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U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555
ATTN: David B. Matthews, Director
Division of New Reactor Licensing

SUBJECT: COMANCHE PEAK NUCLEAR POWER PLANT, UNITS 3 AND 4
DOCKET NUMBERS 52-034 AND 52-035
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION NO. 5731
(SECTION 19)

Dear Sir:

Luminant Generation Company LLC (Luminant) submits herein the response to Request for Additional Information (RAI) No. 5731 (CP RAI #218) for the Combined License Application for Comanche Peak Nuclear Power Plant Units 3 and 4. This RAI addresses whether the Risk Managed Technical Specification methodology being developed provides reasonable assurance that all applicable guidance requirements will be met.

Should you have any questions regarding this response, please contact Don Woodlan (254-897-6887, Donald.Woodlan@luminant.com) or me.

There are no commitments in this letter.

I state under penalty of perjury that the foregoing is true and correct.

Executed on July 14, 2011.

Sincerely,

Luminant Generation Company LLC

Donald R. Woodlan for

Rafael Flores

Attachment: Response to Request for Additional Information No. 5731 (CP RAI #218)

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NRO

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Comanche Peak, Units 3 and 4

Luminant Generation Company LLC

Docket Nos. 52-034 and 52-035

RAI NO.: 5731 (CP RAI #218)

SRP SECTION: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation

QUESTIONS for PRA and Severe Accidents Branch (SPRA)

DATE OF RAI ISSUE: 5/5/2011

QUESTION NO.: 19-5

In RAI 3287 (RAI Letter Number 26), Question 3, the NRC staff requested that the applicant provides a roadmap with specific steps and supporting information, as necessary, that addresses (1) potential improvements of the US-APWR design certification (DC) PRA models, (2) inclusion of site-specific and detailed design models and as-built information, (3) PRA capability that meets Regulatory Guide 1.200 for all ASME supporting requirements, except for the ones that need plant-specific operational experience, as well as application-specific guidance, and (4) modeling uncertainties and strategies for addressing them (e.g., through bounding assumptions or specific compensatory actions) to be considered in conjunction with the specific risk-informed programs. In its response dated September 22, 2009, Luminant provided a timeline of high-level actions that will be taken by the COL licensee to ensure that a plant-specific PRA model, which meets all applicable guidance requirements, will be available before the plant goes in operation. Even though the provided timeline and associated actions in the response to RAI 3287 Question 3 are reasonable, the staff has identified the need for better definition of the issues and for more detailed description of the timeline actions to provide reasonable assurance at the COL stage that all applicable guidance requirements will be met before the plant transitions to operation.

During a public meeting, Luminant and the staff agreed to address these issues by the development of a "methodology" which will be included in the TS Administrative Controls and which will provide reasonable assurance, at the time of COL license issuance, that all applicable guidance requirements will be met before the plant transitions to operation. Such reasonable assurance would be attained by insights obtained through analysis and interpretation of results of specific examples. Appropriately selected examples of Risk Managed Technical Specifications (RMTS) Initiative 4b and Surveillance Frequency Control Program (SFCP) Initiative 5b applications would provide valuable insights, such as regarding the type and impact of uncertainties involved, as well as their treatment, in the risk-informed decision-making process. These insights should be used to focus, improve and strengthen the Comanche Peak Nuclear Power Plant (CPNPP)-specific "methodology" document to attain reasonable assurance that all applicable guidance requirements will be met.

Regarding RMTS Initiative 4b, the staff has identified the following plant configuration scenarios whose demonstrative implementation will provide valuable insights which will help determine whether the RMTS "methodology" being developed provides reasonable assurance that all applicable guidance requirements will be met:

Case 1

The following equipment is out for planned maintenance: (1) One essential service water (ESW) / component cooling water (CCW) train (assume train "B" (in standby)); (2) One Class 1E gas turbine generator (GTG) (assume GTG "B"); (3) One alternate alternating current (AAC) GTG (assume AAC GTG "B"); and (4) Diesel-driven fire suppression pump.

