Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

September 18, 2002
LIC-02-0110

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

References: 1. Docket No. 50-285
2. Letter from NRC (T. J. Kenyon) to OPPD (R. T. Ridenoure) dated July 16, 2002 (NRC-02-105)

SUBJECT: Response to Request for Additional Information (RAI) Regarding Severe Accident Mitigation Alternatives in the Fort Calhoun Station, Unit 1 License Renewal Application

Attached to this letter are the Omaha Public Power District responses to the RAI questions contained in the Reference 2 letter. These responses concern the Severe Accident Mitigation Alternatives discussion in Appendix E of the Fort Calhoun Station, Unit 1 License Renewal Application.

No commitments are made to the NRC in this letter. I declare under penalty of perjury that the foregoing is true and correct. (Executed on September 18, 2002)

If you have any questions or require additional information, please contact T. C. Matthews at (402) 533-6938.

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

TCM/tcm

Attachment

A090

c: E. W. Merschoff, NRC Regional Administrator, Region IV
T. J. Kenyon, NRC Project Manager
W. F. Burton, NRC Project Manager
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska
Winston & Strawn

REQUEST FOR ADDITIONAL INFORMATION ON SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR THE FORT CALHOUN STATION

1. *The Severe Accident Mitigation Alternative (SAMA) analysis appears to be based on the current version of the "living" Probabilistic Risk Assessment (PRA) model for internal events, which is a modification to the original Individual Plant Examination (IPE) reviewed by the U. S. Nuclear Regulatory Commission (NRC) and the version of the PRA that was peer-reviewed in 1999. Please provide the following:*

    a. *The date and/or version of the PRA used for the SAMA analysis (which appears to be Revision 3 of the PRA), and a description of the internal and external peer review of the Level 1, 2 and 3 portions of this PRA.*

Response

The FCS SAMA evaluations utilized Revision 3 to the PRA dated November 2000.

The original peer review of the FCS IPE (Reference 1) was conducted in 1992 prior to the issuance of the IPE. Results of that peer review, which included the Level 1, 2, and 3 analyses, are included in that reference. The IPEEE was peer reviewed in December of 1993. The results of this peer review were also documented and are available for review at FCS.

In March 1999, the Fort Calhoun PRA was peer reviewed by a team of PRA engineers from Westinghouse, four other utilities and a PRA consultant. This peer review was the first conducted in accordance with the CE Owners Group implementation of the nuclear industry peer review process as documented in NEI 00-02. The following paragraphs briefly discuss the conclusions of that peer review.

The peer review team found the Fort Calhoun PRA to be effective for assessing planned plant maintenance and operations configurations and evaluating future plant design changes. The PRA was also found to be adequate for other applications which are supported by deterministic insights and plant expert panel input. The review did identify some areas of weakness in the PRA that should be considered in any application. The review also identified several areas of strength in the Fort Calhoun PRA.

The review team found the Fort Calhoun PRA to be strong in the areas of initiating event identification and containment performance analysis. OPPD had a particularly good treatment of the containment reliability analysis.

The reviewers recommended that the plant dependency analysis be upgraded. As the result of an in-depth investigation of dependencies, one missed dependency for the auxiliary feedwater pump, FW-54, was identified and corrected. Improvement in the documentation of the dependency matrix was also recommended. This activity was tracked by a configuration control program and was integrated in the Revision 3 PRA model used for SAMA assessment.

In all, there were a total of 89 specific review comments. Seven of these review comments/observations were felt to be significant. These items were identified for expedited resolution and were included in the plant's PRA configuration control program.

b. *A description of the major changes (plant and/or modeling changes) made to both the Level 1 and 2 PRA/IPE previously reviewed by the staff, and the version of the PRA peer-reviewed in 1999, and the respective impacts of these changes on Core Damage Frequency (CDF) and release frequency,*

Response

A summary of plant and modeling changes made over the past decade is presented in Table 1-1. In addition to the improvements listed in Table 1-1, the following PRA-based plant improvements that were in progress at the time of the IPE submittal have been completed (refer to Table 6-2 of the IPE (Reference 1)):

