

## 6.0 ANALYSIS

This section describes the specific analysis performed to analyze the safety significance and potential consequences of extending the P-7C LCO period.

### 6.1 VALIDATION OF THE CURRENT MODEL OF RECORD (PSAR2C)

The baseline results for the current model of record are;

Baseline results with current system alignment (at 1E-09 truncation):

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 2.611E-05 (non subsumed) | 2362 |
| End State Gather | 2.489E-05 (subsumed) | 1708 |

Validation of the model was completed by quantification with nominal maintenance unavailabilities to confirm that the stated results were duplicated. The results were correctly replicated.

### 6.2 MAINTENANCE CONDITION

The NOED guidance (Reference 2.2.8) requires the assessment to be performed based on a zero maintenance condition (all values assigned for the probability of equipment being removed from service set to 0). This condition is established by using the existing SAPHIRE change set (MAINT_UNVAIL(0)) which resets the indicated probabilities to zero (Attachment A). In order to assure adequate representation of the transformer out-of-service condition this calculation and the remaining risk calculations were conducted with a truncation value of 1.0E-10.

Baseline results with Maintenance Probabilities reset to zero:

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 2.727E-05 (non subsumed) | 9823 |
| End State Gather | 2.591E-05 (subsumed) | 7745 |

The model includes a change set file for the configuration of equipment assumed to be in-service or standby at the time of an event. The assumed conditions represent an arbitrary choice of system/train alignments expected to be in place for the normal at-power condition. None of the alignments made with this SAPHIRE change set impact the assessment of the service water pump P-7C out-of-service configuration. The change set was modified to represent the current condition in which service water pumps P-7A and P-7B are the in-service pumps and P-7C is the standby pump.

The model includes the service water pumps as the primary source of cooling to components on the critical and non-critical service water distribution headers. In addition, service water to the containment air coolers from the critical service water header is also explicitly modeled. For events with reduced service water capacity (one or more service water pumps unavailable, service water to the non-critical header or to the containment air coolers can be isolated by the operators to reduce service water loads. For events which would result in the generation of a safety injection signal (SIS) or containment high pressure (CHP) the non-critical header would automatically be isolated via closure of the service water control valve to the non-critical header (CV-1359).

The analysis includes an operator action to perform the isolation of service water to containment. Modeling of isolation of the non-critical header only includes the automatic signal to close the valve. Operator action to close the valve is possible but not included in the current model.

The zero maintenance case was re-quantified with the service water pump (P-7C) out-of-service. The results of this case are shown in the following table.
Zero maintenance conditions with service water pump (P-7C) OOS:

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 2.732E-05 (non subsumed) | 10010 |
| End State Gather | 2.596E-05 (subsumed) | 7859 |

## 6.3 INITIATING EVENT FREQUENCY CONSIDERATION

The current PRA model includes a Loss of Service Water initiating event frequency of 1.22E-03. In the current condition the plant is more susceptible to perturbations in the operation of the service water system. Consequently the initiating event frequency for loss of the service water system would be increased during the period. An additional analysis was completed with the loss of service water initiating event frequency increased by an order of magnitude.

Zero maintenance conditions with service water pump (P-7C) OOS and increased initiating event frequency (IE_LOSWS):

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 5.729E-05 (non subsumed) | 10932 |
| End State Gather | 5.555E-05 (subsumed) | 8477 |

## 6.4 INTERNAL EVENT CUTSET REVIEW

A review of the change to the cutsets contributing to core damage as a result of the changes made to represent the removal of service water pump (P-7C) from service determined that there were no changes to the top 100 cutsets. A review of the changed cutsets contributing to core damage as a result of the changes made to the initiating event frequency for a loss of service water initiating event are discussed below.

The top 100 cutsets represent ~ 83% of the increased core damage probability. Thirteen cutsets showing an increased contribution to core damage are described below. The top 100 cutsets are listed in Attachment A. Eighty seven out of one hundred of the listed cutsets did not change.

Cutset 1 Loss of Service Water (Sequence 22-2)

Cutset 1 is the same cutset as the baseline (0 maintenance) case with an increased contribution to core damage as it is the result of a loss of service water initiating event. The cutset represents a loss of primary coolant pump seal cooling and the failure to trip the primary coolant pump(s) in time to prevent seal failure that results in a loss of coolant accident. The loss of service water fails injection pumps due to loss of cooling and containment heat removal.

Cutset 4 Loss of Service Water (Sequence 21-5)

This cutset represents a loss of service water, the loss of service water results in loss of primary coolant pump seal cooling and a consequential seal LOCA due to failure to trip the primary coolant pumps. The loss of service water fails injection pumps due to loss of cooling and containment heat removal

Cutset 9 Loss of Service Water (Sequence 17)

This cutset represents a loss of service water event with failure secondary heat removal via the steam generator, successful initiation of once through cooling (OTC) and failure of the containment heat removal, failure of main feedwater and low pressure feed (feeding steam generators with condensate pumps) due to loss of condenser vacuum, and failure of containment sprays and containment air coolers as a results of the loss of service water cooling to remove heat from the systems. The failure of the auxiliary feedwater system is due to common cause failure of all three pumps to start.

Cutset 19 Loss of Service Water (Sequence 17)

This cutset is similar to cutset 9 above with the difference being the failure of auxiliary feedwater is due to common cause failure of all the pump discharge check valves. The remainder of the cutset is the same as cutset 9.

Cutset 21 Loss of Service Water (Sequence 5)

Cutset 21 is also similar to cutsets 9 and 19. The difference in this cutset is that the failure of auxiliary feedwater is a long term failure to provide an alternate suction source to the auxiliary feedwater pumps. Failure of normal makeup to the condensate storage tank (T-2) is due to failure of the demineralized water transfer pump (P-936) to provide makeup from demineralized water storage tank (T-939). Operators would be aware of the failure of normal makeup when a low level alarm occurs at 73% level in the condensate storage tank. The operator would then have several hours to align an alternate source to the auxiliary feedwater pumps. This cutset includes failure of an operator action to align

service water to pumps to auxiliary feedwater P-8A or P-8B OR fire protection water to auxiliary feedwater pump P-8C. This cutset does not credit the availability of water from primary system makeup storage tank (T-81) via pumped or gravity feed which would provide additional time to align other water sources.

Cutset 23 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above with the difference being the failure of auxiliary feedwater is due to common cause failure of all the check valves in the flow headers from the pump trains to the steam generators. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 28 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above with the difference being the failure of auxiliary feedwater is due to common cause failure of all four flow control valves in the flow headers from the pump trains to the steam generators. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 35 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above with the difference being the failure of auxiliary feedwater is due to spurious low suction trips of auxiliary feedwater pumps P-8A and P-8C and failure of the turbine-driven auxiliary feedwater pump P-8B. The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 36 Loss of Service Water (Sequence 5)

Cutset 36 is similar to cutset 21 (see above). Loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of demineralized water transfer pump (P-936). The difference between this cutset and cutset 21 is that the long term failure is the failure of another operator action related to the alignment of an alternate suction source to the auxiliary feedwater pumps after the contents of the condensate storage tank (T-2) have been depleted.

Cutset 37 Loss of Service Water (Sequence 17)

This cutset is similar to cutsets 9 and 19 above with the difference being the failure of auxiliary feedwater is due to common cause failure of all three auxiliary feedwater pumps to run for the mission time (24 hours). The remainder of the cutset is the same as cutsets 9 and 19.

Cutset 49 Loss of Service Water (Sequence 5)

Cutset 49 is similar to cutset 21 and 36 (see above). Loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the control valve (CV-2010) to automatically open and all flow from the demineralized water storage tank (T-939) to the condensate storage tank (T-2). Additionally the cutset includes failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 60 Loss of Service Water (Sequence 5)

Cutset 60 is also similar to cutset 21 and 36 (see above). In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to loss of the air supply (filter plugging) to the control valve (CV-2010). The cutset includes the failure of the

operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 62 Loss of Service Water (Sequence 5)

Cutset 62 is also similar to cutset 21 and 36 (see above). In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The cutset includes the failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

Cutset 69 Loss of Service Water (Sequence 5)

Cutset 69 is also similar to cutset 21 and 36 (see above). In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The long term failure of the alignment of service water or fire protection water to the auxiliary feedwater pump suction is due to failure of one of the manual valves (MV-FW775) required to align fire protection water to pump P8C (service water to pumps P-8A and P-8B is failed by the initiator).

Cutset 70 Loss of Service Water (Sequence 5)

Cutset 70 is also similar to cutset 21 and 36 (see above). In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to failure of the transfer pump (P-936). The long term failure of the alignment of service water or fire protection water to the auxiliary feedwater pump suction is due to failure of one of the manual valves (MV-FW774) required to align fire protection water to pump P8C (service water to pumps P-8A and P-8B is failed by the initiator).

Cutset 75 Loss of Service Water (Sequence 5)

Cutset 75 is also similar to cutset 21 and 36 (see above). In this cutset the loss of normal makeup from the demineralized water storage tank (T-939) is due to loss of the air supply (filter plugging) to the control valve (CV-2010). The cutset includes the failure of the operator to align an alternate suction source to the operating auxiliary feedwater pump.

## 6.5 LARGE EARLY RELEASE FREQUENCY (LERF)

The Palisades Level II assessment included re-evaluating the containment plant damage states assuming no maintenance unavailability, similar to core damage evaluation above.

Next, the zero maintenance case was quantified with service water pump P-7C out-of-service. The resulting set of endstate frequencies were mapped to 23 containment event trees (CET). The CETs represent the non-system challenges or phenomenological threats to the containment. This logic represents various issues from steam explosions to direct containment heating and the likelihood of such events challenging the containment. The outputs of the CETs are mapped to endstates that characterize the timing of the release (Timing Bins) and the magnitude (Release Magnitude Bins). The LERF results are considered bounding for the external events results (seismic and fire) as well as the internal events analysis.

Timing Bins

Three timing classifications are used, as follows:

1. Early (E) - less than 4 hours from accident initiation

2. Intermediate (I) - greater than or equal to 4 hours, but less than 24 hours

3. Late (L) - greater than or equal to 24 hours.

The definition of the categories is based upon past experience with offsite responses:

0-4 hours is based on a Palisades plant specific analysis discussed in the following section.

4-24 hours is a time frame in which most of the offsite nuclear plant protective measures can be accomplished.

>24 hours are times at which the offsite measures can be assumed to be fully effective.

Release Magnitude Bins

The four severity classifications associated with volatile or particulate releases are defined as follows:

High (H) - A radionuclide release of sufficient magnitude to cause near-term health effects.

Moderate (M) - A radionuclide release with the potential for latent health effects.

Low (L) - A radionuclide release with the potential for minor health effects.

Low-Low (LL) - A radionuclide release that is less than or equal to the containment design base leakage resulting in ***no*** health effects.

A LERF release category equates to a Palisades CET E-H release category.

## 7.0 RESULTS

This section reports the quantitative and qualitative results.

### 7.1 INTERNAL EVENT INCREMENTAL CONDITIONAL CORE DAMAGE PROBABILITY (ICCDP)

The zero maintenance case was quantified with service water pump P-7C out-of-service (refer to Attachment A for the SAPHIRE change set information). The results of the quantification under these conditions are shown in the following table.

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 2.73E-05 (non subsumed) | 9823 |
| End State Gather | 2.59E-05 (subsumed) | 7745 |

Results with service water pump P-7C out of service are shown below:

| | CDF | # Cutsets |
|---|---|---|
| Sequence | 2.73E-05 (non subsumed) | 10010 |
| End State Gather | 2.60E-05 (subsumed) | 7859 |

Results with service water pump P-7C out of service and increasing the initiating event frequency for a loss of service water event are shown below:

| | CDF | # Cutsets |
|---|---|---|

| | | |
|---|---|---|
| Sequence | 5.73E-05 (non subsumed) | 10932 |
| End State Gather | 5.56E-05 (subsumed) | 8477 |

Removing service water pump P-7C from service results in an increase in CDF of 5.0E-08/yr or 5.71E-12/hr.

(2.60E-05 – 2.59E-05)

(5.00E-08/yr/(365days/yr*24hrs/day))

The current allowed outage time (72 hours) represents a Core Damage Probability (CDP) of 4.11E-10 for 72 hours.

5.71E-12/hr*3days*24hrs/day

The CDP associated with an extension of the current allowed outage time is 9.59E-10 for 7 days (168 hours).

5.71E-12/hr*7days*24hrs/day

This results in an Incremental Conditional Core Damage Probability (ICCDP) of 5.48E-10 (9.59E-10 – 4.11E-10) for the extension to a 7 day period.

Removing service water pump P-7C from service and considering impacts of an increase in the loss of offsite power initiating event frequency results in an increase in CDF of 2.96E-05/yr or 3.38E-09/hr.

(5.56E-05 – 2.59E-05)

2.96E-05/yr/(365days/yr*24hrs/day)

The current allowed outage time (72 hours) represents a Core Damage Probability (CDP) of 2.44E-07 for 72 hours.

3.38E-09/hr*3days*24hrs/day

The CDP associated with an extension of the current allowed outage time is 5.68E-07 for 7 days (168 hours).

3.38E-09/hr*7days*24hrs/day

This results in an Incremental Conditional Core Damage Probability (ICCDP) of 3.25E-07 (5.68E-07 – 2.44E-07) for the extension to a 7 day period.

Removing service water pump P-7C from service, considering impacts of an increase in the loss of offsite power initiating event frequency and an increase in the common cause failure of the operating service water pumps to fail to continue to run results (Major Assumption 4.1.3 - 2.132E-05) in an increase in CDF of 2.96E-05/yr or 3.39E-09/hr.

(5.56E-05 – 2.59E-05)

2.97E-05/yr/(365days/yr*24hrs/day)

The current allowed outage time (72 hours) represents a Core Damage Probability (CDP) of 2.44E-07 for 72 hours.

3.39E-09/hr*3days*24hrs/day

The CDP associated with an extension of the current allowed outage time is 3.26E-07 for 24 hours.

3.39E-09/hr*4days*24hrs/day

This results in an Incremental Conditional Core Damage Probability (ICCDP) of 8.16E-08 (3.26E-07 – 2.44E-07) for the extension to a 7 day period.

## 7.2 INTERNAL EVENTS INCREMENTAL CONDITIONAL LARGE EARLY RELEASE PROBABILITY (ICLERP)

As was the case above, for the core damage analysis, the baseline plant damage analysis was first evaluated by quantifying the plant damage states and then mapping the results to the 23 CETs. For example, for the failed P-7C analysis the following CET frequencies were determined:

| CET | Frequency /yr | %Contribution |
|---|---|---|
| CET-DEJP | 7.61E-06 | 23.5% |
| CET-ZEGP | 5.37E-06 | 16.6% |
| CET-DEJS | 4.41E-06 | 13.6% |
| CET-BEGP | 4.04E-06 | 12.5% |
| CET-A2EGR | 3.00E-06 | 9.3% |
| CET-BEGR | 2.56E-06 | 7.9% |
| CET-TEJW | 1.42E-06 | 4.4% |
| CET-BEGV | 1.36E-06 | 4.2% |
| CET-TEJP | 8.21E-07 | 2.5% |
| CET-TEJS | 4.44E-07 | 1.4% |
| CET-TEJQ | 4.55E-07 | 1.4% |
| CET-BEGS | 3.95E-07 | 1.2% |
| CET-A2EGP | 3.68E-07 | 1.1% |
| CET-TEJR | 3.60E-08 | 0.1% |
| CET-DEJR | 7.20E-08 | 0.2% |
| CET-MEJW | 6.13E-09 | 0.0% |
| CET-A1EGR | 5.15E-09 | 0.0% |
| CET-TEJV | 2.38E-09 | 0.0% |
| CET-MEJP | 0.00E+00 | 0.0% |
| CET-MEJV | 0.00E+00 | 0.0% |
| CET-MEJR | 0.00E+00 | 0.0% |
| CET-MEJS | 0.00E+00 | 0.0% |
| CET-MEJQ | 0.00E+00 | 0.0% |
| | 3.24E-05 | 100.0% |

These frequencies are then input to the containment phenomenological event trees resulting in the following releases:

| E-H (LERF) /yr | E-M /yr | I-H /yr | I-M /yr | L-L /yr | L-LL /yr | Plant Damage State Summary /yr |
|---|---|---|---|---|---|---|
| 2.70E-07 | 5.70E-06 | 4.05E-06 | 8.22E-06 | 6.80E-07 | 1.00E-05 | 2.89E-05 |

And similarly, the case was quantified with service water pump P-7C out-of-service. Again, the results of the 181 plant damage bins were mapped to the 23 CETs resulting in the following set of release frequencies:

| E-H (LERF) /yr | E-M /yr | I-H /yr | I-M /yr | L-L /yr | L-LL /yr | Plant Damage State Summary /yr |
|---|---|---|---|---|---|---|
| 2.73E-07 | 5.73E-06 | 4.05E-06 | 8.48E-06 | 7.58E-07 | 1.01E-05 | 2.94E-05 |

The % change in different release categories is shown in the following table:

| Endstate | Increase (%) |
|---|---|
| E-H | 1.0% |
| E-M | 0.6% |
| I-H | 0.0% |
| I-M | 3.2% |
| L-L | 11.5% |
| L-LL | 0.8% |

Removing service water pump P-7C from service results in an increase in LERF of 2.80E-09/yr or 3.20E-13/hr.