While this equipment is out, as allowed by TS with no limiting condition for operation (LCO) in effect, a second ESW/CCW train fails (assume train "D" in standby) and Condition A of LCO 3.7.7 and LCO 3.7.8 is entered. The completion time (CT) is 72 hours. Assume that none of the two ESW/CCW trains can be restored within 72 hours and the requirements of Specification 5.5.18 (RMTS Initiative 4b) are applied and a risk-informed CT is calculated.

Case 2

The following equipment is out for planned maintenance: (1) One Class 1E GTG (assume GTG "A"); (2) One AAC GTG (assume AAC GTG "A"); (3) One safety injection system (SIS) pump (assume pump "A"); (4) One containment spray pump (assume pump "A"); and (5) One turbine- driven (T-D) emergency feedwater (EFW) pump (assume pump "A" and train cross-tie per LCO 3.7.5 is met).

While this equipment is out, as allowed by TS with no LCO in effect, a second Class 1E GTG (assume train "B") is found inoperable and Condition B of LCO 3.8.1 is entered. The completion time (CT) is 72 hours. Assume that none of the two inoperable Class 1E GTGs can be restored within 72 hours and the requirements of Specification 5.5.18 (RMTS Initiative 4b) are applied and a risk-informed CT is calculated.

Case 3

The following equipment is out for planned maintenance (preventive and/or corrective): (1) One train (assume train "A") of each standby safety system; (2) One AAC GTG (assume AAC GTG "A"); (3) DAS EFW system Actuation and emergency core cooling system (ECCS) Actuation functions (Condition A of LCO 3.3.6 requires DAS to be restored within 30 days); and (4) One T-D EFW pump (assume pump "A" and train cross-tie per LCO 3.7.5 is met).

While this equipment is out, as allowed by TS with no LCO in effect (except for DAS which has a CT of 30 days), a second safety injection system (SIS) pump (assume pump "B") is found inoperable (assume one hour after LCO 3.3.6 went into effect) and Condition A of LCO 3.5.2 is entered. The completion time (CT) for LCO 3.5.2 is 72 hours.

Assume that none of the two inoperable SIS pumps can be restored within 72 hours and the requirements of Specification 5.5.18 (RMTS Initiative 4b) are applied and a risk-informed CT is calculated.

For each of the three cases please use the currently available PRA model to calculate risk-informed completion times (RICTs) and submit the results and associated discussion to NRC for the staff's review. Please include a separate discussion for each case which includes the following: (1) major assumptions used in calculations and regarding risk management actions; (2) the calculated core damage frequency (CDF) and large release frequency (LRF) (or large early release frequency (LERF) increase versus time; (3) the calculated incremental core damage probability (ICDP) and ILERP versus time; (4) list of significant contributing cutsets to the increased risk (at least for CDF) with adjusted probabilities reflecting equipment outages (e.g., impact of equipment outages on baseline PRA common-cause failure probabilities and initiating event frequencies); (5) treatment of uncertainties in decision making.

In addition, please discuss additional plant configurations, or configuration controls associated with existing RMTS guidance or other programs, that Luminant believes may provide useful information in developing the RMTS "methodology," if applicable.

Regarding SFCP Initiative 5b, the staff finds that a parametric sensitivity study where every surveillance testing interval shorter than 92 days for risk significant equipment in the SFCP is increased by several factors (e.g., 2, 5 and 10 times) would provide valuable insights and help the staff reach conclusions regarding the adequacy of the SFCP "methodology" being developed. Please perform a parametric sensitivity study as described above and use a few representative examples to illustrate how the guidance in NEI-04-10 will be implemented and how PRA uncertainties will be addressed for these specific cases in the decision-making process.

ANSWER:

Regarding RMTS Initiative 4b, sensitivities studies have been performed for the cases requested by the staff. In the cases below, some of the trains out of service were chosen differently from the case requested by the staff so that the asymmetric modeling assumptions in the PRA would not result in artificially optimistic results. For instance, in Case 2, the equipment in Train D was assumed to be out of service rather than Train A. This is because for initiating events that cause failure of safety functions, such as partial loss of CCW, the PRA model assumes that Train A (and also B for the case of partial loss of CCW) is affected by the initiating event and would be inoperable during the accident. The analysis conditions and results of the sensitivity studies are shown below.