- Item 2, periodically leak testing shutdown cooling isolation valve HCV-347 to reduce the probability of an interfacing system LOCA
- Item 3, installed anti-galloping devices on the 161 KV transmission lines
- Item 4, installed a warning sign to leave a water-tight door open in the event of a flood

Table 1-1

Description	Since IPE	Since peer review	Plant change	Modeling change	CDF improvement	LERF improvement
Modifications of 345 and 161 KV switchyards, making them more robust	Yes	Yes	Yes	No	Yes	Yes
Potable water made available for makeup to Emergency Feedwater Storage Tank	Yes	No	Yes	Yes	Yes	No
Modified Condensate Storage Tank dump valve in response to peer review comment, improving availability of diesel-driven auxiliary feedwater pump	Yes	Yes	Yes	Yes	Yes	No
Raw water made available for makeup to Emergency Feedwater Storage Tank	Yes	No	Yes	Yes	Yes	No
Updated initiating event frequencies based upon CEOG standard	Yes	Yes	No	Yes	No	No
Improved HRA dependency analysis	Yes	Yes	No	Yes	No	No
Revised definitions of LERF and supporting LERF model	Yes	Yes	No	Yes	No	No
Created common cause basic event for ECCS sump strainers	Yes	Yes	No	Yes	No	No
Created basic event for common cause battery demand failure	Yes	Yes	No	Yes	No	No
Corrected logic between containment spray valves and number of operating containment spray pumps	Yes	Yes	No	Yes	No	No
Reversed Component Cooling Water containment isolation valve, to help mitigate interfacing system LOCA; verified by actual test that the valve could withstand the differential pressure	Yes	No	Yes	Yes	No	Yes

Description	Since IPE	Since peer review	Plant change	Modeling change	CDF improvement	LERF improvement
Improved reactor coolant pump seals	Yes	Yes	Yes	Yes	Yes	No
Modified roof hatch to allow makeup of Emergency Feedwater Storage Tank following turbine building fire	Yes	Yes	Yes	Yes	Yes	No
Installed colored tape to establish seismically safe storage areas for transient equipment	Yes	Yes	Yes	No	No	No
Procured portable fan for emergency cooling of Control Room	Yes	Yes	Yes	No	No	No
Installed protective trip override switch for diesel-driven auxiliary feedwater pump	Yes	Yes	Yes	Yes	Yes	No
Revised auxiliary feedwater surveillance tests to minimize number of unavailable SSCs	Yes	Yes	Yes	Yes	Yes	No
Revised ECCS recirculation test to minimize number and duration of unavailable SSCs	Yes	Yes	Yes	Yes	Yes	No
Removed some toxic gas monitors based partially on risk-informed justification	Yes	Yes	Yes	No	No	No
Revised model to account for possible loss of air to ECCS recirculation level switched and SI recirculation flow path	Yes	Yes	No	Yes	No	No
Procured portable pumps for feeding steam generators in the event of a catastrophic flood	Yes	Yes	Yes	Yes	Yes	No
Improved risk assessment and risk management processes to support 10 CFR 50.65(a)(4)	Yes	Yes	No	No	Yes	Yes

c. *A breakdown of the CDF by initiating event (specifically, Loss of Offsite Power (LOOP), General Transients, Station Blackout, anticipated transients without scram (ATWS), Loss-of-Coolant Accidents (LOCAs), Interfacing System LOCA (ISLOCA), and Steam Generator Tube Rupture (SGTR), and other initiators), and the contribution to CDF from external flooding and seismic initiators that are represented in the PRA model.*

Response

The distribution of the FCS core damage frequency among initiating events is presented in Table 1-2:

Table 1-2
Summary of key Core Damage Initiating Events

Initiating Event	CDF (per year)	Contribution to Core Damage (%) [Internal Events]
Loss of Off-Site Power (not resulting in SBO)	3.78E-06	15.67
Station Blackout (SBO)	4.18E-06	17.33
General Transients	3.02E-06	12.5
ATWS	Negligible	Negligible
LOCAs	6.35E-06	26.3
ISLOCA	9.56E-07	4.0
SGTR	2.30E-06	9.5
Internal Flooding	1.28E-06	5.3
Other	2.26E-06	9.4

The CDF associated with external flooding (which is not included in the model) is 3.7E-06 per year. The CDF associated with seismic events (which is in the model) is 1.14E-06 per year, which corresponds to 4.30% of the CDF.

d. *A short description defining all the Plant Damage States (PDSs), and the accident sequences that dominate the PDSs.*

Response

The FCS Level 2 model is based on the structure defined in the IPE (Reference 1). Plant damage states were defined using parameters summarized in the following Table 1-3.