(2.73E-07 – 2.70E-07)

2.80E-08/yr/(365days/yr*24hrs/day)

The current allowed outage time (72 hours) represents a Core Damage Probability (LERP) of 2.3E-11 for 72 hours.

3.2E-13/hr*3days*24hrs/day

The LERP associated with an extension of the current allowed outage time is 5.37E-11 for 7 days (168 hours).

3.2E-13/hr*7days*24hrs/day

This results in an Incremental Conditional Large Early Release Probability (ICLERP) of 3.07E-11 (5.37E-11 – 2.3E-11) for the extension to a 7 day period. It is considered that the experienced small change in the internal events ICLERP value apply to the external events evaluations as well.

Attachment C provides the SAPHIRE Change Sets for both the plant damage state analysis and the CET evaluation.

## 7.3 EXTERNAL EVENTS – SEISMIC

In the Palisades IPEEE (Individual Plant Examination of External Events) (References 2.2.17 and 2.2.18) a seismic risk assessment was used to assess risks due to seismic events. The risk assessment was a hybrid of the conventional PSA and seismic margins analysis.

The service water system modeling used in the external events analysis is the same

model used for internal events analysis. The same system success criteria were also used. The component random failures rates that were used in the IPE (Individual Plant Examination of Internal Events) (Reference 2.2.16) were also used in the SPRA (Seismic Probabilistic Risk Assessment). No adjustments to these probabilities were made. The seismic impact on these components was assessed by including seismic basic events and fragilities. The component fragilities that were identified in Section 3.5.2 of the IPEEE reports were used in the SPRA. The fragilities were input as a median capacity with a lognormal standard deviation (beta), which defined a lognormal fragility curve.

In addition to the seismic basic events, the seismic fault trees were modified to include seismically induced initiating events. The four seismic event tree headings that are seismically induced initiating events are: TBFR (Turbine Building Fire); TBFL (Turbine Building Flood); LOOP (Loss of Offsite Power); and SBL (Small Break Loss of Coolant Accident). All events that are affected by a turbine building fire have an associated basic event of TBFR. All basic events that are affected by a turbine building flood have an associated basic event of TBFL. The affected off-site power related equipment received an associated basic event of LOOP. The initiating event SBLOCA (Small Break Loss of Coolant Accident) was given to all sequences that were quantified by the SBLOCA event tree and was not included in the fault tree as a basic event.

The seismic analysis has not been updated since originally developed for the Individual Plant Examination of External Events (IPEEE) submittal. A review of the results of the IPEEE submittal indicated that the core damage frequency was 8.88E-06 with a high confidence low probability of failure (HCLPF) of 0.217g PGA (peak ground acceleration). There were no specific seismic events identified as dominant contributors to the core damage frequency. Important seismic induced failures identified were; the Fire Protection System, Main Steam Isolation Valves, Diesel Generator Fuel Oil Supply, and an under voltage relay for 2400 volt ac Bus 1D. Several important random failures were identified in the report as important because of their contribution in combination with seismically induced failures. The important random failures (not seismically induced) identified in the report were: diesel generator 1-2, auxiliary feedwater (AFW) pump P-8C, and atmospheric dump valves.

The service water system was determined to be seismically rugged and there were no significant contributions of the service water system to core damage resulting from seismically induced failures. Random failures of the service water system were identified as important contributors as a consequence of seismically induced failure of other system components as discussed below.

As noted, the fire protection system is an important contributor to seismic analysis due to the probability of seismically induced failure of fire protection system components and the condensate storage tank (CST). Seismically induced failure of the condensate storage tank results in an earlier need for alignment of an alternate suction source for the operating auxiliary feedwater pump. The fire protection system provides an alternate suction source to AFW pumps P-8A and P-8B. The seismically induced failures of the fire protection system result in long term failure of auxiliary feedwater pumps P-8A and P-8B due to the unavailability of a suction source. Auxiliary feedwater pump P-8C is important to long term makeup to the steam generators should the fire system become unavailable following a seismic event (as discussed in the results for Accident Classes IA & IB, Section 3.6.5.3.1 of the IPEEE report). The fire protection system has a low fragility and is a significant contributor to seismic risk once the contents of the condensate storage tank (T-2) are depleted and a long term suction source is required for continued operation of the AFW pumps. The seismically induced failure of the fire protection system represents a higher probability of failure of the long term suction to motor-driven auxiliary feedwater pump P-8A and turbine-driven auxiliary feedwater pump P-8B after

the depletion of the available tank T-2 inventory. This increased probability of failure of heat removal via the A and B pump trains results in an increased importance of motor-driven auxiliary feedwater pump P-8C. The importance of pump P-8C is a consequence of the fact that service water (a much more seismically rugged system) is more likely to remain available as a long term suction source to pump P-8C.

Auxiliary Feedwater (AFW) flow requirements in the PRA are 165 gpm to either steam generator. These flow requirements are a small fraction (<2%) of the total flow (8000 gpm) from a single service water pump. At the time of condensate storage tank depletion the flow requirements will be lower. Therefore the PRA model assumes no additional service water pumps are required to be placed in service to provide a suction source for the AFW pumps.

The contribution to core damage from seismic events determined in the IPEEE was 8.88E-06. This represents approximately 13% of the total core damage frequency from the current internal events analysis (2.49E-05), fire (3.3.1E-05), flooding (~<2E-07) and seismic (8.88E-06). Therefore the expected seismic contribution is bounded by the internal events core damage assessment described in this letter.

## 7.4 EXTERNAL EVENTS – FIRE

The Palisades fire analysis used an approach that combined the deterministic evaluation techniques from the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology with classical PRA techniques. The FIVE methodology was used to establish fire boundaries and to evaluate the probability and the timing of damage to components located in a fire area/zone involved in a fire. Based on the results from implementing the FIVE methodology PRA techniques were then employed to determine the probability of core damage associated with fires within the identified fire areas/zones. Fire areas identified by the Fire Protection Program were used as the basis of the fire areas evaluated by the fire risk analysis. These fire areas were evaluated for further division based on combustible loading and fire-spread potential to identify fire zones within fire areas. The fire areas/zones identified were evaluated and quantified using the fault trees and transient event tree from the IPE. The fault and event trees were modified to accurately reflect the fire analysis.

The core damage frequency contribution from internal fires for Palisades is 3.31E-05/yr. The dominant contribution to the fire CDF (>89%) is related to five fire areas: cable spreading room (33.5%); main control room (24.4%); 1D switchgear room (14.7%); turbine building (9.3%); and 1C switchgear room (7.6%).

The principle finding of the fire analysis was that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures (unrelated to damage caused by the fire) or human errors, core damage will not occur. As a result, the study concluded that there are no major vulnerabilities due to fire events at the Palisades Nuclear Power Plant. This is primarily due to the fact that the damage in the important fire areas was to support systems (e.g. ac power or dc power) that resulted in the loss of one division of equipment with adequate equipment unaffected on the other division. During the pump P-7C repair an operable Service Water pump will remain available on each division.

## 7.5 EXTERNAL EVENTS – FLOODING AND OTHER

Other external events (high winds, external floods, transportation, etc.) were screened by demonstrating conformance to the 1975 Standard Review Plan using prior evaluations completed during the Systematic Evaluation Program (SEP) or demonstrating low hazard frequency for aircraft hazards. There were no significant contributors to core damage

frequency from other external events (other than seismic and fire) identified.

## 7.6 UNCERTAINTY EVALUATION

EPRI 1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments [2.2.17] was employed to characterize the uncertainty in the current analysis of record. The results of this assessment are correlated to the associated supporting requirements as provided in ASME/ANS PRA Standard [2.2.18] and to assess potential key sources of model uncertainty relevant to specific applications as described in NUREG-1855 [2.2.19].

Definitions

The following definitions have been provided in EPRI 1016737 and NUREG-1855.

An assumption is a decision or judgment that is made in the development of the PRA model. An assumption is either related to a source of model uncertainty or is related to scope or level of detail.

An assumption related to a model uncertainty is made with the knowledge that a different reasonable alternative assumption exists. A reasonable alternative assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made. It should be noted that "reasonable alternative assumptions" related to sources of model uncertainty can lead to increases or decreases in the calculated risk metrics.

An assumption related to scope or level of detail is one that is made for modeling convenience.

A consensus model, in the most general sense, [is] a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that the NRC has utilized or accepted for the specific risk-informed application for which it is proposed.

A source of model uncertainty is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g. introduction of a new basic event, changes to basic event probabilities, changes in success criterion, introduction of a new initiating event).

A source of model uncertainty is labeled key when it could impact the PRA results that are being used in a decision, and consequently, may influence the decision being made. Therefore, a key source of modeling uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance [guidelines] are met, and therefore, could potentially influence the decision. For example, for an application for a licensing base change using the acceptance [guidelines] of RG 1.174, a source of model uncertainty or related assumption could be considered "key" if it results in uncertainty regarding whether the results lie in Region II or Region I, or if it results in uncertainty regarding whether the result becomes close to the region boundary or not.

These definitions delineate those sources of model uncertainty (and related assumptions) that should be the focus for meeting the QU supporting requirements in the standard as modified by RG-1.200, Revision 1 clarifications, including:

- QU-E1: IDENTIFY sources of model uncertainty.
- QU-E2: IDENTIFY assumptions made in the development of the PRA model.
- QU-E4: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).
- QU-F4: DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in QU-E4).

Other related supporting requirements that are addressed by this appendix include:

- LE-F3: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, consistent with the requirements of Tables 2.2.7-2(d) and 2.2.7-2(e).

- IE-D3, AS-C3, SC-C3, SY-C3, HR-I3, DA-E3, LE-G4, IFPP-B3, IFSO-B3, IFSN-B3, IFEV-B3, and IFQU-B3: DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2 [or LE F3]) associated with ...[each element].

Attachment D, Table D-1 summarizes the findings from the implementation of the process for characterizing the sources of model uncertainty for the current Palisades analysis of record.

## 7.7 REG GUIDE 1.200 GAP ANALYSIS

At the behest of the NRC, the industry undertook a task to develop a consensus standard on the technical adequacy of PRAs for regulatory applications. This effort resulted in publication of ASME RA-S-2002. Concurrently, under the direction of the Nuclear Energy Institute (NEI) and the Owners Groups for each major reactor provider, peer reviews of PRAs were conducted using the guidance in NEI 00-02. The NRC was also concurrently developing guidance for determining the adequacy of risk analyses for use in regulatory applications. The first draft of this guidance was published as Draft Guide 1122 (DG 1122) in September 2002. Following interactions with industry in subsequent years as the ASME Standard was being modified, the NRC recently published DG 1161 in September 2006. This draft version of Regulatory Guide 1.200 (RG 1.200) provides guidance on self assessments to determine the adequacy of PRAs.

Subsequent to the industry peer review of the Palisades PRA; a self assessment (Gap Analysis) was performed. This analysis reviewed the peer review facts against the guidance in DG 1122 and produced a list of recommended actions to address "gaps" between the results of the peer review and the guidance in DG 1122. Palisades has subsequently addressed all A and B level facts and observations (F&Os) from the peer review certification report. DG 1122 allowed for two mechanisms for conducting a self assessment. One was a direct comparison of the PRA against the Standard with additional considerations cited by the NRC to address areas where the NRC did not agree with the Standard (Table A-1 of DG 1122). The other method was to take advantage of the peer review findings and perform additional reviews against the Standard in areas where the NRC found that the peer review process needed additional effort to address NRC concerns with the Standard. The NRC issues were documented in Table B-4 of DG 1122. This was the method used in the Palisades Gap Analysis.

In general, the additional recommendations addressed issues of documentation and/or justification for technical analyses in the PRA. Slightly less than half of the additional recommendations are likely to result in a change to the actual model. Only three additional recommendations were considered likely to result in a noticeable change in the CDF or LERF. These included the removal of EDG repair from the model, the inclusion of additional flow diversion paths for key systems, and the inclusion of potential concurrent unavailabilities (such as train wise maintenance schedules where one train in multiple systems is taken out of service at the same time. The risk impact of the latter issue is bounded by the risk evaluations done to adhere to the a(4) requirements of the Maintenance Rule (10CFR50.69), but may be more significant in the baseline CDF evaluations.

EDG Repair Model

Removal of the EDG repair model does not affect the conclusions of this analysis given the available of the non-safety related diesel.

Flow Diversion

The flow diversion analysis performed to support the PRA update from the current analysis of record (2.2.1) used a combination of qualitative evaluation and detailed hydraulic analyses to identify possible flow diversions in the following systems:

- Service Water (SWS)
- Low Pressure Safety Injection / Shutdown Cooling (LPSI)
- Chemical and Volume Control System (CVCS)
- Component Cooling (CCW)
- Containment Spray (CSS)
- High Pressure Safety Injection (HPSI)
- Auxiliary Feed water (AFW)

The result of the analysis is a series of annotated P&ID's which illustrate the flow diversions that were considered with accompanying documentation describing the evaluations performed for each (345 possible flow diversions were documented) assessed path. Analyses considered both single and multiple failures of equipment under various system configurations and transient events. In cases where a qualitative evaluation was indeterminate, detailed analyses were performed using Pipe-Flo Professional 2007a and GOTHIC.

The results of this evaluation do not affect the conclusions of this study regarding P-7C.

Coincident Unavailabilities

Coincident unavailability is associated with maintenance for redundant equipment, both intra- system and inter- system. Coincident unavailability is a result of a planned, repetitive activity and can arise for systems with installed spares.

To evaluate coincident unavailability, all the unavailability data was compiled, and coincident events were marked for each train. In addition to reviewing the maintenance rule unavailability data for coincident unavailability, the risk management work week reviews from the LAN were also downloaded and reviewed.

The following identifies the equipment associated with each train:

- Train A equipment: C-2A & C-2C, C-6B, ED-15 & ED-17, K-6A, P-52C, P-54B & P-54C, P-55C, P-56A, P-66B, P-67B, P-7B, P-8A & P-8B, and PRV-1042.
- Train B equipment: C-2B, C-6A, ED-16 & ED-18, K-6B, P-52B, P-54A, P-55A & P-55B, P-56B, P-66A, P-67A, P-7A & 7C, P-8C and PRV-1043.

Plant experience showed that in most cases only one piece of equipment from a train is removed from service at a time. A review of the three plus years of unavailability data showed that there was limited, repetitive coincident unavailability; most cases involved only two components, and occurred only once in the three year data window.

There were, however, a few cases in which plant experience showed that two components from the same train were recurrently removed from service at the same time. In these cases, coincident unavailability was modeled; the following identifies the combinations of equipment for coincident unavailability:

1. P-54B and P-66B;
2. P-54B and P-67B;
3. P-54C and P-67B;
4. P-8A and P-8B;
5. P-54A and P-66A; and
6. P-54A and P-67A.

Coincident unavailability included only the time that both components were simultaneously unavailable. If one component was unavailable for an extra hour, the hour was used in the individual unavailability. This analysis is included in the planned update to the current analysis of record (2.2.1). The results of this assessment do not affect the conclusions of the P-7C analyses.

Summary

The resolution of these issues as well as other model updates including HRA, component data, initiating event data, common cause logic and data, logic model changes to support NFPA-805, simplified LERF analyses, additional uncertainty analysis, updated internal event flooding etc. are being incorporated into the soon-to-be released model update. It is considered that these changes do not affect the conclusions of this analysis.