<u>Case 1</u>

a. Major assumptions

The plant is operating with the following equipment out of service:

- o ESW/CCW Train D (Train B in standby)
- o Class 1E GTG D
- o AAC GTG B
- o Diesel-driven fire protection water supply pump

The cross-tie between CCWS Train C and D is assumed to be isolated by closing the tie line valves (NCS-MOV-007D-S, NCS-MOV-020D-S), since these valves will be closed when the ESW/CCW trains are out of service.

At time t=0, ESW/CCW Train B fails (Train D in standby).

The plant enters Condition A of LCO 3.7.7 and LCO 3.7.8 is entered. The cross tie-line between CCWS Train A and B is isolated by closing tie line valves NCS-MOV-007B-S, NCS-MOV-020B-S immediately after Condition A of LCO 3.7.7 and LCO 3.7.8 is entered. As a risk management action, additional maintenance outage for the Class 1E GTG, AAC or fire protection water supply system is prohibited.

The initiating event frequency of loss of CCW event was re-estimated considering the outage of equipment, and was used as the input to the PRA model.

b. CDF and LRF

The CDF and LRF for the plant configuration before Condition A of LCO 3.7.7 and LCO 3.7.8 are entered are 3.2E-06/RY and 9.2E-07/RY, respectively. After entering LCO 3.7.7 and LCO 3.7.8, CDF and LRF increase to 9.9E-05/RY and 8.0E-06/RY, respectively. The calculated CDF and LRF values are summarized in Table 1.

c. ICDP and ILRP

The calculated time variations of ICDP and ILRP values are show in Figures 1 and 2, respectively. The calculated ICDP and ILRP values at the time of reaching the 30-day backstop are 8.1E-07/RY and 6.5E-07/RY, respectively.

d. Significant contributing cutsets

The ten significant cutsets that contribute to increased core damage risk are shown in Table 2. The most significant contributor is the cutset that represents the failure to establish alternate CCW following the loss of CCW initiating event.

The initiating event frequency for loss of CCW has been adjusted to reflect the operability of the CCW and ESW pumps. The CCF group and the MGL parameters for the GTGs, CCW pumps and ESW pumps have not been adjusted.

The CCF groups have not been adjusted for the sensitivity case since it will not affect the calculated total CDF and LRF. Even though basic events representing CCF events of group sizes exceeding the number of available components would appear in the cutsets, the total CDF and LRF, which is a result of quantification of the cutset lists, will be the same with the case where the CCF group sizes have been adjusted. The treatment of CCF groups is the same for Cases 2 and 3.

e. Treatment of Uncertainties

Human error probabilities to initiate alternate component cooling dominate the ICDP value. The calculated risk-informed completion time would be shorter than the 30 days backstop if a human error probability ten times the base case has been assumed. Sensitivity studies assuming higher human error probabilities (such as factors of two and ten) would be candidates for uncertainty analyses when using the PRA results for decision making in the application stage of risk informed completion times.

Case 2

a. Major assumptions

The plant is operating with the following equipment out of service:

- Class 1E GTG D
- AAC GTG B
- SIS pump D
- Containment spray/residual heat removal (CS/RHR) pump D
- T-D EFW pump D

All EFW pump discharge cross-connect line isolation valves are opened in accordance with Condition A of LCO 3.7.5.

At time t=0, the Class 1E GTG C fails.