Table 1-3 Parameters used for Defining PDSs	
Parameter	**Description**
RCS Pressure	RCS pressure at the time of core melt
RCS Leakage Rate	RCS Inventory Loss Rate at the time of core melt
Core Melt Timing	Time of onset of core melt as measured from event initiation
Containment Spray Status	Containment spray status at time of core melt
Containment Heat Removal Status	Containment heat removal status at time of core melt
Reactor Cavity Status	Level of water accumulation in the Reactor Cavity at the time of core melt
Containment Isolation Status	Containment isolation status at time of core melt

Using the above definitions 520 potential plant damage states (PDSs) were defined. Of those, 12 dominant PDSs were identified as having individual contributions of >1%. In total, these PDSs contribute a little more than 50% of the plant CDF. These PDSs and their percentage contribution to the internal events CDF are presented in Table 1-4. Additional details on the plant damage state definition may be found in Table 4.3.1.3 of the IPE (Reference 1).

Table 1-4 Summary of Dominant Plant Damage States*	
Plant Damage State (PDS)	**Typical plant accident sequence**
31, 47	Transient initiated station blackout or loss of feedwater event with containment cooling unavailable
479	Containment bypassed via failure of RCP seal cooler failure
439	Transient induced LOCA with failure of RCS inventory control. Events contributing to the damage state include induced RCP seal failures and inadvertent opening of pressurizer PORVs or safeties. Containment heat removal is available.
455	Small LOCA without HPSI ; containment cooling available via fan coolers.
184	SGTR with failure of long term inventory control

Table 1-4 Summary of Dominant Plant Damage States*	
Plant Damage State (PDS)	**Typical plant accident sequence**
425	Small LOCA without HPSI and containment cooling available via sprays
193	Transient induced RCP seal LOCA with containment heat removal unavailable
462	Small LOCA with loss of long term cooling and containment cooling available
235	Containment bypassed via failure of SDC suction line.
429	Transient induced LOCA with failure of RCS inventory control. Containment heat removal is available via fan coolers
39	Transient initiated station blackout with containment heat removal unavailable

*For additional information see Table 4.3.1.3 of the FCS IPE (Reference 1).

e. *An assessment of the uncertainties associated with the calculated CDF and risk (e.g., the mean and median CDF estimates and the 5^{th} and 95^{th} percentile values of the uncertainty distribution), and the impact on SAMA identification and screening results if risk reduction estimates were based on the upper end of the distribution rather than the mean value (Section 5 5 only provides a portion of this information), and*

Response

Based on the analysis performed, the parameters of the estimated core damage frequency (per year) distribution attributed to internal and external initiated events (excluding external flood and fire) are shown in Table 1-5 below.

Table 1-5

Mean:	2.52E-05
5^{th} Percentile:	1.22E-05
50^{th} Percentile (Median):	1.97E-05
95^{th} Percentile:	4.68E-05
Error Factor:	1.96

The core damage frequency at FCS is assumed to be log normally distributed. The above distribution incorporates data uncertainty for equipment and human errors included in the PSA model. Use of the bounding 95th percentile risk would have minimal impact on the screening and SAMA identification as that step considered the risk ranking of equipment importance and identification of dominant cutsets. Therefore, the precise value of the risk impact benefit was not critical. The risk benefit did have an impact in selecting which of the potential SAMAs considered in the analysis would be cost beneficial. The impact of uncertainty on the selection of SAMAs is discussed in the response to RAI 2.

f. A breakdown of the population dose (person-rem per year) by containment release mode in the following form:

Containment Release Mode	*Fraction of Population Dose*
SGTR (Late and Early)	
Interfacing Systems LOCAs	
Early containment failure	
No vessel breach, no containment failure	
Late containment failure	
No containment failure (vessel breach)	

In addition, please provide, separately, the contribution of hydrogen and CO combustion to early and late containment failure probability, and the fraction of population dose attributable to external flooding and seismic initiators.