## 8.0 CONCLUSIONS

The internal events core damage analysis calculated an Incremental Conditional Core Damage Probability (ICCDP) of 3.25E-07 (5.68E-07 – 2.44E-07) for allowed outage time extension to a 7 day period for service water pump P-7C when consideration of a increase in the loss of service water initiating event frequency is included. Without the increase in initiating event frequency the internal events core damage analysis calculated an Incremental Conditional Core Damage Probability (ICCDP) of 5.48E-10 allowed outage time extension to a 7 day period for service water pump P-7C. The internal events analysis is considered bounding for the evaluated external events including fire,

flood and seismic. Moreover, the calculated ICLERP was conservatively estimated to be 3.07E-11.

Therefore extending the present LCO duration for an additional 4 days, results in a change in risk that is less than the prescribed limit in the Nuclear Regulatory Commission (NRC) Inspection Manual, Part 9900 guidance thresholds of less an ICCDP of 5E-7 and an ICLERP of 5E-8.

Completing the rebuild of the pump within an additional 24 hours beyond the current 72 hour limit versus and addition 4 days results in an Incremental Conditional Core Damage Probability (ICCDP) of 8.16E-08 when considering an order of magnitude increase in the loss of service water initiating event frequency and a factor of 10 increase (relative to the current analysis of record) in the common cause failure of P-7A and P-7B to run or 1.37E-10 when only the pump out of service condition is considered. The Incremental Conditional Large Early Release Probability (ICLERP) would be reduced to 7.7E-12 for an additional 24 hour period.

SAPHIRE CDF Change Set Data and Results

**Table of Contents**

SAPHIRE CDF Change Set Data and Results

## SAPHIRE Zero Maintenance Unavailabilities Change Set

```
* PROBABILITY HEADER
* Name          , CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT,
Flag, UncValue2
* CLASS HEADER
* Name,  Group, CompType, CompId, System, Location, FailMode, Train, Init, Att1,..,Att16
* CLASS PROBABILITY HEADER
* CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT, Flag, UncValue2
PSAR2C, MAINT_UNAVAIL(0)          =
^PROBABILITY
A-PMOO-P-8A             , 1, , 0.000E+000, , , , , , ,
A-PMOO-P-8B             , 1, , 0.000E+000, , , , , , ,
A-PMOO-P-8C             , 1, , 0.000E+000, , , , , , ,
C-PMOO-P-52A            , 1, , 0.000E+000, , , , , , ,
C-PMOO-P-52B            , 1, , 0.000E+000, , , , , , ,
C-PMOO-P-52C            , 1, , 0.000E+000, , , , , , ,
D-BCOO-ED-15            , 1, , 0.000E+000, , , , , , ,
D-BCOO-ED-16            , 1, , 0.000E+000, , , , , , ,
D-BCOO-ED-17            , 1, , 0.000E+000, , , , , , ,
D-BCOO-ED-18            , 1, , 0.000E+000, , , , , , ,
E-DGOO-K-6A             , 1, , 0.000E+000, , , , , , ,
E-DGOO-K-6B             , 1, , 0.000E+000, , , , , , ,
E-PMOO-P-18A            , 1, , 0.000E+000, , , , , , ,
F-PMOO-P-41             , 1, , 0.000E+000, , , , , , ,
F-PMOO-P-9A             , 1, , 0.000E+000, , , , , , ,
F-PMOO-P-9B             , 1, , 0.000E+000, , , , , , ,
G-PMOO-P-55A            , 1, , 0.000E+000, , , , , , ,
G-PMOO-P-55B            , 1, , 0.000E+000, , , , , , ,
G-PMOO-P-55C            , 1, , 0.000E+000, , , , , , ,
G-PMOO-P-56A            , 1, , 0.000E+000, , , , , , ,
G-PMOO-P-56B            , 1, , 0.000E+000, , , , , , ,
H-PMOO-P-66A            , 1, , 0.000E+000, , , , , , ,
H-PMOO-P-66B            , 1, , 0.000E+000, , , , , , ,
I-ADOO-M-2              , 1, , 0.000E+000, , , , , , ,
I-ADOO-M-2-1            , 1, , 0.000E+000, , , , , , ,
I-ADOO-M-2-2            , 1, , 0.000E+000, , , , , , ,
I-CMOO-C-2A             , 1, , 0.000E+000, , , , , , ,
I-CMOO-C-2B             , 1, , 0.000E+000, , , , , , ,
I-CMOO-C-2C             , 1, , 0.000E+000, , , , , , ,
L-PMOO-P-67A            , 1, , 0.000E+000, , , , , , ,
L-PMOO-P-67B            , 1, , 0.000E+000, , , , , , ,
P-BSOO-F-BUS            , 1, , 0.000E+000, , , , , , ,
P-BSOO-R-BUS            , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB25R8          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB27F7          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB27H9          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB27R8          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB29F7          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB29H9          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB29R8          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB31F7          , 1, , 0.000E+000, , , , , , ,
P-CBOO-ABB31H9          , 1, , 0.000E+000, , , , , , ,
Q-CMOO-C-6A             , 1, , 0.000E+000, , , , , , ,
Q-CMOO-C-6B             , 1, , 0.000E+000, , , , , , ,
Q-CMOO-C-6C             , 1, , 0.000E+000, , , , , , ,
S-PMOO-P-54A            , 1, , 0.000E+000, , , , , , ,
S-PMOO-P-54B            , 1, , 0.000E+000, , , , , , ,
S-PMOO-P-54C            , 1, , 0.000E+000, , , , , , ,
U-PMOO-P-7A             , 1, , 0.000E+000, , , , , , ,
U-PMOO-P-7B             , 1, , 0.000E+000, , , , , , ,
U-PMOO-P-7C             , 1, , 0.000E+000, , , , , , ,
V-FNOO-V-1A             , 1, , 0.000E+000, , , , , , ,
V-FNOO-V-2A             , 1, , 0.000E+000, , , , , , ,
V-FNOO-V-3A             , 1, , 0.000E+000, , , , , , ,
^CLASS
^EOS
```

SAPHIRE CDF Change Set Data and Results

## SAPHIRE Change Set for P-7C Out of Service

```
* PROBABILITY HEADER
* Name          , CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT,
Flag, UncValue2
* CLASS HEADER
* Name,  Group, CompType, CompId, System, Location, FailMode, Train, Init, Att1,..,Att16
* CLASS PROBABILITY HEADER
* CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT, Flag, UncValue2
PSAR2C, P-7C-OOS                =
^PROBABILITY
U-PMME-P-7C             , T, , , , , , , , ,
U-PMMG-P-7C             , T, , , , , , , , ,
U-PMOO-P-7C             , T, , , , , , , , ,
U-HSE-P-7A-STBY         , F, , , , , , , , ,
U-HSE-P-7B-STBY         , F, , , , , , , , ,
U-HSE-P-7C-STBY         , F, , , , , , , , ,
U-PMCC-P-7AB-MG         , 1, , 2.132E-005, , , , , , ,
^CLASS
^EOS
```

## SAPHIRE Change Set for Loss of Service Water Initiating Event Frequency Increased by an Order of Magnitude

```
* PROBABILITY HEADER
* Name          , CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT,
Flag, UncValue2
* CLASS HEADER
* Name,  Group, CompType, CompId, System, Location, FailMode, Train, Init, Att1,..,Att16
* CLASS PROBABILITY HEADER
* CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT, Flag, UncValue2
PSAR2C, IE_LOSWS                =
^PROBABILITY
IE_LOSWS                , 1, , 1.220E-002, , , , , , ,
^CLASS
^EOS
```

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 1 | 11.04 | 2.87E-06 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-03 |
| | | | PP-PMMT-CCW-MBLOCA | PRIMARY COOLANT PUMP SEAL FAILURE GIVEN A SBO AND CONSEQUENTIAL MEDIUM BREAK LOCA | 2.35E-03 |
| 2 | 21.38 | 2.69E-06 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 3 | 26.28 | 1.27E-06 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 4 | 30.15 | 1.00E-06 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMD-CV-3027 | AIR OPERATED VALVE CV-3027 FAILS TO REMAIN OPEN | 4.44E-04 |
| 5 | 34.02 | 1.00E-06 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMD-CV-3056 | AIR OPERATED VALVE CV-3056 FAILS TO REMAIN OPEN | 4.44E-04 |
| 6 | 37.1 | 7.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | A-AVOA-AFWFLADJ | OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FAILURE OF ONE HDR | 1.45E-03 |
| | | | H-ZZOA-OTC-CDTNL-HEP-2 | COND HEP: A-AVOA-AFWFLADJ * B-XVOB-ADVS-MAN * H-ZZOA-OTC-INIT | 3.66E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 7 | 40.18 | 7.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | A-AVOA-AFWFLADJ | OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FAILURE OF ONE HDR | 1.45E-03 |
| | | | H-ZZOA-OTC-CDTNL-HEP-2 | COND HEP: A-AVOA-AFWFLADJ * B-XVOB-ADVS-MAN * H-ZZOA-OTC-INIT | 3.66E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 8 | 42.49 | 5.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | L-ZZOA-SDC-CDTNL-HEP-1 | CONDITIONAL HEP: W-AVOA-PZR-SPRAY * L-ZZOA-SDC-INIT | 1.53E-01 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 9 | 44.75 | 5.88E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVOB-RAS-VLVS | OPERATOR FAILS TO ENABLE ESS RECIRC VALVES TO CLOSE ON RAS | 2.60E-04 |
| 10 | 45.97 | 3.16E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVCC-3027-56MB | BOTH SIRWT RECIRC VALVES CV-3027 & CV-3056 COMMON CAUSE FTC | 1.40E-04 |
| 11 | 47.1 | 2.94E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-LSOH-SIRW-HI | SIRW TANK LEVEL SWITCHES MISCALIBRATED HIGH | 1.30E-04 |
| 12 | 48.23 | 2.94E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-LSOH-SIRW-LOW | SIRW TANK LEVEL SWITCHES MISCALIBRATED LOW | 1.30E-04 |
| 13 | 49.15 | 2.39E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVCC-SUMP-MA | COMMON CAUSE FAILURE OF CV-3029 & CV-3030 TO OPEN | 1.06E-04 |
| 14 | 50 | 2.20E-07 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | /RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 1.00E+00 |
| | | | RXC-MECH-FAULTS | Mechanical Scram Faults | 8.40E-07 |
| | | | /TTF | Turbine Trip | 9.90E-01 |
| 15 | 50.84 | 2.18E-07 | IE_TRANS-WC | TRANSIENT WITH THE MAIN CONDENSER AVAILABLE (IE FREQ) | 1.97E-01 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 16 | 51.55 | 1.84E-07 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | E-DG-ENGINE-REC-4HR | EDG ENGINE RECOVERY IN 4 HOURS | 4.30E-01 |
| | | | E-DGCC-K-6A&B&NSR-MG | EDG1-1 EDG1-2 AND NSR COMMON CAUSE FAILURE TO RUN | 3.44E-04 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | P-LOOP-REC-CORR-4HR | OFFSITE POWER CORRECTION FACTOR FOR EDG 24 HR RUN TIME-4 HR | 3.27E-01 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| | | | REC-4HR | Recovery of Offsite Power in 4 Hours (prior to battery depletion) | 4.70E-01 |
| 17 | 52.04 | 1.27E-07 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-03 |
| | | | PP-PMMT-CCW-SBLOCA | PRIMARY COOLANT PUMP SEAL FAILURE GIVEN A SBO AND CONSEQUENTIAL SMALL BREAK LOCA | 1.04E-04 |
| 18 | 52.48 | 1.15E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-PMCC-P8C66ABME | COMMON CAUSE FAILURE OF P-8C | 5.10E-05 |
| 19 | 52.89 | 1.06E-07 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | E-DGCC-K-6A&B&NSR-ME | EDG1-1 EDG1-2 AND NSR COMMON CAUSE FAIL TO START | 2.78E-05 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| | | | REC-4HR | Recovery of Offsite Power in 4 Hours (prior to battery depletion) | 4.70E-01 |
| 20 | 53.29 | 1.03E-07 | IE_TRANS-WC | TRANSIENT WITH THE MAIN CONDENSER AVAILABLE (IE FREQ) | 1.97E-01 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 21 | 53.68 | 1.01E-07 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | RVC | Pressurizer Safeties Closed | 8.61E-03 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 22 | 54.06 | 9.99E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMA-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO DE-ENERGIZE | 1.14E-02 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 23 | 54.44 | 9.99E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | B-HCMA-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO DE-ENERGIZE | 1.14E-02 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 24 | 54.81 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-3 | RELAY TVX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 25 | 55.18 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-3 | RELAY TVX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 26 | 55.55 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-3 | RELAY TX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 27 | 55.92 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-3 | RELAY TX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 28 | 56.29 | 9.58E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TPMT-PT-0104A | PRESSURE TRANSMITTER PT-0104A FAILS TO FUNCTION | 2.45E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 29 | 56.66 | 9.58E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TPMT-PT-0104B | PRESSURE TRANSMITTER PT-0104B FAILS TO FUNCTION | 2.45E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 30 | 56.96 | 7.90E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TFMT-FT-0306 | SDC INJECTION LINE FLOW TRANSMITTER FT-0306 FAILURE | 2.02E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 31 | 57.26 | 7.81E-08 | IE_LOMF-TRA | LOSS OF FEEDWATER TRAIN A (IE FREQ) | 7.07E-02 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 32 | 57.56 | 7.81E-08 | IE_LOMF-TRB | LOSS OF FEEDWATER TRAIN B (IE FREQ) | 7.07E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 33 | 57.84 | 7.22E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | A-PMMG-P-8B | AFW TURBINE PUMP P-8B FAILS TO RUN | 5.82E-02 |
| | | | P-CBOB-BYREG | WHEN "TRUE" OP RECOVERY OF THE BYPASS REG IS CREDITED | 5.00E-01 |
| | | | P-IVCC-INVALL-MT | COMMON CAUSE FAILURE OF FOUR INVERTERS TO CONTINUE TO OPERAT | 1.02E-06 |
| 34 | 58.1 | 6.64E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-03 |
| | | | A-PMCC-P8ABC-ME | COMMON CAUSE FAILURE OF ALL 3 AFW PUMPS P-8A/B/C TO START | 5.45E-05 |
| 35 | 58.36 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-0306 | SDC HX BYPASS VALVE HIC-0306B FAILS TO FUNCTION | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 36 | 58.62 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-3025A | SDC HX DISCHRG VALVE HAND INDIC CONTROLLER HIC-3025A FAIL | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 37 | 58.88 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-3025B | SDC HX DISCHRG VALVE HAND INDIC CONTROLLER HIC-3025B FAIL | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 38 | 59.14 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-CEPO-POC-0306 | SDC HX BYPASS POSITION CONTROLLER POC-0306 FAILS | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 39 | 59.4 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | L-CEPO-POC-3025 | SDC HX DISCHARGE POSITION CONTROLLER POC-3025 FAILS | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 40 | 59.66 | 6.63E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-MVCC-ESS-ALL12 | COMMON CAUSE FTO OF ALL 8 HPSI MOVS AND ALL 4 LPSI MOVS | 2.94E-05 |
| 41 | 59.92 | 6.63E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-MVCC-ESS-ALL8 | COMMON CAUSE FTO OF ALL 8 HPSI MOVS | 2.94E-05 |
| 42 | 60.16 | 6.19E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | I-FLMK-F-28 | CV-3025 LOCAL IA SUPPLY FILTER F28 PLUGGED | 1.58E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 43 | 60.4 | 6.19E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | Q-FLMK-F-310 | SDC HX INLET VALVE HPA SUPPLY FILTER F-310 PLUGGED | 1.58E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 44 | 60.64 | 6.17E-08 | IE_MLBLOCA | LOSS OF COOLANT ACCIDENT - MED LRGE BRK [>6" and <18"] (IE FREQ) | 3.43E-05 |
| | | | H-AVOT-HL-INJ | OPERATOR FAILS TO ALIGN HOT LEG INJECTION | 1.80E-03 |
| 45 | 60.87 | 6.04E-08 | IE_ISLOCA | INTERFACING SYSTEMS LOCA (IE FREQ) | 1.00E+00 |
| | | | L-MVMJ-MO-3015 | MOTOR OPERATED VALVE 3015 LEAKS (IE EVENT) | 4.85E-03 |
| | | | L-MVMJ-MO-3016 | MOTOR OPERATED VALVE 3016 LEAKS | 1.33E-05 |
| | | | L-PIPE-GC-14 | PIPE FAILS DUE TO PRIMARY CYCLE PRESSURE (GC 14 INCH) | 9.37E-01 |
| 46 | 61.1 | 6.03E-08 | IE_MBLOCA | LOSS OF COOLANT ACCIDENT - MEDIUM BREAK [>2" and <6"] (IE FREQ) | 3.35E-05 |
| | | | H-AVOT-HL-INJ | OPERATOR FAILS TO ALIGN HOT LEG INJECTION | 1.80E-03 |
| 47 | 61.32 | 5.80E-08 | IE_LOMC | LOSS OF MAIN CONDENSER VACUUM (IE FREQ) | 5.25E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 48 | 61.54 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | U-KVMA-SV-0821 | SV-0821 FAILS TO DE-ENERGIZE | 3.93E-03 |
| 49 | 61.76 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | U-KVMA-SV-0821 | SV-0821 FAILS TO DE-ENERGIZE | 3.93E-03 |
| 50 | 61.98 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | Y-KVMA-SV-0938 | CCW TO SDC HX AIR SUPPLY SV-0938 FTD | 3.93E-03 |
| 51 | 62.2 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | Y-KVMA-SV-0938 | CCW TO SDC HX AIR SUPPLY SV-0938 FTD | 3.93E-03 |
| 52 | 62.4 | 5.21E-08 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | A-OOOT-CSTMK-CDTNL-HEP-2 | COND HEP: L-ZZOA-SDC-INIT * A-OOOT-CSTMKUP * P-CBOB-BUS1E | 1.43E-01 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | L-ZZOA-SDC-INIT | OPERATOR FAILS TO INITIATE SDC | 1.55E-02 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| 53 | 62.59 | 4.99E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-SIRWT-MA | BOTH SIRWT SUPPLY CK VALVES CK-ES3239 & CK-ES3240 CCAUSE FTO | 2.21E-05 |
| 54 | 62.78 | 4.99E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-SUMP-MA | BOTH SUMP SUPPLY CK VALVES CK-ES3166 & CK-ES3181 CCAUSE FTO | 2.21E-05 |
| 55 | 62.97 | 4.97E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | A-PMMG-P-8B | AFW TURBINE PUMP P-8B FAILS TO RUN | 5.82E-02 |
| | | | P-CBOB-BYREG | WHEN "TRUE" OP RECOVERY OF THE BYPASS REG IS CREDITED | 5.00E-01 |
| | | | P-IVCC-INV-123MT | COMMON CAUSE FAILURE OF THREE INVERTERS #1 | 7.03E-07 |
| 56 | 63.16 | 4.96E-08 | IE_LOMSIV | SPURIOUS MSIV CLOSURE (IE FREQ) | 4.49E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 57 | 63.35 | 4.96E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMB-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO ENERGIZE | 1.14E-02 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 58 | 63.54 | 4.96E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMB-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO ENERGIZE | 1.14E-02 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 59 | 63.73 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-HPSIPP-MA | BOTH HPSI PUMP DICHARGE CK VLVES CK-ES3177 & 3186 CCAUSE FTO | 2.14E-05 |
| 60 | 63.92 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-RECIRC-MA | BOTH HPSI PUMP RECIRC CK VLVS TO SIRWT COMMON CAUSE FTO | 2.14E-05 |
| 61 | 64.11 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-SUCT-MA | BOTH HPSI PUMP SUMP SUCTION CK VLVS COMMON CAUSE FTO | 2.14E-05 |
| 62 | 64.29 | 4.76E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-RECIRC-MA | BOTH SIRWT RECIRC CK VALVES CK-ES3331 & ES3332 CCAUSE FTO | 2.11E-05 |
| 63 | 64.46 | 4.42E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| 64 | 64.62 | 4.08E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-PMCC-P66AB-ME | COMMON CAUSE FAILURE OF P-66A AND P-66B TO START | 1.81E-05 |
| 65 | 64.78 | 4.04E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | MTC1 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.00E-02 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | /RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 1.00E+00 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | RXC-MECH-FAULTS | Mechanical Scram Faults | 8.40E-07 |
| | | | /TTF | Turbine Trip | 9.90E-01 |
| 66 | 64.93 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 67 | 65.08 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| 68 | 65.23 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 69 | 65.38 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |
| 70 | 65.52 | 3.71E-08 | IE_LOMF-TRA | LOSS OF FEEDWATER TRAIN A (IE FREQ) | 7.07E-02 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 71 | 65.66 | 3.71E-08 | IE_LOMF-TRB | LOSS OF FEEDWATER TRAIN B (IE FREQ) | 7.07E-02 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 72 | 65.79 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 73 | 65.92 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 74 | 66.05 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029A | SUMP TO EAST ESS AIR SUPPLY SV-3029A FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030B | SUMP TO WEST ESS AIR SUPPLY SV-3030B FTE | 3.93E-03 |
| 75 | 66.18 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029B | SUMP TO EAST ESS AIR SUPPLY SV-3029B FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030B | SUMP TO WEST ESS AIR SUPPLY SV-3030B FTE | 3.93E-03 |
| 76 | 66.31 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029A | SUMP TO EAST ESS AIR SUPPLY SV-3029A FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030A | SUMP TO WEST ESS AIR SUPPLY SV-3030A FTE | 3.93E-03 |
| 77 | 66.44 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029B | SUMP TO EAST ESS AIR SUPPLY SV-3029B FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030A | SUMP TO WEST ESS AIR SUPPLY SV-3030A FTE | 3.93E-03 |
| 78 | 66.57 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 79 | 66.7 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 80 | 66.83 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782B | ADV CV-0782 AIR SUPPLY SV-0782B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 81 | 66.96 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781C | ADV CV-0781 AIR SUPPLY SV-0781C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 82 | 67.09 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781B | ADV CV-0781 AIR SUPPLY SV-0781B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 83 | 67.22 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780C | ADV CV-0780 AIR SUPPLY SV-0780C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 84 | 67.35 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782C | ADV CV-0782 AIR SUPPLY SV-0782C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 85 | 67.48 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779B | ADV CV-0779 AIR SUPPLY SV-0779B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 86 | 67.61 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779C | ADV CV-0779 AIR SUPPLY SV-0779C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 87 | 67.74 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780B | ADV CV-0780 AIR SUPPLY SV-0780B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 88 | 67.87 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782C | ADV CV-0782 AIR SUPPLY SV-0782C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| | | | X-HSE-SGA-BLDN | SET TO 'T' - ESDE ON SG E-50A (House Event) | 1.00E+00 |
| 89 | 68 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781B | ADV CV-0781 AIR SUPPLY SV-0781B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| | | | X-HSE-SGA-BLDN | SET TO 'T' - ESDE ON SG E-50A (House Event) | 1.00E+00 |
| 90 | 68.13 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781C | ADV CV-0781 AIR SUPPLY SV-0781C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| | | | X-HSE-SGA-BLDN | SET TO 'T' - ESDE ON SG E-50A (House Event) | 1.00E+00 |
| 91 | 68.26 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782B | ADV CV-0782 AIR SUPPLY SV-0782B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| | | | X-HSE-SGA-BLDN | SET TO 'T' - ESDE ON SG E-50A (House Event) | 1.00E+00 |
| 92 | 68.39 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780C | ADV CV-0780 AIR SUPPLY SV-0780C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| | | | X-HSE-SGB-BLDN | SET TO 'T' - ESDE ON SG E-50B (House Event) | 1.00E+00 |
| 93 | 68.52 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780B | ADV CV-0780 AIR SUPPLY SV-0780B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| | | | X-HSE-SGB-BLDN | SET TO 'T' - ESDE ON SG E-50B (House Event) | 1.00E+00 |
| 94 | 68.65 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779C | ADV CV-0779 AIR SUPPLY SV-0779C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| | | | X-HSE-SGB-BLDN | SET TO 'T' - ESDE ON SG E-50B (House Event) | 1.00E+00 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 95 | 68.78 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779B | ADV CV-0779 AIR SUPPLY SV-0779B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| | | | X-HSE-SGB-BLDN | SET TO 'T' - ESDE ON SG E-50B (House Event) | 1.00E+00 |
| 96 | 68.9 | 3.08E-08 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | E-DG-ENGINE-REC-4HR | EDG ENGINE RECOVERY IN 4 HOURS | 4.30E-01 |
| | | | E-DGMG-K-6A | DIESEL GENERATOR 1-1 FAILS TO RUN | 3.86E-02 |
| | | | E-DGMG-K-6B | DIESEL GENERATOR 1-2 FAILS TO RUN | 3.86E-02 |
| | | | E-DGMG-K-NSR | NSR DIESEL GENERATOR FAILS TO RUN | 3.86E-02 |
| | | | P-LOOP-REC-CORR-4HR | OFFSITE POWER CORRECTION FACTOR FOR EDG 24 HR RUN TIME-4 HR | 3.27E-01 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| | | | REC-4HR | Recovery of Offsite Power in 4 Hours (prior to battery depletion) | 4.70E-01 |
| 97 | 69.02 | 3.01E-08 | IE_LOMF | LOSS OF MAIN FEEDWATER (IE FREQ) | 2.72E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 98 | 69.13 | 2.94E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-AVMB-CV-0782 | ADV ON SG A CV-0782 FAILS TO CLOSE | 3.34E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 99 | 69.24 | 2.94E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-AVMB-CV-0781 | ADV ON SG A CV-0781 FAILS TO CLOSE | 3.34E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |

SAPHIRE CDF Top 100 Cutsets

**Table B-1: PSAR2c Zero Maintenance, P-7C Out of Service (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 100 | 69.35 | 2.94E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-AVMB-CV-0779 | ADV ON SG B CV-0779 FAILS TO CLOSE | 3.34E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |

SAPHIRE CDF Top 100 Cutsets

| Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets) | | | | | |
|---|---|---|---|---|---|
| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
| 1 | 51.61 | 2.87E-05 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | PP-PMMT-CCW-MBLOCA | PRIMARY COOLANT PUMP SEAL FAILURE GIVEN A SBO AND CONSEQUENTIAL MEDIUM BREAK LOCA | 2.35E-03 |
| 2 | 56.44 | 2.69E-06 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 3 | 58.73 | 1.27E-06 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 4 | 61.01 | 1.27E-06 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | PP-PMMT-CCW-SBLOCA | PRIMARY COOLANT PUMP SEAL FAILURE GIVEN A SBO AND CONSEQUENTIAL SMALL BREAK LOCA | 1.04E-04 |
| 5 | 62.82 | 1.00E-06 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMD-CV-3027 | AIR OPERATED VALVE CV-3027 FAILS TO REMAIN OPEN | 4.44E-04 |
| 6 | 64.63 | 1.00E-06 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMD-CV-3056 | AIR OPERATED VALVE CV-3056 FAILS TO REMAIN OPEN | 4.44E-04 |
| 7 | 66.07 | 7.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | A-AVOA-AFWFLADJ | OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FAILURE OF ONE HDR | 1.45E-03 |
| | | | H-ZZOA-OTC-CDTNL-HEP-2 | COND HEP: A-AVOA-AFWFLADJ * B-XVOB-ADVS-MAN * H-ZZOA-OTC-INIT | 3.66E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 8 | 67.51 | 7.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | A-AVOA-AFWFLADJ | OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FAILURE OF ONE HDR | 1.45E-03 |
| | | | H-ZZOA-OTC-CDTNL-HEP-2 | COND HEP: A-AVOA-AFWFLADJ * B-XVOB-ADVS-MAN * H-ZZOA-OTC-INIT | 3.66E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |

SAPHIRE CDF Top 100 Cutsets

| Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets) | | | | | |
|---|---|---|---|---|---|
| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
| 9 | 68.71 | 6.64E-07 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-PMCC-P8ABC-ME | COMMON CAUSE FAILURE OF ALL 3 AFW PUMPS P-8A/B/C TO START | 5.45E-05 |
| 10 | 69.79 | 5.99E-07 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-ZZOA-SDC-CDTNL-HEP-1 | CONDITIONAL HEP: W-AVOA-PZR-SPRAY * L-ZZOA-SDC-INIT | 1.53E-01 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 11 | 70.85 | 5.88E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVOB-RAS-VLVS | OPERATOR FAILS TO ENABLE ESS RECIRC VALVES TO CLOSE ON RAS | 2.60E-04 |
| 12 | 71.42 | 3.16E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVCC-3027-56MB | BOTH SIRWT RECIRC VALVES CV-3027 & CV-3056 COMMON CAUSE FTC | 1.40E-04 |
| 13 | 71.95 | 2.94E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-LSOH-SIRW-HI | SIRW TANK LEVEL SWITCHES MISCALIBRATED HIGH | 1.30E-04 |
| 14 | 72.48 | 2.94E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-LSOH-SIRW-LOW | SIRW TANK LEVEL SWITCHES MISCALIBRATED LOW | 1.30E-04 |
| 15 | 72.91 | 2.39E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVCC-SUMP-MA | COMMON CAUSE FAILURE OF CV-3029 & CV-3030 TO OPEN | 1.06E-04 |
| 16 | 73.31 | 2.20E-07 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | /RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 1.00E+00 |
| | | | RXC-MECH-FAULTS | Mechanical Scram Faults | 8.40E-07 |
| | | | /TTF | Turbine Trip | 9.90E-01 |
| 17 | 73.7 | 2.18E-07 | IE_TRANS-WC | TRANSIENT WITH THE MAIN CONDENSER AVAILABLE (IE FREQ) | 1.97E-01 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 18 | 74.03 | 1.84E-07 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | E-DG-ENGINE-REC-4HR | EDG ENGINE RECOVERY IN 4 HOURS | 4.30E-01 |
| | | | E-DGCC-K-6A&B&NSR-MG | EDG1-1 EDG1-2 AND NSR COMMON CAUSE FAILURE TO RUN | 3.44E-04 |
| | | | P-LOOP-REC-CORR-4HR | OFFSITE POWER CORRECTION FACTOR FOR EDG 24 HR RUN TIME-4 HR | 3.27E-01 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| | | | REC-4HR | Recovery of Offsite Power in 4 Hours (prior to battery depletion) | 4.70E-01 |
| 19 | 74.26 | 1.30E-07 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-CVCC-AFWPP3-MA | ALL 3 AFW PP CK VALVES CK-FW726 | 1.07E-05 |
| 20 | 74.47 | 1.15E-07 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-PMCC-P8C66ABME | COMMON CAUSE FAILURE OF P-8C | 5.10E-05 |
| 21 | 74.66 | 1.07E-07 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-OOOT-CSTMKUP | OPERATOR FAILS TO MAKEUP TO CST | 2.66E-03 |
| | | | A-PMME-P-936 | P-936 FAILS TO START | 3.29E-03 |
| 22 | 74.85 | 1.06E-07 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | E-DGCC-K-6A&B&NSR-ME | EDG1-1 EDG1-2 AND NSR COMMON CAUSE FAIL TO START | 2.78E-05 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| | | | REC-4HR | Recovery of Offsite Power in 4 Hours (prior to battery depletion) | 4.70E-01 |
| 23 | 75.04 | 1.06E-07 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-CVCC-AFWINJ-MA | ALL 4 AFW INJ CHECK VALVES FTO DUE TO COMMON CAUSE | 8.65E-06 |
| 24 | 75.23 | 1.03E-07 | IE_TRANS-WC | TRANSIENT WITH THE MAIN CONDENSER AVAILABLE (IE FREQ) | 1.97E-01 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 25 | 75.41 | 1.01E-07 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | RVC | Pressurizer Safeties Closed | 8.61E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 26 | 75.59 | 9.99E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMA-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO DE-ENERGIZE | 1.14E-02 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 27 | 75.77 | 9.99E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMA-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO DE-ENERGIZE | 1.14E-02 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 28 | 75.95 | 9.83E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-AVCC-AFW-4-MA | ALL 4 AFW AOV'S CCAUSE FTO CV-0727/CV-0736/CV-0736A/CV-0749 | 8.06E-06 |
| 29 | 76.12 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-3 | RELAY TVX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 30 | 76.29 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-3 | RELAY TX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 31 | 76.46 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TX-3 | RELAY TX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| 32 | 76.63 | 9.61E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-3 | RELAY TVX-3 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 33 | 76.8 | 9.58E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TPMT-PT-0104A | PRESSURE TRANSMITTER PT-0104A FAILS TO FUNCTION | 2.45E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 34 | 76.97 | 9.58E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TPMT-PT-0104B | PRESSURE TRANSMITTER PT-0104B FAILS TO FUNCTION | 2.45E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 35 | 77.14 | 9.24E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-PMMG-P-8B | AFW TURBINE PUMP P-8B FAILS TO RUN | 5.82E-02 |
| | | | A-PSOH-AFWLOSUC | MISCALIBRATION OF ALL AFW LOW SUCTION PRESSURE SWITCHES | 1.30E-04 |
| 36 | 77.3 | 8.90E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-OOOT-CSTMK-CDTNL-HEP-2 | COND HEP: L-ZZOA-SDC-INIT * A-OOOT-CSTMKUP * P-CBOB-BUS1E | 1.43E-01 |
| | | | A-PMME-P-936 | P-936 FAILS TO START | 3.29E-03 |
| | | | L-ZZOA-SDC-INIT | OPERATOR FAILS TO INITIATE SDC | 1.55E-02 |
| 37 | 77.44 | 7.97E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-PMCC-P8ABC-MG | COMMON CAUSE FAILURE OF ALL 3 AFW PUMPS P-8A/B/C TO RUN | 6.53E-06 |
| 38 | 77.58 | 7.90E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-TFMT-FT-0306 | SDC INJECTION LINE FLOW TRANSMITTER FT-0306 FAILURE | 2.02E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 39 | 77.72 | 7.81E-08 | IE_LOMF-TRB | LOSS OF FEEDWATER TRAIN B (IE FREQ) | 7.07E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 40 | 77.86 | 7.81E-08 | IE_LOMF-TRA | LOSS OF FEEDWATER TRAIN A (IE FREQ) | 7.07E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 41 | 77.99 | 7.22E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | A-PMMG-P-8B | AFW TURBINE PUMP P-8B FAILS TO RUN | 5.82E-02 |
| | | | P-CBOB-BYREG | WHEN "TRUE" OP RECOVERY OF THE BYPASS REG IS CREDITED | 5.00E-01 |
| | | | P-IVCC-INVALL-MT | COMMON CAUSE FAILURE OF FOUR INVERTERS TO CONTINUE TO OPERAT | 1.02E-06 |
| 42 | 78.11 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-0306 | SDC HX BYPASS VALVE HIC-0306B FAILS TO FUNCTION | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 43 | 78.23 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-3025A | SDC HX DISCHRG VALVE HAND INDIC CONTROLLER HIC-3025A FAIL | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 44 | 78.35 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-HCMT-HIC-3025B | SDC HX DISCHRG VALVE HAND INDIC CONTROLLER HIC-3025B FAIL | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 45 | 78.47 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-CEPO-POC-0306 | SDC HX BYPASS POSITION CONTROLLER POC-0306 FAILS | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 46 | 78.59 | 6.64E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | L-CEPO-POC-3025 | SDC HX DISCHARGE POSITION CONTROLLER POC-3025 FAILS | 1.70E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 47 | 78.71 | 6.63E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-MVCC-ESS-ALL12 | COMMON CAUSE FTO OF ALL 8 HPSI MOVS AND ALL 4 LPSI MOVS | 2.94E-05 |
| 48 | 78.83 | 6.63E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-MVCC-ESS-ALL8 | COMMON CAUSE FTO OF ALL 8 HPSI MOVS | 2.94E-05 |
| 49 | 78.94 | 6.20E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-AVOA-CV-2010 | OPERATOR FAILS TO OPEN CV-2010 FOR T-939 MAKEUP TO CST | 2.59E-03 |
| | | | A-KVMB-SV-2010 | CST MAKEUP CV-2010 SOLENOID SV-2010 FTE | 3.93E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | A-OOOT-CSTMK-CDTNL-HEP-1 | COND HEP: A-AVOA-CV-2010 * A-OOOT-CSTMKUP * Y-AVOB-RAS-VLVS | 4.99E-01 |
| 50 | 79.05 | 6.19E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | I-FLMK-F-28 | CV-3025 LOCAL IA SUPPLY FILTER F28 PLUGGED | 1.58E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 51 | 79.16 | 6.19E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | Q-FLMK-F-310 | SDC HX INLET VALVE HPA SUPPLY FILTER F-310 PLUGGED | 1.58E-02 |
| | | | W-AVOA-PZR-SPRAY | OPERATOR FAILS TO DEPRESSURIZE PCS WITH PZR SPRAY/AUX SPRAY | 1.30E-03 |
| 52 | 79.27 | 6.17E-08 | IE_MLBLOCA | LOSS OF COOLANT ACCIDENT - MED LRGE BRK [>6" and <18"] (IE FREQ) | 3.43E-05 |
| | | | H-AVOT-HL-INJ | OPERATOR FAILS TO ALIGN HOT LEG INJECTION | 1.80E-03 |
| 53 | 79.38 | 6.04E-08 | IE_ISLOCA | INTERFACING SYSTEMS LOCA (IE FREQ) | 1.00E+00 |
| | | | L-MVMJ-MO-3015 | MOTOR OPERATED VALVE 3015 LEAKS (IE EVENT) | 4.85E-03 |
| | | | L-MVMJ-MO-3016 | MOTOR OPERATED VALVE 3016 LEAKS | 1.33E-05 |
| | | | L-PIPE-GC-14 | PIPE FAILS DUE TO PRIMARY CYCLE PRESSURE (GC 14 INCH) | 9.37E-01 |
| 54 | 79.49 | 6.03E-08 | IE_MBLOCA | LOSS OF COOLANT ACCIDENT - MEDIUM BREAK [>2" and <6"] (IE FREQ) | 3.35E-05 |
| | | | H-AVOT-HL-INJ | OPERATOR FAILS TO ALIGN HOT LEG INJECTION | 1.80E-03 |
| 55 | 79.59 | 5.80E-08 | IE_LOMC | LOSS OF MAIN CONDENSER VACUUM (IE FREQ) | 5.25E-02 |
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 56 | 79.69 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | U-KVMA-SV-0821 | SV-0821 FAILS TO DE-ENERGIZE | 3.93E-03 |
| 57 | 79.79 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | U-KVMA-SV-0821 | SV-0821 FAILS TO DE-ENERGIZE | 3.93E-03 |
| 58 | 79.89 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | R-REMD-TVX-4 | RELAY TVX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | Y-KVMA-SV-0938 | CCW TO SDC HX AIR SUPPLY SV-0938 FTD | 3.93E-03 |
| 59 | 79.99 | 5.79E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | R-REMD-TX-4 | RELAY TX-4 FAILS TO REMAIN DE-ENERGIZED | 6.52E-03 |
| | | | Y-KVMA-SV-0938 | CCW TO SDC HX AIR SUPPLY SV-0938 FTD | 3.93E-03 |
| 60 | 80.09 | 5.69E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-FLMK-F-P936 | P-936 SUCTION STRAINER PLUGS | 1.76E-03 |
| | | | A-OOOT-CSTMKUP | OPERATOR FAILS TO MAKEUP TO CST | 2.66E-03 |
| 61 | 80.18 | 5.21E-08 | IE_LOOP | Loss of Offsite Power | 1.11E-02 |
| | | | A-OOOT-CSTMK-CDTNL-HEP-2 | COND HEP: L-ZZOA-SDC-INIT * A-OOOT-CSTMKUP * P-CBOB-BUS1E | 1.43E-01 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | L-ZZOA-SDC-INIT | OPERATOR FAILS TO INITIATE SDC | 1.55E-02 |
| | | | REC-30MIN | Recovery of Offsite Power in 30 min (prior to S/G dryout) | 7.30E-01 |
| 62 | 80.27 | 5.19E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-AVOA-CV-2010 | OPERATOR FAILS TO OPEN CV-2010 FOR T-939 MAKEUP TO CST | 2.59E-03 |
| | | | A-OOOT-CSTMK-CDTNL-HEP-1 | COND HEP: A-AVOA-CV-2010 * A-OOOT-CSTMKUP * Y-AVOB-RAS-VLVS | 4.99E-01 |
| | | | A-PMME-P-936 | P-936 FAILS TO START | 3.29E-03 |
| 63 | 80.36 | 4.99E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-SIRWT-MA | BOTH SIRWT SUPPLY CK VALVES CK-ES3239 & CK-ES3240 CCAUSE FTO | 2.21E-05 |
| 64 | 80.45 | 4.99E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-SUMP-MA | BOTH SUMP SUPPLY CK VALVES CK-ES3166 & CK-ES3181 CCAUSE FTO | 2.21E-05 |
| 65 | 80.54 | 4.97E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | A-PMMG-P-8B | AFW TURBINE PUMP P-8B FAILS TO RUN | 5.82E-02 |
| | | | P-CBOB-BYREG | WHEN "TRUE" OP RECOVERY OF THE BYPASS REG IS CREDITED | 5.00E-01 |
| | | | P-IVCC-INV-123MT | COMMON CAUSE FAILURE OF THREE INVERTERS #1 | 7.03E-07 |
| 66 | 80.63 | 4.96E-08 | IE_LOMSIV | SPURIOUS MSIV CLOSURE (IE FREQ) | 4.49E-02 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | MTC2 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.30E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 67 | 80.72 | 4.96E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMB-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO ENERGIZE | 1.14E-02 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 68 | 80.81 | 4.96E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-HCMB-HIC-0780A | SDCR CONTROLLER HIC-0780A FAILS TO ENERGIZE | 1.14E-02 |
| | | | H-ZZOA-OTC-INIT | OPERATOR FAILS TO INITIATE ONCE THROUGH COOLING | 2.90E-03 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 69 | 80.9 | 4.90E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-PMME-P-936 | P-936 FAILS TO START | 3.29E-03 |
| | | | A-XVMA-MV-FW775 | FPS TO AFW MANUAL VALVE MV-FW775 FAILS TO OPEN | 1.22E-03 |
| 70 | 80.99 | 4.90E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |
| | | | A-PMME-P-936 | P-936 FAILS TO START | 3.29E-03 |
| | | | A-XVMA-MV-FW774 | FPS TO AFW MANUAL VALVE MV-FW774 FAILS TO OPEN | 1.22E-03 |
| 71 | 81.08 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-HPSIPP-MA | BOTH HPSI PUMP DICHARGE CK VLVES CK-ES3177 & 3186 CCAUSE FTO | 2.14E-05 |
| 72 | 81.17 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-RECIRC-MA | BOTH HPSI PUMP RECIRC CK VLVS TO SIRWT COMMON CAUSE FTO | 2.14E-05 |
| 73 | 81.26 | 4.84E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | H-CVCC-SUCT-MA | BOTH HPSI PUMP SUMP SUCTION CK VLVS COMMON CAUSE FTO | 2.14E-05 |
| 74 | 81.35 | 4.76E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-CVCC-RECIRC-MA | BOTH SIRWT RECIRC CK VALVES CK-ES3331 & ES3332 CCAUSE FTO | 2.11E-05 |
| 75 | 81.44 | 4.75E-08 | IE_LOSWS | LOSS OF SERVICE WATER SYSTEM (IE FREQ) | 1.22E-02 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | A-FLMK-F-P936 | P-936 SUCTION STRAINER PLUGS | 1.76E-03 |
| | | | A-OOOT-CSTMK-CDTNL-HEP-2 | COND HEP: L-ZZOA-SDC-INIT * A-OOOT-CSTMKUP * P-CBOB-BUS1E | 1.43E-01 |
| | | | L-ZZOA-SDC-INIT | OPERATOR FAILS TO INITIATE SDC | 1.55E-02 |
| 76 | 81.52 | 4.42E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| 77 | 81.59 | 4.08E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-PMCC-P66AB-ME | COMMON CAUSE FAILURE OF P-66A AND P-66B TO START | 1.81E-05 |
| 78 | 81.66 | 4.04E-08 | IE_CNTRLSD | CONTROLLED MANUAL SHUTDOWN (IE FREQ) | 2.43E+00 |
| | | | MTC1 | PERCENTAGE OF TIME W/MTC NOT SUFFICIENTLY POSITIVE | 2.00E-02 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | /RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 1.00E+00 |
| | | | RXC-MECH-FAULTS | Mechanical Scram Faults | 8.40E-07 |
| | | | /TTF | Turbine Trip | 9.90E-01 |
| 79 | 81.73 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 80 | 81.8 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3027 | SIRWT RECIRC VALVE CV-3027 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 81 | 81.87 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| 82 | 81.94 | 3.93E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-AVMB-CV-3056 | SIRWT RECIRC VALVE CV-3056 FTC | 4.42E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| 83 | 82.01 | 3.71E-08 | IE_LOMF-TRA | LOSS OF FEEDWATER TRAIN A (IE FREQ) | 7.07E-02 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 84 | 82.08 | 3.71E-08 | IE_LOMF-TRB | LOSS OF FEEDWATER TRAIN B (IE FREQ) | 7.07E-02 |
| | | | G-PMOE-P-55ABC | OPERATOR FAILS TO INITIATE CHARGING FLOW | 1.10E-01 |
| | | | /RVC | Pressurizer Safeties Closed | 9.91E-01 |
| | | | /RVO | Pressurizer Safeties Open | 9.99E-01 |
| | | | RXC-ELEC-FAULTS | Electrical Scram Signal Faults | 4.81E-06 |
| 85 | 82.14 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 86 | 82.2 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056B | SIRWT RECIRC VALVE SOLENOID SV-3056B FTE | 3.93E-03 |
| 87 | 82.26 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029A | SUMP TO EAST ESS AIR SUPPLY SV-3029A FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030B | SUMP TO WEST ESS AIR SUPPLY SV-3030B FTE | 3.93E-03 |
| 88 | 82.32 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029B | SUMP TO EAST ESS AIR SUPPLY SV-3029B FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030B | SUMP TO WEST ESS AIR SUPPLY SV-3030B FTE | 3.93E-03 |
| 89 | 82.38 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Z-KVMB-SV-3029A | SUMP TO EAST ESS AIR SUPPLY SV-3029A FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030A | SUMP TO WEST ESS AIR SUPPLY SV-3030A FTE | 3.93E-03 |
| 90 | 82.44 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | Z-KVMB-SV-3029B | SUMP TO EAST ESS AIR SUPPLY SV-3029B FTE | 3.93E-03 |
| | | | Z-KVMB-SV-3030A | SUMP TO WEST ESS AIR SUPPLY SV-3030A FTE | 3.93E-03 |
| 91 | 82.5 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027A | SIRWT RECIRC VALVE SOLENOID SV-3027A FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 92 | 82.56 | 3.49E-08 | IE_SBLOCA | LOSS OF COOLANT ACCIDENT - SMALL BRK [>0.4" and <2"] (IE FREQ) | 2.26E-03 |
| | | | Y-KVMB-SV-3027B | SIRWT RECIRC VALVE SOLENOID SV-3027B FTE | 3.93E-03 |
| | | | Y-KVMB-SV-3056A | SIRWT RECIRC VALVE SOLENOID SV-3056A FTE | 3.93E-03 |
| 93 | 82.62 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782B | ADV CV-0782 AIR SUPPLY SV-0782B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 94 | 82.68 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781C | ADV CV-0781 AIR SUPPLY SV-0781C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 95 | 82.74 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0781B | ADV CV-0781 AIR SUPPLY SV-0781B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 96 | 82.8 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780C | ADV CV-0780 AIR SUPPLY SV-0780C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |

SAPHIRE CDF Top 100 Cutsets

**Table B-2: PSAR2c Zero Maintenance, P-7C Out of Service, & Loss of Service Water Initiating Event Frequency Increase (Top 100 Cutsets)**

| Cut No. | % Total | Prob./Frequency | Basic Event | Description | Event Prob. |
|---|---|---|---|---|---|
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 97 | 82.86 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0782C | ADV CV-0782 AIR SUPPLY SV-0782C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRA | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG A (developed event) | 5.00E-01 |
| 98 | 82.92 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779B | ADV CV-0779 AIR SUPPLY SV-0779B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 99 | 82.98 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0779C | ADV CV-0779 AIR SUPPLY SV-0779C FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |
| 100 | 83.04 | 3.46E-08 | IE_SGTR | STEAM GENERATOR TUBE RUPTURE (IE FREQ) | 3.01E-03 |
| | | | B-KVMA-SV-0780B | ADV CV-0780 AIR SUPPLY SV-0780B FTD | 3.93E-03 |
| | | | B-XVOB-ADVS-MAN | OPERATOR FAILS TO CLOSE MANUAL VALVES TO CLOSE ADV | 4.03E-02 |
| | | | L-ZZOA-SDC-CDTNL-HEP-2 | CONDITIONAL HEP: B-XVOB-ADVS-MAN * L-ZZOA-SDC-INIT | 1.45E-01 |
| | | | SGTRB | FT TOP : STEAM GENERATOR TUBE RUPTURE ON SG B (developed event) | 5.00E-01 |