The plant enters Condition B of LCO 3.8.1. As a risk management action, additional maintenance outage for the CCWS, ESWS, AAC or fire protection water supply system is prohibited.

b. CDF and LRF

The CDF and LRF for the plant configuration before Condition B of LCO 3.8.1 is entered are 6.1E-06/RY and 1.1E-06/RY, respectively. After entering LCO 3.8.1, CDF and LRF increase to 3.2E-05/RY and 1.9E-05/RY, respectively. The calculated CDF LRF values are summarized in Table 1.

c. ICDP and ILERP

The calculated time variations of ICDP and ILRP values are show in Figures 1 and 2, respectively. The ILRP reaches the criterion of 1E-06 at 20 days after entering Condition A of LCO 3.8.1.

d. Significant contributing cutsets

The ten significant cutsets that contribute to increased core damage risk are shown in Table 3. The most significant contributor is the cutset that represents a consequential LOOP given a reactor trip following a partial loss of CCW initiating event. This cutset corresponds to an event where functions of Trains C and D of the CCWS are lost as an initiating event (partial loss of CCW) and a LOOP occurs resulting from grid perturbation caused by the plant trip. In this event, electrical power to Class 1E buses C and D cannot be restored since the supporting Class 1E GTGs and AAC are out of service. Safety functions of Trains A and B are not available as a result of the initiating event. Since RCP seal cooling or injection is not available, eventually loss of RCP seal cooling occurs and the core will be uncovered.

The most significant contributing cutset to increase in LRF is the same as the one for CDF. This cutset dominates 90% of the LRF under the plant condition after entering Condition A of LCO 3.8.1.

e. Treatment of Uncertainties

Initiating event frequency of partial loss of CCW event and the human error probability to change over the EFW pump water source are key uncertainties that impact the ICDP value. The calculated risk-informed completion time is sensitive to these elements. Sensitivity studies assuming higher human error probabilities, as well as applying different initiating event frequencies and modeling asymmetries for partial loss of CCW event, will be candidates for uncertainty analyses when using the PRA results for decision making in application stage of risk informed completion times.

Case 3

a. Major assumptions

The plant is operating with the following equipment/functions out of service:

- SIS pump D
- CS/RHR pump D
- Class 1E GTG D
- DAS EFW actuation function
- DAS ECCS pump actuation function
- T-D EFW pump D
- AAC GTG B

All EFW pump discharge cross-connect line isolation valves are opened in accordance with Condition A of LCO 3.7.5.

At time t=0, SIS pump C is found inoperable.

The plant enters Condition B of LCO 3.5.2. As a risk management action, additional maintenance outage for the CS system, GTGs, and EFW system is prohibited.

b. CDF and LRF

The CDF and LRF for the plant configuration before Condition A of LCO 3.5.2 is entered are 9.5E-06/RY and 1.4E-06/RY, respectively. After entering LCO 3.5.2, CDF and LRF increase to 6.3E-05/RY and 2.2E-06/RY, respectively. The calculated CDF LRF values are summarized in Table 1

c. ICDP and ILERP

The calculated time variations of ICDP and ILRP values are show in Figures 1 and 2, respectively. The calculated ICDP and ILRP values at the time of reaching the 30-day backstop are 5.1E-06 and 1.7E-07, respectively.

d. Significant contributing cutsets

The ten significant cutsets that contribute to increased core damage risk are shown in Table 4. The most significant contributor is the cutset that represents a consequential LOOP given a reactor trip following a partial loss of CCW initiating event. This cutset corresponds to an event where functions of Trains C and D of the CCWS are lost as an initiating event (partial loss of CCW) and a LOOP occurs resulting from grid perturbation caused by the plant trip. In this event, electrical power to Class 1E buses C and D cannot be restored since the supporting Class 1E GTGs and AAC are out of service. Safety functions of Trains A and B are not available as a result of the initiating event. Since RCP seal cooling or injection is not available, eventually, loss of RCP seal cooling occurs, and the core will be uncovered.

The most significant contributing cutset to increase in LRF is partial loss of CCW initiating event followed by consequential LOOP.

e. Treatment of Uncertainties

Initiating event frequency of partial loss of CCW event is the key source of uncertainty that impact the ICDP and ILRP values. Asymmetric modeling of the partial loss of CCW event and LOCA initiating events lead to higher ICDP and ILRP values for this analysis case, because the initiating events are assumed to occur in the trains where equipments are available. Sensitivity studies assuming different initiating event frequencies for partial loss of CCW event, and impact of asymmetric modeling of the initiating event will be candidates for uncertainty analyses when using the PRA results for decision making in the application stage of risk informed completion times.