Response

The population dose distribution is presented in Table 1-6, below.

Table 1-6: Population Dose for Various Containment Release Modes

Containment Release Mode	Fraction of Population Dose (internal events)	Fraction of Population Dose (internal and external events)
SGTR (Late and Early)	4.64E-01	3.90E-01
Interfacing Systems LOCAs	2.48E-01	2.09E-01
Early containment failure	1.56E-01	2.84E-01

Table 1-6: Population Dose for Various Containment Release Modes		
Containment Release Mode	**Fraction of Population Dose (internal events)**	**Fraction of Population Dose (internal and external events)**
No vessel breach, no containment failure	1.32E-02	1.11E-02
Late containment failure	1.14E-01	1.02E-01
No containment failure (vessel breach)	4.64E-03	3.90E-03

For the case with external and internal events combined (column 3 above), the fraction of the dose that is attributed to seismic events and external floods is 0.158 and 0.04, respectively.

The FCS design is robust to combustion challenges (See Reference 1). A review of the FCS Level 2 PRA indicates that the high design pressure (60 psig) and high ultimate strength (>190 psig) associated with the FCS containment results in a very low (<1%) contribution of containment failure directly caused by combustion of hydrogen. The impact of CO combustion on the FCS containment is negligible. Concurrent hydrogen combustion with a high pressure RV failure and a high pressure melt ejection event contributes to the early containment failure fraction presented above.

Reference for Response 1:

Letter from W.G. Gates (OPPD) to Document Control Desk (NRC). "NRC Generic Letter 88-20 Submittal for Fort Calhoun Station 'Individual Plant Examination for Severe Accident Vulnerabilities.'" December 1, 1993.

2. *The discussions on page 5-5 of the application state that the CDF for fire events based on the Fire Induced Vulnerability Evaluation (FIVE) methodology is 2.78 x 10^{-5} per year, and the 95th percentile CDF value for internal events (plus some dominant external event contributors) is about 4.68x10^{-5} per year. The last paragraph on page 5-5 states that, based on the ratio of the sum of these frequencies to the baseline CDF, the estimated uncertainty factor for application to SAMA assessment should be approximately 3 (i.e. 7.46 x 10^{-5}/ 2.45 x 10^{-5} ≅ 3). On the other hand, on page 4-38, it is stated, "In general, if the expected cost exceeded twice the calculated benefit, the SAMA was considered not to be cost beneficial." The information provided on page 5-5 does not appear to support the decision to use a factor of two multiplier to account for various uncertainties, including external events. Please justify the use of a factor of two multiplier in the SAMA evaluation process, in view of the uncertainty assessment that indicates a factor of three may be more appropriate. Would any*

of your conclusions for individual SAMAs change if a factor of three were considered, compared to a factor of two?

Response

In estimating a bounding uncertainty, the FIVE methodology was used to establish upper bound fire risks. In generating the FIVE equivalent CDF, it was assumed that once the fire has initiated, the functional capability of all equipment in the room is lost (regardless of location with respect to the event) and that recovery actions are unsuccessful. Thus, the FIVE fire risk estimate provides a significant overestimate of the fire CDF. Consequently, combining the FIVE CDF with the 95^{th} percentile internal events CDF will significantly overstate the 95^{th} percentile CDF, compared with the realistic combined CDF had a detailed fire PRA been performed. While this conservative combination resulted in a 95^{th} percentile CDF to mean internal events CDF ratio of 3, a lower multiplier of 2 was selected for the cost benefit assessment. This ratio results in delta CDF predictions that exceed the 95^{th} percentile CDF for internal events. To ensure a conservative cost benefit estimate, the SAMA was evaluated using bounding impacts of the beneficial impact of the change (e.g., assuming availability of batteries that don't deplete). Finally, for many of the SAMAs evaluated, the contribution of the modification to a change in the fire CDF was negligible.