SAPHIRE LERF Change Set Data and Results

**Table of Contents**

## SAPHIRE LERF Change Set Data and Results

```
CET=
CET-QUANT-BASELINE


* PROBABILITY HEADER
* Name          , CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT,
Flag, UncValue2
* CLASS HEADER
* Name,  Group, CompType, CompId, System, Location, FailMode, Train, Init, Att1,..,Att16
* CLASS PROBABILITY HEADER
* CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT, Flag, UncValue2
SAMA-CET, CET-QUANT-BASELINE      =
^PROBABILITY
CET_DEJP   , , ,7.605E-6, , , , , , ,
CE-TP-PORVS, 1, ,6.319E-1, , , , , , ,
CET_ZEGP   , , ,5.37E-6, , , , , , ,
CE-MP-PORVS, 1, ,0.E0, , , , , , ,
CET_DEJS   , , ,4.405E-6, , , , , , ,
CE-TW-PORVS, 1, ,5.366E-1, , , , , , ,
CET_BEGP   , , ,4.039E-6, , , , , , ,
CE-MW-PORVS, 1, ,1.E0, , , , , , ,
CET_A2EGR   , , ,2.996E-6, , , , , , ,
CE-TV-PORVS, 1, ,1.E0, , , , , , ,
CET_BEGR   , , ,2.556E-6, , , , , , ,
CE-MV-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJW   , , ,1.304E-6, , , , , , ,
CE-TR-PORVS, 1, ,1.E0, , , , , , ,
CET_BEGV   , , ,1.073E-6, , , , , , ,
CE-MR-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJP   , , ,7.832E-7, , , , , , ,
CE-TS-PORVS, 1, ,1.E0, , , , , , ,
CET_TEJQ   , , ,4.175E-7, , , , , , ,
CE-MS-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJS   , , ,4.437E-7, , , , , , ,
CE-TQ-PORVS, 1, ,9.87E-1, , , , , , ,
CET_BEGS   , , ,3.95E-7, , , , , , ,
CE-MQ-PORVS, 1, ,0.E0, , , , , , ,
CET_A2EGP   , , ,3.68E-7, , , , , , ,
CE-BP-PORVS, 1, ,9.986E-1, , , , , , ,
CET_DEJR   , , ,3.583E-8, , , , , , ,
CE-BR-PORVS, 1, ,9.993E-1, , , , , , ,
CET_TEJR   , , ,3.605E-8, , , , , , ,
CE-BS-PORVS, 1, ,1.E0, , , , , , ,
CET_MEJW   , , ,6.126E-9, , , , , , ,
CE-BV-PORVS, 1, ,1.E0, , , , , , ,
CET_A1EGR   , , ,5.145E-9, , , , , , ,
CE-DS-PORVS, 1, ,9.979E-1, , , , , , ,
CET_TEJV   , , ,2.382E-9, , , , , , ,
CE-DP-PORVS, 1, ,5.327E-1, , , , , , ,
CET_MEJP   , , ,0.E0, , , , , , ,
CE-DR-PORVS, 1, ,6.495E-1, , , , , , ,
CET_MEJV   , , ,0.E0, , , , , , ,
CE-A2P-PORVS, 1, ,9.428E-1, , , , , , ,
CET_MEJR   , , ,0.E0, , , , , , ,
CE-A2R-PORVS, 1, ,9.923E-1, , , , , , ,
CET_MEJS   , , ,0.E0, , , , , , ,
CE-A1R-PORVS, 1, ,1.E0, , , , , , ,
CET_MEJQ   , , ,0.E0, , , , , , ,
CE-ZP-PORVS, 1, ,9.967E-1, , , , , , ,
CET_CEJW , , ,7.907E-8, , , , , , ,
^CLASS
^EOS
```