Regarding SFCP, sensitivities studies have been performed for three SRs. The SRs were chosen considering risk importance of the equipment and test intervals.

- Reactor trip breakers (SR 3.3.1.3)
- Class 1E GTGs (SR 3.8.1.2)
- Class 1E batteries (SR 3.7.6.2 and SR3.8.6.3)

The internal events at-power PRA model is used to determine the risk impact of test interval extensions. The affect of test interval extension on single failures as well as on common cause failures was considered. The following assumptions were applied during quantification:

- Unreliability of equipment increases commensurately with test intervals. All failures are assumed to occur during standby.
- Affects of extended test intervals on failures to run are uncertain. In the sensitivity case for Class 1 GTGs, it was conservatively assumed that the probabilities of failures to run also increase commensurately with test intervals.
- Unavailability due to test and maintenance assumed unchanged. Longer test intervals may result in a decrease of unavailability due to tests, but this factor was not considered in the calculation.

The sensitivity case studies performed for these SRs are summarized in Table 6. The results are as follows:

- Reactor trip breakers (SR 3.3.1.3)

  The reactor trip breaker actuation test interval was assumed to be extended to 124 days, which is a factor of two longer than the TS requirement of 62 days. Extension of the test interval was assumed to impact the mechanical portion, shunt trip portion, and UV trip portion of the reactor trip breakers. If test interval has been increased by a factor of two, the increase in CDF is estimated to be 1.0E-08/RY. The increase in core damage risk is three magnitudes lower than the 1.0E-05/RY criteria for ΔCDF used in the NEI guidance.

  In the case where reactor trip breakers have failed, MG-SET, which can be operated by the DAS can be used to initiate reactor trip. Uncertainty associated with the reliability of reactor trip by DAS can affect the results of the sensitivity case. However, considering the large margin against the ΔCDF criteria identified in this sensitivity case, it is anticipated that uncertainties associated with DAS reliability will not affect the conclusion.

- Class 1E GTGs (SR 3.8.1.2)

  The GTG start test interval, which is 31 days in the TS, was assumed to be extended to 62 days and to 6 months. Extension of the test interval was assumed to impact the reliabilities of GTGs to start and to continue to run. If test interval is extended to 6 months, which is a factor of six longer than the TS requirement, the increase in CDF is estimated to be 2.3E-06/RY. This CDF increase is approximately one fourth of the ΔCDF criteria used in the NEI guidance and is acceptable.

  GTGs are new types of components introduced in the US-APWR design and uncertainty reliability data associated lack of operating experience should be carefully considered when applying test interval changes. Sensitivity study results show that extension of GTG start test intervals may result in considerable risk increase relative to the base case CDF. Drastic test interval extension should not be performed before GTG reliability data is accumulated.

- Class 1E batteries (SR 3.8.6.2 and SR 3.8.6.2)

  Battery start test interval, which is 31 days in the TS, was assumed to be extended to 62 days. Extension of the test interval was assumed to impact the reliabilities of batteries to achieve its function during LOOP events. If test interval is extended to 62 days, which is a factor of two longer than the TS requirement, the increase in CDF is estimated to be less than 1.0E-08/RY. The CDF increase is three orders of magnitude below the ΔCDF criteria used in the NEI guidance.

Table 1 - Summary of CDF and LRF Values for each RMTS Sensitivity Case

		t<0 (before LCO violation)	t≥0
Case 1	CDF (/RY)	3.2E-06	9.9E-05
	LRF (/RY)	9.2E-07	8.0E-06
Case 2	CDF (/RY)	6.1E-06	3.2E-05
	LRF (/RY)	1.1E-06	1.9E-05
Case 3	CDF (/RY)	9.5E-06	6.3E-05
	LRF (/RY)	1.4E-06	2.2E-06

Table 2 - Significant Contributing Cutsets to Increased Core Damage Risk of Case 1