The use of a factor of 3 would increase the expected benefit by 50%. Based on a review of Table 4.16 of the environmental report, this could impact the following SAMAs by increasing the benefit to a level where the estimated cost could approach or exceed implementation of certain SAMAs. The SAMAs potentially affected are: SAMA 4, SAMA 10, SAMA 54, SAMA 56, SAMA 185, SAMA 187, SAMA 188 and SAMA 190. The results of this reassessment are presented in Table 2-1. Of these, SAMA 4 is already in the list for implementation, SAMA 10, SAMA 56, and SAMA 188 remain with clearly negative net values, and SAMA 54 was dropped as a result of the cumulative impact of the implementation of the seven SAMAs. SAMAs 185, 187 and 190 would produce positive net values when a factor of 3 is used as a multiplier. However, SAMAs 183 and 190 are not expected to have an impact on the fire risk, and SAMA 187 is expected to have a minimal impact on the fire risk; therefore, the use of a factor of 3 would not be applicable. Thus, the use of a factor of 2 for that application is acceptable.

Table 2-1
Re-Assessment of Selected SAMAs

SAMA	Description	Internal + Seismic	Internal + Seismic + Fire	Cost to Implement ($K)	Comment
		2x Benefit ($K)	3x Benefit ($K)		
4	Improved procedure for Loss of RCP seal cooling	54	81	30	Net positive value. Implementation included in final SAMA list

Table 2-1
Re-Assessment of Selected SAMAs

SAMA	Description	Internal + Seismic	Internal + Seismic + Fire	Cost to Implement ($K)	Comment
		2x Benefit ($K)	3x Benefit ($K)		
10	Install N-9000 Seals	54	81	400	Negative net value
54	Incorporate alternate battery charging capability	222	333	>150	Net positive value. Implementation of other SAMA modifications identified addresses this concern and significantly reduces benefit
56	Improve DC bus load shedding	222	333	>200	Detailed cost estimate not provided The likelihood of successfully implementing and managing a battery load to 24 hour is very small SAMA is not recommended No change
185	Remove SI-2C from auto-start	88	132	90	Net negative value when uncertainty from internal events and seismic considered. Change will not alter fire results therefore factor of 3 need not apply No change.
187	Enhance operation of FW-54	28	42	>40	Net negative alue when considering only internal events and seismic. Change will have minimal impact on fire results; therefore, factor of 3 need not apply. No change.
188	Enhance External Flood Procedures	32	48	70	Net negative value. No change.

Table 2-1 Re-Assessment of Selected SAMAs					
SAMA	Description	Internal + Seismic	Internal + Seismic + Fire	Cost to Implement ($K)	Comment
		2x Benefit ($K)	3x Benefit ($K)		
190	Enhance EOPs to avert potential TI-SGTR	34	51	>30	Net positive value. Impact is partially addressed in SAMA 183. Owner's group task to address this issue generically for CE plants is in progress OPPD will follow OG efforts and review when procedure is defined. Change will not alter fire results; therefore, factor of 3 need not apply.

3. *On page 4-36, it is stated, "...the top 100 cutsets of the Level 1 PRA update were examined to identify the important contributors to plant risk...." Please provide the following:*

   a *More specifics on how CDF importance measures were used in support of the SAMA identification and screening process. (OPPD states, on page 4-37, that risk measures were used for "preliminary screening," but it is not clear if and how they were used in the rest of the SAMA identification process.)*

<u>Response</u>

The "screening" process used in the SAMA assessment was intended to identify important risk contributors. Components identified as risk significant in that process were given special attention. However, in evaluating the SAMA fixes all SAMAs that have been considered by LRA applicants to date were considered candidate changes regardless of the specific components involved.