## SAPHIRE LERF Change Set Data and Results

```
CET=
CET-QUANT-P7C-FAIL


* PROBABILITY HEADER
* Name          , CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT,
Flag, UncValue2
* CLASS HEADER
* Name,  Group, CompType, CompId, System, Location, FailMode, Train, Init, Att1,..,Att16
* CLASS PROBABILITY HEADER
* CalcType,  UncType, Prob, Lambda, Tau, UncValue, UncCorr, MissionT, Flag, UncValue2
SAMA-CET, CET-QUANT-P7C-FAIL      =
^PROBABILITY
CET_DEJP   , , ,7.605E-6, , , , , , ,
CE-TP-PORVS, 1, ,6.487E-1, , , , , , ,
CET_ZEGP   , , ,5.37E-6, , , , , , ,
CE-MP-PORVS, 1, ,0.E0, , , , , , ,
CET_DEJS   , , ,4.405E-6, , , , , , ,
CE-TW-PORVS, 1, ,5.362E-1, , , , , , ,
CET_BEGP   , , ,4.039E-6, , , , , , ,
CE-MW-PORVS, 1, ,1.E0, , , , , , ,
CET_A2EGR   , , ,2.996E-6, , , , , , ,
CE-TV-PORVS, 1, ,1.E0, , , , , , ,
CET_BEGR   , , ,2.556E-6, , , , , , ,
CE-MV-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJW   , , ,1.418E-6, , , , , , ,
CE-TR-PORVS, 1, ,1.E0, , , , , , ,
CET_BEGV   , , ,1.362E-6, , , , , , ,
CE-MR-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJP   , , ,8.206E-7, , , , , , ,
CE-TS-PORVS, 1, ,1.E0, , , , , , ,
CET_TEJS   , , ,4.437E-7, , , , , , ,
CE-MS-PORVS, 1, ,0.E0, , , , , , ,
CET_TEJQ   , , ,4.545E-7, , , , , , ,
CE-TQ-PORVS, 1, ,9.881E-1, , , , , , ,
CET_BEGS   , , ,3.95E-7, , , , , , ,
CE-MQ-PORVS, 1, ,0.E0, , , , , , ,
CET_A2EGP   , , ,3.68E-7, , , , , , ,
CE-BP-PORVS, 1, ,9.986E-1, , , , , , ,
CET_TEJR   , , ,3.605E-8, , , , , , ,
CE-BR-PORVS, 1, ,9.993E-1, , , , , , ,
CET_DEJR   , , ,7.199E-8, , , , , , ,
CE-BS-PORVS, 1, ,1.E0, , , , , , ,
CET_MEJW   , , ,6.126E-9, , , , , , ,
CE-BV-PORVS, 1, ,1.E0, , , , , , ,
CET_A1EGR   , , ,5.145E-9, , , , , , ,
CE-DS-PORVS, 1, ,9.979E-1, , , , , , ,
CET_TEJV   , , ,2.382E-9, , , , , , ,
CE-DP-PORVS, 1, ,5.327E-1, , , , , , ,
CET_MEJP   , , ,0.E0, , , , , , ,
CE-DR-PORVS, 1, ,8.256E-1, , , , , , ,
CET_MEJV   , , ,0.E0, , , , , , ,
CE-A2P-PORVS, 1, ,9.428E-1, , , , , , ,
CET_MEJR   , , ,0.E0, , , , , , ,
CE-A2R-PORVS, 1, ,9.923E-1, , , , , , ,
CET_MEJS   , , ,0.E0, , , , , , ,
CE-A1R-PORVS, 1, ,1.E0, , , , , , ,
CET_MEJQ   , , ,0.E0, , , , , , ,
CE-ZP-PORVS, 1, ,9.967E-1, , , , , , ,
CET_CEJW , , ,7.907E-8, , , , , , ,
^CLASS
^EOS
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**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **Initiating Event Analysis (to support meeting IE-D3)** | | | | | | |
| 1. Grid stability | Recently the stability of at least some local areas of the electric power grid has been questioned. The potential duration and complexities of recovery from such events are hard to dismiss. Three different aspects relate to this issue:<br>1a. LOOP Initiating Event Frequency<br>1b. Conditional LOOP Frequency<br>1c. Availability of dc power to perform restoration actions | LOOP sequences including consequential LOOP sequences | LOOP events have been minimized at Palisades, in part due to the installation of the safeguards transformer in 1990. The safeguards transformer provides power to the safety related 2400V AC buses 1C and 1D and the non-safety related Bus 1E. The safeguards transformer is connected directly to the F Bus in the switchyard and does not require transferring loads upon a plant trip. A loss of the safeguards transformer or F bus would lead to a fast transfer to the startup transformer on the R bus. As such, the LOOP initiating event priors utilize industry data that have been screened for applicability at Palisades, and a two-stage Bayes update was performed to develop a plant-specific LOOP frequency.<br><br>Moreover, a supplemental diesel generator was installed in June 2006. The non-safety related | 1) Screening of non-applicable events including events at Palisades is appropriate to best represent the current configuration at the site.<br>Additionally, as a result of the 2003 Northeast blackout, it was considered appropriate to increase the likelihood of a site LOOP event by 25% given that other nuclear power plants tripped in the East Central Area Reliability (ECAR) region. The 0.25 factor was proposed as a bounding value for the fractional LOOP. The fractional-LOOP is intended to account for low grid operating margin more typically found in the summer months during one-fourth of the year, i.e., a low enough margin that a nuclear power plant was forced to trip. | 1) The LOOP initiator frequency is included as a unique initiating event in the model. | The overall approach for the LOOP frequency and fail to recover probabilities utilized is considered appropriate to best represent the plant-specific features at Palisades.<br>However, alternative hypothesis exist from NUREG/CR-6890 [D-4] and NUREG/CR-6928 [D-5] that provide generic LOOP frequencies that are about twice as high as that currently used in the Palisades model, and as such, the LOOP initiating event frequency is identified as a candidate source of model uncertainty. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | diesel generator is physically located inside the protected area, just southeast of start-up transformer and consists of a trailer type diesel generator set with a diesel engine and self contained engine cooling and lubrication systems. The components are pre-assembled and contained in a portable type, mobile, heated, ventilated and lighted, tractor-trailer. The supplemental diesel is designed for local manual start only; no remote control is available. Protection features for engine shutdown and tripping of generator output circuit breaker include: engine over-speed, over-crank, high water temperature, and low oil pressure. The load profile is limited to performing heat removal via secondary cooling with an auxiliary feedwater pump or once-through cooling (OTC – Feed & Bleed) via high pressure injection and a PORV and necessary supporting equipment.<br>In 2008 a blackout procedure was implemented to continue | | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | feeding a steam generator given loss of dc power. This procedure provides guidance to the operators on how to provide makeup with the turbine driven feedwater pump without available instrumentation. | | | |
| | | | The industry wide data in NUREG/CR-6890 [D-4] and EPRI reports through 2008 were screened for applicability at Palisades and failure to recover probabilities were derived for the applicable time frames in the model. | 2) The industry-wide recovery data as applicable is appropriate to best estimate the fail to recover probabilities associated with the current configuration at the site. | 2) LOOP recovery failures are included for time periods of 30 minutes, 2 hours, and 4 hours from sequence initiation depending on the accident sequence progression. | |
| | | | The consequential LOOP failure probability is based on the annual frequency normalized for a 24-hour PRA mission time. | 3) Given the current off-site power configuration at Palisades, the use of the derived plant-specific data best represents the likelihood of a consequential LOOP given some other initiating event. However, no credit for recovery from these consequential LOOP events is taken. | 3) Consequential LOOP events are possible from all other IEs, but no credit for off-site power recovery is incorporated into the sequence modeling. | Given that alternate values of ~2E-3 and ~2E-2 are available from the accepted industry generic values [D-6] given a reactor trip or LOCA, respectively, then the consequential LOOP failure probabilities are identified as a candidate source of model uncertainty. |
| | | | Offsite power restoration is dictated by procedure. Restoration is possible via manual breaker control. | 4) The specific failure modes of the offsite restoration are implicitly included via the use of the LOOP recovery probabilities that were screened for applicability at Palisades. | 4) No additional adjustments or system model changes are incorporated when using the different LOOP recovery probabilities. | Realistic using the best available data for the recovery times and recovery probabilities utilized. This should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 2. Support System Initiating Events | Increasing use of plant-specific models for support system initiators (e.g. loss of SW, CCW, or IA, and loss of ac or dc buses) have led to inconsistencies in approaches across the industry. A number of challenges exist in modeling of support system initiating events:<br>2a. Treatment of common cause failures<br>2b. Potential for recovery | Support system event sequences | Support System Initiating Events (SSIE) are included for several loss of ac bus and dc bus initiators as well as for loss of SW, loss of CCW, and loss of IA.<br>All of the support system IE frequencies are based on available generic prior information from NUREG/CR-5750 [25]. | 1) NUREG/CR-5750 provides an appropriate source of generic prior data for the various support system initiators that are applicable to Palisades. | 1) Event sequences are developed for each loss of support system initiator and the given support system is rendered unavailable in the accident sequence development. | The treatment is deemed acceptably representative as dependencies are appropriately captured. This should not be a source of model uncertainty in most applications. |
| | | | Potential for common cause failures within the IA, SW, and CCW initiators are implicitly included via the use of generic data.<br>Additionally, unique initiating events for all potential combinations of losses of instrument ac buses are included. | 2) The use of the generic alpha factors based on industry wide experience is applicable for the ac instrument bus initiators at the site. | 2) The CCF Initiating Events for loss of ac instrument buses dominate the overall contribution to CDF compared to the individual loss of bus initiators. | The treatment is deemed acceptable with a slight conservative bias slant since the alpha factors are known to be high when utilized in an annualized fashion and compared to plant-specific experience. This should not be a source of model uncertainty in most applications. |
| | | | The support system initiating events are generally used as is with no additional credit for recovery. The exception is that a plant-specific analysis has been developed to determine the likelihood of recovery from a LOSW event based on the types of failures contributing to that initiator | 3) The lack of credit for recovery from the support system initiating events will not significantly impact the CDF and LERF distribution. | 3) No basic events included in model for recovery from the loss of support system initiators except for the recovery factor utilized for the loss of service water. | Slight conservative bias because generally no credit is taken for recovery. This should not be a source of model uncertainty in most applications.<br>However, the recovery value utilized for loss of service water events is identified as a candidate source of |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | that could be easily recovered (e.g. basket strainer clogging). | | | model uncertainty. |
| 3. LOCA initiating event frequencies | It is difficult to establish values for events that have never occurred or have rarely occurred with a high level of confidence. The choice of available data sets or use of specific methodologies in the determination of LOCA frequencies could impact base model results and some applications. | LOCA sequences | Although NUREG/CR-5750 includes industry-average baselines for LOCAs, the Palisades LOCA IE frequencies are based on the CEOG methodology described in EA-PSA-00-0010 [D-9]. | 1) The CEOG methodology provides an appropriate estimate for the LOCA initiating event frequencies for Palisades and these values were noted as acceptable by the NRC Expert Elicitation committed during their review of the Palisades PTS evaluation [D-26]. | 1) The LOCA initiating event frequencies can impact risk results directly. There are four categories utilized in the model (i.e., Small, Medium, Medium-Large, and Large). | The Small Break LOCA and Large Break LOCA frequencies are higher than the values from NUREG-1829 [D-10] that were derived through an expert elicitation process. The Medium Break LOCA frequency is lower than that provided in NUREG-1829, and no equivalent exists for the Medium-Large category. As such, the LOCA frequencies are identified as a candidate source of model uncertainty. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **Accident Sequence Analysis (to support meeting AS-C3)** | | | | | | |
| 4. Operation of equipment after battery depletion | Station Blackout events are important contributors to baseline CDF at nearly every US NPP. In many cases, battery depletion may be assumed to lead to loss of all system capability. Some PRAs have credited manual operation of systems that normally require dc for successful operation (e.g. turbine driven systems such as RCIC and AFW). | Credit for continued operation of these systems in sequences with batteries depleted (e.g. long term SBO sequences). | No credit is taken for continued operation of most equipment without dc power. However, credit is taken for AFW restoration/continuation after battery depletion in SBO scenarios as instructed by the plant EOP Supplement #19. | 1) Procedurally directed and practiced action is feasible to execute in SBO conditions. The standard Palisades HRA methodology utilizing the EPRI HRA Calculator approach is an appropriate method for determining the likelihood of failure of this action. | 1) The event sequence modeling for SBO scenarios is set up to avert core damage from occurring with success of this action. | Credit for the viability of AFW continuation/restoration is identified as a candidate source of model uncertainty. |
| 5. RCP seal LOCA treatment – PWRs | The assumed timing and magnitude of RCP seal LOCAs given a loss of seal cooling can have a substantial influence on the risk profile. | Accident sequences involving loss of seal cooling | Utilize PWROG consensus model approach for CE plants [D-11]. | 1) PWROG consensus model approach is directly applicable to Palisades. | 1) Four different probability values representing two different PCP seal break sizes are utilized in the model for the various accident scenarios analyzed. | Consensus model approach utilized. Therefore the RCP seal LOCA treatment is not identified as a candidate source of model uncertainty. |
| 6. Recirculation pump seal leakage treatment – BWRs w/ Isolation Condensers | Recirculation pump seal leakage can lead to loss of the Isolation Condenser. While *recirculation pump seal* leakage is generally modeled, there is no consensus approach on the likelihood of such leaks. | Accident sequences with long-term use of isolation condenser | N/A | N/A | N/A | N/A |
| **Success Criteria (to support meeting SC-C3)** | | | | | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 7. Impact of containment venting on core cooling system NPSH | Many BWR core cooling systems utilize the suppression pool as a water source. Venting of containment as a decay heat removal mechanism can substantially reduce NPSH, even lead to flashing of the pool. The treatment of such scenarios varies across BWR PRAs. | Loss of containment heat removal scenarios with containment venting successful. | N/A | N/A | N/A | N/A |
| 8. Core cooling success following containment failure or venting through non hard pipe vent paths | Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive | Long term loss of decay heat removal scenarios. | No credit is taken for continued injection in loss of containment heat removal scenarios. | 1) Loss of NPSH or inventory issues will eventually lead to termination of injection systems taking suction from the containment sump. Therefore, all loss of containment heat removal scenarios are assumed to eventually result in core damage. | 1) No injection is credited in total loss of containment heat removal scenarios. | No credit for these systems in loss of containment heat removal scenarios may represent a slight conservative bias slant. This should not be a source of model uncertainty in most applications. |

***Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)***

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | reference on the proper treatment of these issues. | | | | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 9. Room heatup calculations | Loss of HVAC can result in room temperatures exceeding equipment qualification limits. Treatment of HVAC requirements varies across the industry and often varies within a PRA. There are two aspects to this issue. One involves whether the SSCs affected by loss of HVAC are assumed to fail (i.e. there is uncertainty in the fragility of the components). The other involves how the rate of room heatup is calculated and the assumed timing of the failure. | Dependency on HVAC for system modeling and timing of accident progressions and associated success criteria. | Plant specific calculations are referenced to determine the HVAC requirements in the model. | 1) EA-C-PAL-98-1574 [D-12] was performed to analyze the heat up of the engineered safeguards rooms without room ventilation. These rooms house the HPSI, LPSI, containment spray, and auxiliary feedwater pump P-8C.<br>EA-C-PAL-98-1574 evaluated the temperature profiles in the east and west engineered safeguards rooms following a large break loss of coolant accident concurrent with a loss of offsite power, and a failure of the room coolers. The calculation demonstrates that for the assumed 24 hour mission time of the PRA, safeguards room components can function without room cooling. | 1) There are no room cooling dependencies for the HPCI, LPSI, CSS, and AFW P-8C systems included in the PRA model. | Realistic using the best estimates for expected plant response. This should not be a source of model uncertainty in most applications. |
| | | | | 2) EA-GOTHIC-AFW-01 [D-13] and EA-GOTHIC-AFW-02 [D-14] were preformed to calculate conservative temperature profiles over time in the AFW pump room. The analysis evaluate a room heat-up due to a plant transient with loss of forced ventilation and room heat-up due to a high energy line break in the turbine building.<br>These analyses demonstrate that forced ventilation is not required for AFW components to function for either a HELB or | 2) There are no room cooling dependencies for the HPCI, LPSI, CSS, and AFW P-8C systems included in the PRA model. | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | | design basis room heat-up. | | |
| | | | | 3) EA-APR-95-023 [D-15] calculated the room temperature profile under Appendix R boundary conditions for the 1-C and 1-D switchgear rooms considering loss of forced ventilation.<br>This calculation demonstrated that Forced ventilation is not required in either the 1-C or 1-D switchgear rooms for equipment functionality. | 3) There are no room cooling dependencies for the 1-C and 1-D switchgear rooms included in the PRA model. | Realistic using the best estimates for expected plant response. This should not be a source of model uncertainty in most applications. |
| | | | | 4) EA-CA025644-01 [D-16] evaluated the room heat-up using design basis assumptions and acceptance criteria and EA-CA023959-01 [D-17] performed several sensitivity analyses on various configurations of room ventilation and evaluated temperatures in the voltage regulator cabinet.<br>The analyses demonstrated that ventilation is required to maintain the design basis bulk room air temperature of 120°F if a diesel generator is in service. | 4) Based on these evaluations, DG HVAC requirements are included in the model. | |
| | | | | 5) LOCA analysis (NAI-1198-002 [D-18] and HELB analysis (SR-6, Rev. 3A [D-19]) were performed to demonstrate that CCW room cooling is not required. | 5) There are no HVAC equipment dependences in the component cooling water rooms. | |
| | | | | 6) EA-APR-95-023 [D-15] | 6) No specific | Realistic using the best |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | | conservatively evaluated heating in the cable spreading room based on maximum ambient conditions and room heat load. Based on this calculation, the CSR will exceed the design basis criteria of 105°F in about 5.7 hours without HVAC. However, the CSR temperature is monitored in the control room, and an ARP exists to open doors and initiate portable fans if the CSR temperature reaches 100°F. Given the conservative nature of the 105°F criteria leading to actual component failures and the likelihood of success of enabling the ARP instructions, the Cable Spreading Room ia assumed to not require HVAC for continued operation of the components in the CSR. | requirements for CSR HVAC are included in the model. | estimates for expected plant response. This should not be a source of model uncertainty in most applications. |
| | | | | 7) EA-APR-95-023 [D-15] was performed to evaluate the affects of loss of battery room ventilation on room heat-up. This analysis showed that the battery room will exceed its design basis temperature in about 10 hours without HVAC. However, since the batteries are only nominally credited for four hours when chargers are unavailable and since the chargers can support all of the dc loads without the batteries then HVAC is assumed to not be required in the battery | 7) No specific requirements for Battery Room HVAC are included in the model. | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | | rooms. | | |
| | | | | 8) EA-APR-95-023 [D-15] evaluated control room heat-up during the 72-hour period following a loss of ventilation. The analysis assumed offsite power is available to maximize control room heat load, and assumed no heat transfer out of the room through concrete walls. A sensitivity study was performed to evaluate cases with and without the operators setting up portable emergency ventilation.<br>Without emergency ventilation, the temperature reaches the limit for habitability of 110°F at approximately 3.5 hours and exceeds the technical specification limit of 120°F in 15 hours. | 8) Failure to model HVAC control room cooling in the Palisades internal events PRA is not considered an issue:<br>• Because of the high design temperature limits of the major control room components,<br>• the general conservative modeling assumptions employed throughout the analysis,<br>• and the philosophy of the operators with respect to remaining in the control room during such an event.<br>Additionally, if emergency ventilation is provided via the proposed Honda Tempest portable exhaust fans using outside (auxiliary building) air at 95°F, the control room temperature quickly decreases, not rising above 101.5°F at the | The lack of representation may be a form of completeness uncertainty, but should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | | | end of the 72-hr transient. | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 10. Battery life calculations | Station Blackout events are important contributors to baseline CDF at nearly every US NPP. Battery life is an important factor in assessing a plant's ability to cope with an SBO. Many plants only have design basis calculations for battery life. Other plants have very plant/condition-specific calculations of battery life. Failing to fully credit battery capability can overstate risks, and mask other potentially contributors & insights. Realistically assessing battery life can be complex. | Determination of battery depletion time(s) and the associated accident sequence timing and related success criteria. | Design basis calculations indicate that about 4 hours of battery life is available depending on scenario specifics. Credit for 4 hours per division is utilized in the model for scenarios without chargers available. | 1) Given a plant SBO, battery depletion is expected to occur in about 4 hours with or without DC load shedding. | 1) Depletion of the batteries results in loss of control power and failure of most systems which rely on dc power. Continued operation of AFW is credited, however, as noted in topic number 4 above. | Realistic with slight conservative bias slant introduced by use of design basis calculations for the dc battery life determination. This should not be a source of model uncertainty in most applications. |
| 11. Number of PORVs required for bleed and feed – PWRs | PWR EOPs direct opening of all PORVs to reduce RCS pressure for initiation of bleed and feed cooling. Some plants have performed plant-specific analysis that demonstrate that less than all PORVs may be sufficient, depending on ECCS characteristics and initiation timing. | System logic modeling representing success criterion and accident sequence timing for performance of bleed and feed and sequences involving success or failure of feed and bleed. | Plant-specific evaluations using RETRAN confirm that 1 PORV is sufficient for success of once-through-cooling. | 1) One of two PORVs is required for success of once-through-cooling. | 1) The system fault tree model requires one of two PORVs for success of once-through-cooling. | The representation is realistic and should not be a source of model uncertainty in most applications. |
| 12. Containment sump / strainer performance | All PWRs are improving ECCS sump management practices, | Recirculation from sump (PWRs) or from the suppression pool | Sump strainer failure is modeled with a common cause failure of both | 1) Modifications installed in response to GSI-191 are considered to have restored | 1) The sump strainer failure rate is based on industry hourly failure | There is uncertainty associated with the likelihood of the |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | including installation of new sump strainers at most plants. There is not a consistent method for the treatment of ECCS sump performance.<br>All BWRs have improved their suppression pool strainers to reduce the potential for plugging. However, there is not a consistent method for the treatment of suppression pool strainer performance. | (BWRs) system modeling and sequences involving injection from these sources<br>(Note that the modeling should be relatively straightforward, the uncertainty is related to the methods or references used to determine the likelihood of sump strainer and common cause failure of the strainers.) | strainer assemblies. | sump performance to the original design basis. While detailed evaluations of debris generation and transport for a range of break sizes and locations were performed for the GSI-191 efforts, analysis and testing was performed to demonstrate strainer assembly success under very limiting conditions. Industry sponsored analyses and tests were not designed to provide data from which to infer sump strainer failure rates applicable to scenario specific break sizes, locations and sub-scenario effects (transport, chemical effects, bed formation dynamics, etc). | rate data for strainer plugging for the 24 hour mission time with a standard treatment of common cause. | common cause failure of the strainers. As such, this is identified as a candidate source of model uncertainty, but is included as part of the identification of common cause failures as a generic source of model uncertainty. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 13. Impact of failure of pressure relief | Certain scenarios can lead to RCS/RPV pressure transients requiring pressure relief. Usually, there is sufficient capacity to accommodate the pressure transient. However, in some scenarios, failure of adequate pressure relief can be a consideration. Various assumptions can be taken on the impact of inadequate pressure relief. | Success criterion for prevention of RPV overpressure<br>(Note that uncertainty exists in both the determination of the global CCF values that may lead to RPV overpressure and what is done with the subsequent RPV overpressure sequence modeling.) | The heat removal capacity of the steam generators at Palisades is such that a demand on pressurizer SRVs is not expected post reactor trip. This heat removal capability is routinely demonstrated on plant trips without a demand on pressurizer SRVs. Given this demonstrated heat removal capacity, PORV block valves are normally closed during operation. Demands on pressurizer SRVs would occur only during ATWS conditions, following a steam line break in which re-pressurization of the primary coolant system is allowed to occur, following SG dryout or if pressurizer SRV setpoints were to be set too low or drift down. | 1) The PRA assumes no demand on pressurizer SRVs during a transient unless setpoint drift results in a premature actuation. | 1) The potential for setpoint drift of SRVs is included in the consequential LOCA heading of the transient event trees. The probability of inadvertent SRV operation is based on generic operating experience for CE plants. | The potential for a premature demand on SRVs during transients is realistic and is based on generic operating experience. This should not be a source of model uncertainty in most applications. |