Rank	Cutsets Freq. (/RY)	Percent (%)	Cutsets	Basic Event Name
1	5.5E-05	55.6	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			ACWOO02CT-DP2	(HE) FAIL TO ESTABLISH THE ALTERNATE CCWS BY NON-ESSENTIAL CHILLED WATER SYSTEM COOLING TOWER
			ACWOO02FS	(HE) FAIL TO ESTABLISH THE ALTERNATE CCWS BY FIRE PROTECTION WATER SUPPLY SYSTEM
			RCP----SEAL	RCP SEAL LOCA
2	4.4E-06	4.4	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWCF2PTAD001AD-ALL	EFS-MPP-001A,D (EFW PUMP) FAIL TO START (CCF)
			RCP----SEAL	RCP SEAL LOCA
3	2.0E-06	2.0	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			CHICF2PMBD001-ALL	CVS-MPP-001A,B (CHI PUMP) FAIL TO START (CCF)
			RCP----SEAL	RCP SEAL LOCA
4	1.1E-06	1.1	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWCF2PTSR001AD-ALL	EFS-MPP-001A,D (EFW PUMP) FAIL TO RUN DURING FIRST HOUR OF OPERATION (CCF)
			RCP----SEAL	RCP SEAL LOCA
5	1.1E-06	1.1	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			EFWPTAD001A	EFS-MPP-001A (A-EFW PUMP) FAIL TO START
			RCP----SEAL	RCP SEAL LOCA
6	1.1E-06	1.1	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			EFWPTAD001D	EFS-MPP-001D (D-EFW PUMP) FAIL TO START
			RCP----SEAL	RCP SEAL LOCA
7	9.6E-07	1.0	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			RCP----SEAL	RCP SEAL LOCA
			SGNBTSWCCF3	NON-SAFETY (PCMS) APPLICATION SOFTWARE CCF
8	9.0E-07	0.9	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EPSDLLRAACA-L2	A-AAC FAIL TO LOAD AND RUN AFTER FIRST HOUR OF OPERATION
			OPSLOOP	CONSEQUENTIAL LOOP GIVEN A REACTOR TRIP
			RCP----SEAL	RCP SEAL LOCA
9	8.7E-07	0.9	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			ACWOO02FS	(HE) FAIL TO ESTABLISH THE ALTERNATE CCWS BY FIRE PROTECTION WATER SUPPLY SYSTEM
			ACWTMPZ351A	VWS-MPP-351A (A-CONDENSER WATER PUMP) TEST & MAINTENANCE
			RCP----SEAL	RCP SEAL LOCA
10	8.1E-07	0.8	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			EFWTMTA001D	EFS-MPP-001D (D-EFW PUMP) TEST & MAINTENANCE
			RCP----SEAL	RCP SEAL LOCA

Table 3 - Significant Contributing Cutsets to Increased Core Damage Risk of Case 2

Rank	Cutsets Freq. (/RY)	Percent (%)	Cutsets	Basic Event Name
1	1.7E-05	53.6	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			OPSLOOP	CONSEQUENTIAL LOOP GIVEN A REACTOR TRIP
			RCP----SEAL	RCP SEAL LOCA
2	2.6E-06	8.1	!19LOOP	LOSS OF OFFSITE POWER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			HPIOO02FWBD	(HE) FAIL TO OPEN SAFETY DEPRESSURIZATION VALVE AND START SAFETY INJECTION PUMP
3	1.2E-06	3.8	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			RCP----SEAL	RCP SEAL LOCA
4	5.9E-07	1.9	!19LOOP	LOSS OF OFFSITE POWER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			PZRMVOD117A	RCS-MOV-117A FAIL TO OPEN
5	4.7E-07	1.5	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWPTAD001A	EFS-MPP-001A (A-EFW PUMP) FAIL TO START
			RCP----SEAL	RCP SEAL LOCA
6	4.1E-07	1.3	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			ACWOO02CT-DP2	(HE) FAIL TO ESTABLISH THE ALTERNATE CCWS BY NON-ESSENTIAL CHILLED WATER SYSTEM COOLING TOWER
			ACWOO02FS	(HE) FAIL TO ESTABLISH THE ALTERNATE CCWS BY FIRE PROTECTION WATER SUPPLY SYSTEM
			RCP----SEAL	RCP SEAL LOCA
7	3.2E-07	1.0	!03SLOCA	SMALL PIPE BREAK LOCA
			EPSDLLREGTGB	B-CLASS 1E GTG FAIL TO LOAD AND RUN AFTER FIRST HOUR OF OPERATION
			OPSLOOP	CONSEQUENTIAL LOOP GIVEN A REACTOR TRIP
8	2.9E-07	0.9	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			OPSLOOP	CONSEQUENTIAL LOOP GIVEN A REACTOR TRIP
9	2.4E-07	0.8	!19LOOP	LOSS OF OFFSITE POWER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			EPSCBFO52RAT-A	EPS 52/RATA (BREAKER) FAIL TO OPEN
10	2.4E-07	0.8	!19LOOP	LOSS OF OFFSITE POWER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			EPSCBFO52UAT-A	EPS 52/UATA (BREAKER) FAIL TO OPEN