In order to identify potentially risk significant plant changes, the plant components significant to risk were identified via use of Risk Achievement Worth (RAW) and Fussel-Vesely Importance (F-V) measures. Lists of components and actions with high RRW values (>1.1) or F-V values >0.005 were assembled and reviewed to establish potential means of improving the component's or action's reliability, or utilizing alternate systems/components to meet the intent of the component. These risk measures were supplemented by a review of the top 100 cutsets. Level 2 and 3 results were considered in that changes intended to reduce the frequency or consequences of containment bypass (e.g., SGTR or ISLOCA) scenarios were given special attention. As a result of the high containment pressure capability of the

FCS containment design, the plant is robust to early containment failures associated with hydrogen combustion and direct containment heating (See References 1 and 2).

b *A list of key equipment failures and human actions that dominate CDF and the large early and late release frequencies, which have the greatest potential for reducing the risk of severe accidents at FCS, along with the results of any supporting importance analyses (e g., Fussel-Vesely and/or risk reduction importance measures).*

Response

Tables 3-1 and 3-2 identify the plant equipment failure modes and operator actions that are important to CDF and LERF, respectively . This table contains equipment and operator actions with (F-V) measures associated with the dominant plant cutsets.

Table 3-1
Equipment/Operator Actions Important to CDF

Event Failure Mode Description	F-V	Component or Action
Battery #1 Depleted In 8 Hrs	1.88E-01	Battery
Battery #2 Depleted In 8 Hrs	1.87E-01	Battery
IA Accumulators Depleted	1.63E-01	Instrument Air
Failure to Recover From Premature RAS Signal	1.55E-01	RAS
Diesel Generator D1 Fails to Run – EPS	1.18E-01	EDG
Common Cause Failure of DG-D1 & DG-D2 to Run	1 16E-01	EDG
Diesel Generator D2 Fails to Run - EPS	1 14E-01	EDG
RCP Seal Leakage Given Loss of Cooling	8 48E-02	RCP Seal
Independent Failure of the 2500# HPSI Header	5 76E-02	HPSI
Independent Failure of the 1500# HPSI Header	5 72E-02	HPSI
Operator Fails to Line Up Raw Water Backup Flow	5 10E-02	Operator Action
Operator Fails to Minimize DC Loads on Battery #1 and #2	4 91E-02	Operator Action
Battery #2 Depleted in 2.6 Hrs	4 64E-02	Battery
Battery #1 Depleted In 2 6 Hrs	4.64E-02	Battery

Table 3-2 Equipment/Operator Actions Important to LERF		
Description	**F-V**	**Component or Action**
IA Accumulators Depleted	4 23E-01	Instrument Air
Failure to Recover from Premature RAS Signal	4.20E-01	Operator action
CCF of IA After-filters CA-11A & CA-11B to Operate	1.07E-01	Instrument Air
CCF of IA Pre-filters CA-13A & CA-13B to Operate	1.07E-01	Instrument Air
Operator Fails to Isolate Leak by Closing HCV-438C or 438D from Control Room	8 89E-02	HCV-348
Independent Failures of the Receiver Relief Valves	8.89E-02	Instrument Air
CCF of the Three Compressors to Run	7 42E-02	Instrument Air
Operator Successfully Minimizes DC Loads on Battery #1 and #2	6 20E-02	Operator Action
MOV HCV-348 Fails Open at T=0	6 16E-02	HCV-348
12" RHR SDC Piping Outside Containment Ruptures (Sch 40 304SS)	6 14E-02	RHR Piping
Battery #2 Depleted in 8 Hours	6.10E-02	Battery
Battery #1 Depleted in 8 Hours	6.05E-02	Battery
MOV HCV-347 Internal Leakage (Assume Annual Testing)	6.02E-02	HCV-347
Latent or Demand Failure of Battery #1	5.49E-02	Battery
CCF of Comp. Inlet Air Filters to Operate	5 35E-02	Instrument Air
Latent or Demand Failure of Battery #2	5 16E-02	Battery
MOV HCV-348 Internal Leakage (Initiator)	4 94E-02	HCV-348
Failure to Prevent Core Melt Following Loss of All Power After 8 Hours	4 36E-02	Operator Action
Independent Failures of 125VDC Panel AI-41A	4.11E-02	125VDC

Late releases were typically caused by overpressurization of the containment or basemat melt-through. To overpressurize containment the containment fan coolers and containment spray system must be unavailable. This scenario is dominated by a loss of power and failure of EDGs.

c. *The percentage contribution to the CDF of the top 100 cut sets.*

<u>Response</u>

The top 100 cutsets constitute 64% of the plant CDF.