| | | | | 2) A demand on all three pressurizer SRVs is assumed under ATWS conditions. Failure of any of the SRVs to open is assumed to lead directly to core damage. | 2) An SRV heading is included in the ATWS event tree with a success criterion requiring all three SRVs to open (given that the PORV block valves are normally closed). | The treatment that all three pressurizer SRVs are needed following an ATWS is deemed acceptable with a slight conservative bias slant. This should not be a source of model uncertainty in most applications. |
| | | | | 3) Re-pressurization of the primary coolant system and a demand on SRVs is assumed following a steam line break in which the affected steam generator is isolated. | 3) The MSLB event tree assumes isolation of the affected SG will lead to re-pressurization of the primary coolant system unless the operators take action to limit the pressure rise. A demand on the pressurizer SRVs can then result in a small LOCA if an SRV sticks open. | Secondary cooling is assumed to be lost to the affected SG with certainty leaving only one SG available following a steam line break. This is a conservative assumption that leads to demands on the pressurizer SRVs unless the operator intervenes. This should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | | 4) SG dryout is assumed to lead to the need to initiate OTC, which obviates the need to consider SRV operation. | 4) Initiation of OTC is effectively a manually initiated demand on pressurizer pressure control components. | Initiating OTC precludes operation of the SRVs and is a realistic assumption. Ignoring the challenge to the pressurizer SRVs on failure to initiate OTC and the potential for one failing open is a conservative assumption from a CDF perspective. As such, this should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **Systems Analysis (to support meeting SY-C3)** | | | | | | |
| 14. Operability of equipment in beyond design basis environments | Due to the scope of PRAs, scenarios may arise where equipment is exposed to beyond design basis environments (w/o room cooling, w/o component cooling, w/ deadheading, in the presence of an un-isolated LOCA in the area, etc.). | System and accident sequence modeling of available systems and required support systems. | Credit for operation of systems beyond there design-basis environment is typically only taken if calculations exist to support their continued use. Exceptions are listed in the next column. | 1) EA-APR-95-023 [D-15] conservatively evaluated heating in the cable spreading room based on maximum ambient conditions and room heat load. Based on this calculation, the CSR will exceed the design basis criteria of 105°F in about 5.7 hours without HVAC. However, the CSR temperature is monitored in the control room, and an ARP exists to open doors and initiate portable fans if the CSR temperature reaches 100°F. Given the conservative nature of the 105°F criteria leading to actual component failures and the likelihood of success of enabling the ARP instructions, the Cable Spreading Room ia assumed to not require HVAC for continued operation of the components in the CSR. | 1) No specific requirements for CSR HVAC are included in the model. | Realistic using the best estimates for expected plant response. This should not be a source of model uncertainty in most applications. |
| | | | | 2) EA-APR-95-023 [D-15] showed that the battery room will exceed its design basis temperature in about 10 hours without HVAC. However, since the batteries are only nominally credited for four hours when chargers are unavailable and since the chargers can support all of the dc loads without the batteries then HVAC is assumed to not be required in the battery rooms. | 2) No specific requirements for Battery Room HVAC are included in the model. | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **Human Reliability Analysis (to support meeting HR-I3)** | | | | | | |
| 15. Credit For ERO | Most PRAs do not give much, if any credit, for initiation of the Emergency Response Organization (ERO), including actions included in plant-specific SAMGs and the new B5b mitigation strategies. The additional resources and capabilities brought to bear via the ERO can be substantial, especially for long-term events. | System or accident sequence modeling with incorporation of HFEs and HEP value determination in both the Level 1 and Level 2 models | Generally, credit for initiation of actions from the ERO is not taken in the Level 1 and Level 2 sequence analysis. | 1) For actions in which more than 75 minutes is available for diagnosis, a recovery factor on the cognitive portion of the HEPs can include credit for ERO (namely, Technical Support Center) response. | 1) Per the EPRI HRA methodology [D-20], the cognitive portion of the HEP can be reduced for HFEs where more than 75 minutes is available for diagnosis, but the execution portion of the HEP is not adjusted. However, it should be noted that the reduction in the cognitive portion of the HEP also considers recovery from other sources (self-check, STA, shift manager and extra crew) such that the ERO reduction factor is typically not utilized. | Slight conservative bias treatment in the ERO recovery factor value is typically not utilized. This should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **Internal Flooding (to support meeting IFPP-B3, IFSO-B3, IFSN-B3, IFEV-B3, and IFQU-B3)** | | | | | | |
| 16. Piping failure mode | One of the most important, and uncertain, inputs to an internal flooding analysis is the frequency of floods of various magnitudes (e.g., small, large, catastrophic) from various sources (e.g., clean water, untreated water, salt water, etc.). EPRI has developed some data, but the NRC has not formally endorsed its use. | Likelihood and characterization of internal flooding sources and internal flood event sequences | Potential sources of floods (pipes, tanks, etc.) were identified from plant walk downs and evaluations of plant drawings. Pipe diameters and lengths were determined from the walk downs and from isometric drawings.<br>One by one, each piping flood source was assumed to suffer a catastrophic guillotine break (i.e., the maximum size break). Floods were assumed to proceed without mitigation, and all equipment in an area was assumed to be impacted (submerged) by the flood. Pipes connected to tanks identified as flood sources were assumed to rupture in such a way that the entire inventory of the tank would drain into an area.<br>Plant specific analyses were used to characterize the flow rates associated with the guillotine breaks. The flow rates were used to assign the sources into categories of "spray" – less than 100 gpm, "flood" - 100 gpm or greater, up to 2000 gpm, and "major flood"- 2000 gpm or | 1) Plant wide flood initiating event frequency is comprised of pipe-related failures and maintenance contribution.<br>2) Plant wide flood initiating event frequency can be derived using generic (industry) data for floods, updated with plant-specific experience.<br>3) Maintenance contribution is equal to the plant wide frequency minus the pipe failure frequencies (derived using EPRI 1013141 [D-22]).<br>4) Maintenance contribution is comprised of "at power" and "at shutdown" parts – only the "at power" part contributes to flood frequency for at power internal flood evaluation.<br>5) For flood and major flood categories, pipe breaks are guillotine breaks.<br>6) For pipe breaks, flooding continues without mitigation, and with infinite supply.<br>7) In general, flooding exceeds capacity of drain system.<br>8) All equipment in a given room or flood area is | 1) Multiple initiating events are defined and quantified, based on system, diameter, and consequences of flood.<br>2) Propagation between rooms occurs because flood proceeds without mitigation, flood level rises, and doorways open due to water pressure. Thus, more equipment becomes involved (submerged) by flood.<br>3) Maintenance contribution (and thus, overall flood frequency) is conservative because no credit for operator recovery is included.<br>4) Assumption of "infinite source capacity" results in the inclusion of initiating events (flood sources, e.g., pipes) that might otherwise be | Impacts of flooding, both in terms of equipment impacted and in terms of initiating event frequency, are judged to be bounding (conservative) due to conservative treatment of initiating event frequency and flood progression.<br>• Guillotine break assumed<br>• Unmitigated flooding<br>• Essentially "infinite" source capacity<br>• Multiple diameters of pipes contribute to flooding (and thus add to overall frequency)<br>• Maintenance contribution separate from and in addition to piping failures<br>• Maintenance is assumed to occur at power, and is assumed to involve steps that could actually produce a flood if performed incorrectly equipment than if flood was terminated. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | greater. | assumed to be submerged as a result of a flood (or major flood) within the area. | screened. | |
| | | | When pipe lengths were known for a flood source, pipe failure frequencies were calculated using the EPRI report "Pipe Rupture Frequencies for Internal Flooding PRAs, Revision 1," Technical Report 1013141, Final Report, March 2006 [D-22]. The appropriate frequency for pipe rupture or leak (spray) on a per foot, per year basis was chosen based on system, pipe diameter, and category (flood, major flood, spray). This was multiplied by number of feet.<br>When pipe lengths were not known (due to inaccessibility of areas, for example), pipe rupture frequencies were taken from EA-PSA-RI-ISI-00-INDIRECT ANALYSIS, "RI-ISI Indirect Effects Evaluation," Rev. 0, August 12, 2000 [D-23], for the pipe segment(s) comprising the flood source. (Note – there were very few of these instances.) | 9) Spray events have only localized impacts.<br>10) No maintenance contribution to spray events – maintenance staff is assumed to halt the event immediately upon its occurrence.<br>11) For maintenance contribution to flooding, no maintenance staff recovery (i.e., action to halt the flood) is credited.<br>12) Maintenance activity on any component identified as a candidate is assumed to take place, and is assumed to contain steps that could place the component in jeopardy of producing a flood.<br>13) If a pipe break flood initiating event group is comprised of pipes of a range of diameters, the diameter associated with the highest per foot, per year failure frequency is chosen from EPRI 1013141. | | • Propagation between rooms results in substantially more equipment being impacted (submerged and failed) than if flood was mitigated or if finite source assumed.<br>In summary, the frequency of plant wide flooding, pipe rupture, and of maintenance errors resulting in flood at power, are candidate sources of model uncertainty. However, as noted above, the plant specific approach utilized is believed to produce results that bound the uncertainty. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| **LERF Analysis (to support meeting LE-G4)** | | | | | | |
| 17. Core melt arrest in-vessel | Typically, the treatment of core melt arrest in-vessel has been limited. However, recent NRC work has indicated that there may be more potential than previously credited. An example is credit for CRD in BWRs. | Level 2 containment event tree sequences | No credit is given for recovery of offsite power after core damage but before a radioactive release since the likelihood may be small and the time window will vary for different scenarios. | Power recovery during the time window between core damage and radioactive release is unlikely. | Because no credit is taken, there is no top event in the model to account for power recovery between core damage and radioactive release. | Lack of credit for power recovery is slightly conservative. Therefore, this should not be a source of model uncertainty in most applications. |
| 18. Thermally induced failure of hot leg/SG tubes – PWRs | NRC analytical models and research findings continue to show that TI-SGTR is more probable than predicted by the industry. There is a need to come to agreement with NRC on the thermal hydraulics modeling of TI SGTR. | Level 2 containment event tree sequences | The Palisades Level 2 analysis used in this analysis is considered a much more detailed analysis than that described in the guidance from WCAP-16341-P [D-21].<br>A Palisades specific version of MAAP was created to address the integrated effect of the plant-specific features on overall containment performance and fission product release. Some significant modifications and enhancements added to MAAP in development of CPMAAP included:<br>-Elevation Head in Accumulator Discharge Model.<br>-Tellurium (Te) Release During Direct Containment Heating Model. | Conditional probabilities for TI-SGTR are given in Table E-17 of the WCAP are assumed to be applicable. The analysis uses the average tube degradation values. | TI-SGTR can be a contributor to LERF at Palisades. Therefore, variations in the likelihood of TI-SGTR will have a direct effect on the calculation of LERF. Uncertainties are not expected to affect the structure of the model. | Induced steam generator tube ruptures are a significant issue for most PWRs, including Palisades. Therefore, the likelihood of TI-SGTR is identified as a candidate source of model uncertainty. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | -Non-Cladding Hydrogen Source Model.<br>-Steam Generator Level Correction at Full Power.<br>-Improved Numeric/Logic for Modeling Solid Steam Generator.<br>-Dead band Model for Secondary Relief Valves.<br>-PCS Insulation Melting Model.<br>-PCS Pressure Boundary Creep Rupture Model - A thermal creep rupture model based on the Larsen-Miller Parameter (LMP) method was added to CPMAAP to evaluate the response of the hot legs, surge line, and steam generator tubes.<br>-Hydrogen Detonation Cell Width Model.<br>-Core Debris Flow to the Auxiliary Building Model.<br>-Palisades Specific ESF Modeling.<br>These hard-coded modeling changes were implemented some 20 years ago. As a point of comparison the NRC's recently completed STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES (SOAR CA) | | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | | | just included the Larsen-Miller Parameter (LMP) method in an update to the MELCOR code. | | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 19. Vessel failure mode | The progression of core melt to the point of vessel failure remains uncertain. Some codes (MELCOR) predict that even vessels with lower head penetrations will remain intact until the water has evaporated from above the relocated core debris. Other codes (MAAP), predict that lower head penetrations might fail early. The failure mode of the vessel and associate timing can impact LERF binning, and may influence HPME characteristics (especially for some BWRs and PWR ice condenser plants). | Level 2 containment event tree sequences | Four possible causes for early containment failure at the time of reactor vessel breach are addressed in this analysis – ex-vessel steam explosion, hydrogen burn, direct containment heating, and alpha mode failure (in-vessel steam explosion). While following the general guidance in WCAP-16341-P, this analysis uses a more plant-specific approach to early containment failure. | 1) The ex-vessel steam explosion is a greater issue for free-standing reactor cavities (as opposed to excavated cavities). Because Palisades is a free-standing cavity, containment failure due to ex-vessel steam explosions are assigned a likelihood of 0.01. | 1) Ex-vessel steam explosions contribute to containment failure for sequences with a wet reactor cavity at the time of vessel failure. | Probability is generally a low contributor and slightly conservative, so should not be a source of model uncertainty in most applications. |
| | | | | 2) Scenarios that lead to hydrogen burns at plants like Palisades are limited to about 50% zirconium oxidation for CFE5- and CFE3H-type scenarios and 40% for CFE1-type scenarios. However, probabilities for containment failure are based on the Palisades detailed analyses that range from 0.01 to 0.0001. | 2) Hydrogen burn sequences at 40% oxidation can be contributors to LERF at Palisades. Therefore, variations in the likelihood of containment failure due to hydrogen burn will have a direct effect on the calculation of LERF. | Hydrogen burn can be a contributor to LERF at Palisades. Therefore, the values utilized for containment failure due to hydrogen burns are identified as a candidate source of model uncertainty. |
| | | | | 3) Direct containment heating is also addressed by the Palisades analysis [D-24]. The conditional containment failure probabilities due to direct containment heating is 0.005 to cover all scenarios | 3) DCH is not expected to be a significant contributor to LERF. The value may be conservative compared to previous Palisades' analysis, but is used at this point for consistency with the WCAP. | Probability is generally a low contributor and slightly conservative, so should not be a source of model uncertainty in most applications. |
| | | | | 4) Per reference [D-24] the alpha mode failure (due to in-vessel steam explosion) presents a low probability (1.0E-4) of containment failure at low pressures and is negligible at high pressures. | 4) Alpha mode failure is not a significant contributor to any sequence. | Probability is now accepted as a very low contributor, so should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 20. Ex-vessel cooling of lower head | The lower vessel head of some plants may be submerged in water prior to the relocation of core debris to the lower head. This presents the potential for the core debris to be retained in-vessel by ex-vessel cooling. This is a complex analysis impacted by insulation, vessel design and degree of submergence. | Level 2 containment event tree sequences | The Palisades' in-core instrumentation enters the reactor vessel through the upper head. Therefore, there are no penetrations in the lower reactor vessel head. The lower reactor vessel head is also un-insulated. If the outside of the lower head is submerged in water and the Primary System pressure is sufficiently low, calculations performed for the Palisades indicate that sufficient heat can be removed through the lower head wall to prevent vessel failure under most circumstances. | 1) Success of the cavity flooding system at low pressures requires successful containment spray injection and recirculation along with successful operation of the flooding system to route water to the reactor cavity. | 1) Success of the cavity flooding system is modeled to require CSS operation and routing of the water to the cavity by existing piping. | Operation of CSS is based on system fault trees. The likelihood of CFS properly routing water to the cavity is assumed, but is not a key factor. These factors should not be a source of model uncertainty in most applications. |
| | | | | 2) Once water fills the cavity and surrounds the lower portion of the reactor vessel, sufficient heat transfer from the molten fuel to the water in the reactor cavity must occur. In high pressure sequences, it is assumed that heat transfer is insufficient. | 2) Given a flooded cavity, a probability that adequate thermal-hydraulic conditions exist (under low pressures) is applied to capture phenomenological uncertainty in the ability to retain the molten core in the vessel. Under high pressures, no credit is given. | Thermal hydraulic uncertainty concerning the ability to cool the lower head and molten core from outside the vessel may be significant. Therefore, the likelihood that the debris is not coolable when the RPV is at low pressure and water is in the cavity is identified as a candidate source of model uncertainty. |
| 21. Core debris contact with containment | In some plants, core debris can come in contact with the containment shell (e.g., some BWR Mark Is, some PWRs including free-standing steel containments). Molten core debris can challenge the integrity of the containment boundary. Some analyses have | Level 2 containment event tree sequences | Direct contact of core debris with the containment shell is not an issue at Palisades. However, Palisades has a related, unique issue regarding the design of its reactor cavity and containment sump. The ESF sump is located directly below the reactor cavity floor (the floor of the reactor cavity is the ceiling | 1) The plant modification promotes freezing of a small mass of the debris, effectively plugging the drainpipes. This should prevent immediate debris transport to the sump. If the debris cannot be cooled within the confines of the cavity, it may later melt through the cavity floor (or the floor could ultimately fail due the corium weight), but the delay will greatly extend the time | 1) Sequences leading to this situation are identified as CAB releases (Core-to-Auxiliary Building) and are treated as non-LERF releases. | Based on the plant modification, there is high confidence that CAB scenarios will not lead to LERF. Therefore, core debris contact with containment should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| | demonstrated that core debris can be cooled by overlying water pools. | | of the sump). There are two 1-inch drains, which connect the sump to the reactor cavity through the ceiling of the sump. During previous analyses, this was identified as a potential LERF path via the Auxiliary Building, and plant modifications were performed to correct the issue. These drains have been filled with ceramic beads to slow the accident progression so that it is no longer an early release path. | available for evacuation and other mitigation measures. | | |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 22. ISLOCA IE Frequency Determination | ISLOCA is often a significant contributor to LERF. One key input to the ISLOCA analysis are the assumptions related to common cause rupture of isolation valves between the RCS/RPV and low pressure piping. There is no consensus approach to the data or treatment of this issue. Additionally, given an overpressure condition in low pressure piping, there is uncertainty surrounding the failure mode of the piping. | ISLOCA initiating event sequences | High to low pressure interfaces potentially leading to an ISLOCA include LPSI and SDC.<br>High pressure interfaces considered include HPSI and charging. | 1) For LPSI, the check valves between the system and the primary coolant is considered as the initiator and given a year-long mission time. The intervening in series MOVs and check valves are assumed to have a quarterly surveillance interval based on periodic surveillance tests of their operability and integrity. Consideration is given to the exposure to primary coolant conditions that occurs during stroke testing of ECCS valves.<br>For SDC, the isolation MOVs are electrically disabled. The inboard valve is given a yearly mission time to determine a rupture probability while the outboard valve is given a 24 hr mission time as there is a relief valve in between that would indicate failure of the inboard valve.<br>2) On exposure to primary coolant pressure, piping rupture probabilities are based on a statistical best fit that relates hoop stress in the pipe to the probability of failure considering pipe thickness, diameter and a given primary coolant system pressure.<br>Similar approaches are taken for high pressure interfaces with the primary coolant system including HPSI and charging. | 1) The approach taken to assessing the probability of failure of interfacing components is based on the exposure time for interfacing components to primary coolant system conditions and actual operation and surveillance testing.<br>2) A best fit of hoop stress versus failure probability is used to determine piping failure probability outside containment assuming a 2250 psi primary coolant system pressure whether for high or low pressure piping.<br>Unique contributions from each flow path are incorporated into the ISLOCA fault tree that is directly integrated into the overall model. | The use of actual exposure times and surveillance intervals provides a realistic estimate of interfacing component failure probabilities.<br>The use of a correlation between hoop stresses and failure probability for piping outside containment is realistic, but slightly conservative given the primary coolant system pressure assumed in developing the correlation.<br>The approach for the ISLOCA frequency determination is considered to represent the proper treatment given the current understanding of these issues. This should not be a source of model uncertainty in most applications. |

**Table D-1: Issue Characterization for Sources of Model Uncertainty for Palisades (QU-F4 and LE-F3)**

| Topic (to meet QU-E1) | Discussion of Issue | Part of Model Affected | Plant-Specific Approach Taken | Assumptions Made (to meet QU-E2) | Impact on Model (to meet QU-E4) | Characterization Assessment |
|---|---|---|---|---|---|---|
| 23. Treatment of Hydrogen combustion in BWR Mark III and PWR ice condenser plants | The amount of hydrogen burned, the rate at which it is generated and burned, the pressure reduction mitigation credited by the suppression pool, structures, etc. can have a significant impact on the accident sequence progression development. | Level 2 containment event tree sequences | Hydrogen burns can challenge the integrity of the containment by creating high pressure excursions. The amount of hydrogen released into containment depends upon the amount of core damage at the time of vessel failure. Scenarios that lead to hydrogen burns at plants like Palisades are limited to about 50% zirconium oxidation, the probability of early containment failure at Palisades ranges from .01 to 0.0001 due to hydrogen burn. Steam inerting that prevents hydrogen combustion due to failure of containment sprays is also considered. | 1) Failure of containment spray that allows a high steam concentration in containment prevents hydrogen combustion.<br>MAAP calculations support steam fractions greater than 55% during representative scenarios, which is sufficient to prevent hydrogen combustion. | 1) Failure of containment due to hydrogen burn is modeled as only possible in conjunction with failure of containment sprays. | Hydrogen combustion can be a contributor to LERF at Palisades. Therefore, this is identified as a candidate source of model uncertainty. |

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