Table 4 - Significant Contributing Cutsets to Increased Core Damage Risk of Case 3

Rank	Cutsets Freq. (/RY)	Percent (%)	Cutsets	Basic Event Name
1	1.7E-05	27.0	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			OPSLOOP	CONSEQUENTIAL LOOP GIVEN A REACTOR TRIP
			RCP----SEAL	RCP SEAL LOCA
2	8.2E-06	13.0	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			CHIOO01CHIB	(HE) FAIL TO START THE STANDBY CHARGING INJECTION PUMP B
			RCP----SEAL	RCP SEAL LOCA
3	6.7E-06	10.7	!03SLOCA	SMALL PIPE BREAK LOCA
			SWSPMBD001B	EWS-MPP-001B (B-ESW PUMP) FAIL TO START
4	5.7E-06	9.2	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			CHIPMBD001B-R	CVS-MPP-001B (B-CHI PUMP) FAIL TO RE-START
			RCP----SEAL	RCP SEAL LOCA
5	3.6E-06	5.7	!03SLOCA	SMALL PIPE BREAK LOCA
			SWSMVOD503B	EWS-MOV-503B FAIL TO OPEN
6	1.7E-06	2.7	!13TRANS	GENERAL TRANSIENT
			SGNBTHWCCF	DIGITAL I&C HARDWARE CCF
7	1.2E-06	1.9	!15LOCCW	LOSS OF COMPONENT COOLING WATER
			EFWOO01006AB	(HE) FAIL TO CHANGEOVER EFW PIT
			RCP----SEAL	RCP SEAL LOCA
8	9.7E-07	1.6	!03SLOCA	SMALL PIPE BREAK LOCA
			SGNPIFD4001B	SLS-B POWER I/F B (DIGITAL PART) FAILURE
9	9.3E-07	1.5	!02MLOCA	MEDIUM PIPE BREAK LOCA
			SWSPMBD001B	EWS-MPP-001B (B-ESW PUMP) FAIL TO START
10	6.5E-07	1.0	!16PLOCW	PARTIAL LOSS OF COMPONENT COOLING WATER
			CHICF2PMBD001-ALL	CVS-MPP-001A,B (CHI PUMP) FAIL TO START (CCF)
			RCP----SEAL	RCP SEAL LOCA

Table 5 - Sensitivity Study Results for SFCP

	US -APWR TS		Sensitivity case		
Equipment	SR	Test Interval	Test Interval	CDF	ΔCDF (1)
Reactor Trip Breaker	SR 3.3.1.4	62 days	4 months	1.03E-06	1.0E-08
Class 1E GTG	SR 3.8.1.2	31 days	62 days	1.40E-06	3.8E-07
			6 months	3.27E-06	2.3E-06
Class 1E Battery	SR 3.8.6.2 SR 3.8.6.3	31 days	2 months	1.02E-06	<1E-8

(1) Increase of risk from the base case CDF of 1.02E-06

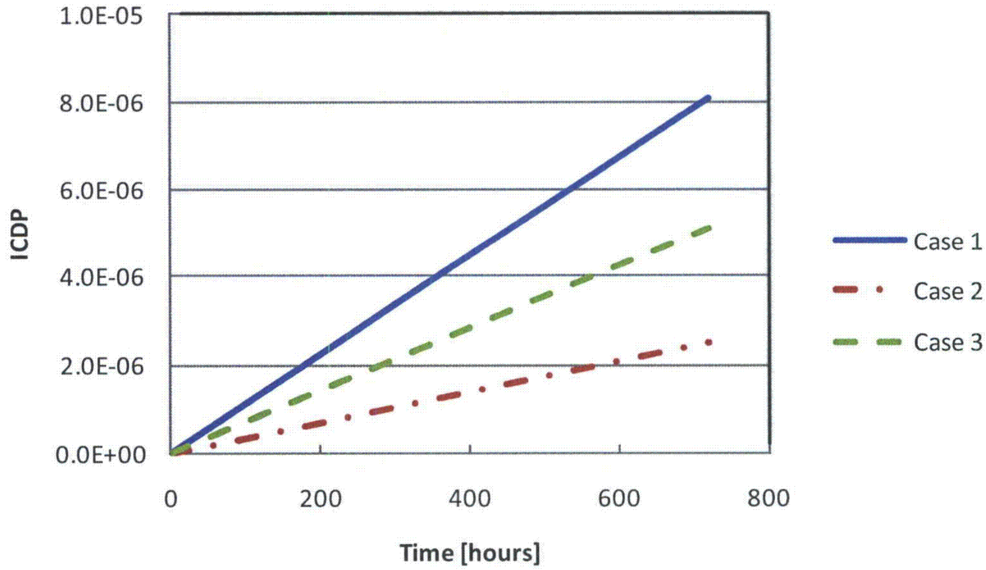

Figure 1 - Variation of ICDP with Time

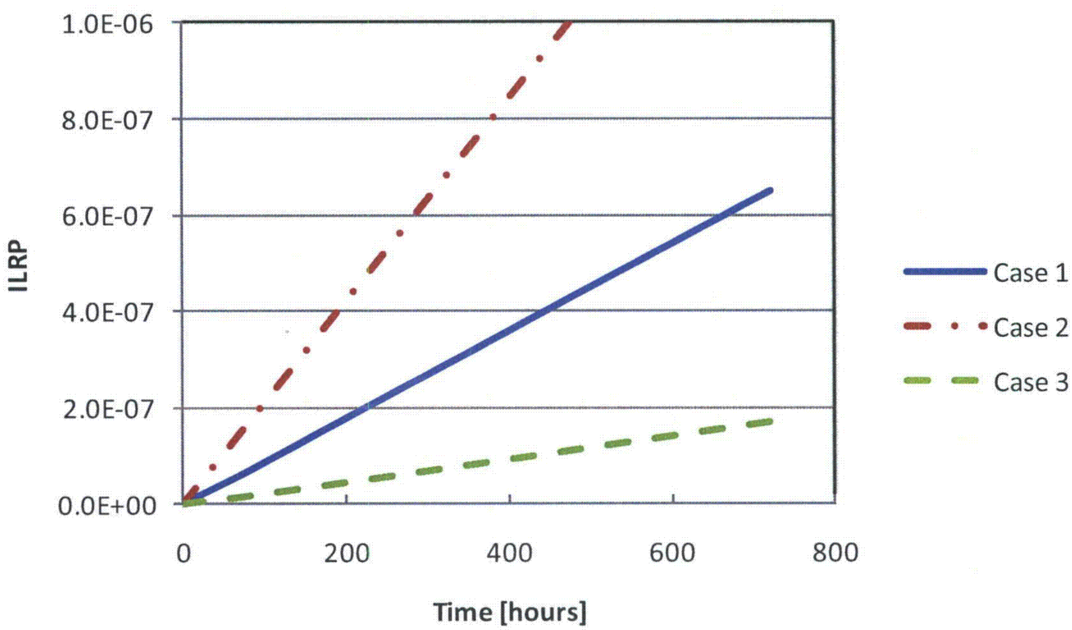

Figure 2 - Variation of ILRP with Time

Impact on R-COLA

None.

Impact on S-COLA

None.

Impact on DCD

None.