References for Response 3

1. Letter from W.G. Gates (OPPD) to Document Control Desk (NRC). "NRC Generic Letter 88-20 Submittal for Fort Calhoun Station 'Individual Plant Examination for Severe Accident Vulnerabilities.'" December 1, 1993.
2. NUREG/CR-6475,"Resolution of the Direct Containment Heating Issue for Combustion Engineering and Babcock and Wilcox Plants," M. Pilch, et. al., November, 1998

4. *In Section 5.2.1.2, Source Terms, OPPD states that the source terms were obtained from the latest Level 2 FCS PRA model analysis. Please provide more detailed information (e.g., a tabular list) on the release categories, including the definition, fractional releases, frequency, containment matrix (relationship between PDSs and release categories), and the associated conditional consequences, used in the SAMA analyses.*

Response

The FCS Level 2 model is based on the structure defined in Reference 1 (IPE). Plant damage states were defined as summarized in Table 4-1.

Table 4-1
Plant Damage State Parameters

Parameter	Description
RCS Pressure	RCS pressure at the time of core melt
RCS Leakage Rate	RCS Inventory Loss Rate at the time of core melt
Core Melt Timing	Time of onset of core melt as measured from event initiation
Containment Spray Status	Containment spray status at time of core melt
Containment Heat Removal Status	Containment heat removal status at time of core melt
Reactor Cavity Status	Level of water accumulation in the Reactor Cavity at the time of core melt
Containment Isolation Status	Containment isolation status at time of core melt

Using these definitions. 510 plant damage states are defined. Of the 510 PDSs, only ten states have CDF contributions greater than 1%. Furthermore, the sum of these PDSs encompasses more than 50% of the plant CDF. Refer to the details on the PDSs found in Table 4.3.1.3 of the IPE (Reference 1) and to the details on release fractions by release category found in Table 4.8.2.6 of the FCS IPE. These tables were used to assemble this presented information.

The PDSs are propagated into release classes. The dominant release class definitions are summarized in Table 4-2.

Table 4-2
Description of Dominant Release Categories

Release Class Category	Description
1.1,1.3	Intact containment with spray scrubbing of releases
1.2,1.4	Intact containment without spray scrubbing of releases
3.1	Late containment failure with spray scrubbing of releases
3.10	Late containment failure without spray scrubbing of releases
4.1	Early containment failure with spray scrubbing of releases available
4.4	Early containment failure without spray scrubbing of releases available
6.11	SGTR with cycling Safety Relief Valve
6.20	SGTR with a stuck open Safety Relief Valve
7.1	Large Inter-system LOCA via SDC suction line

A summary of the associated Revision 3 release classes is presented in Table 4-3. Risk significant releases are typically limited to events with either loss of containment isolation or containment bypass. The potential for increased likelihood of thermally induced SGTR (TI-SGTR) was modeled by assuming 10% of the events that result in high pressure core damage sequences will proceed to a SGTR in the presence of a stuck open secondary safety valve.

Table 4-3
Revision 3 PSA Release Class Map

Initiating Event	%CDF	R3 CDF	SO MSSV	INTACT	EARLY HPME	LOSS OF ISOLATION	Late	SGTR Cycling	Bypass	Total IE CDF
			(RC 6.20)	RC 1	RC 4	RC 6.1	RC 3	RC 6.11	RC 7	
SGTR	9.506	2.300E-06	2.300E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.070E-06	0.00E+00	2.300E-06
Loss of 125 VDC	2.583	6.250E-07	0.000E+00	3.933E-07	9.588E-09	3.709E-08	1.839E-07	1.065E-09	0.00E+00	6.250E-07
Loss of DC	1.463	3.540E-07	0.000E+00	2.228E-07	5.431E-09	2.101E-08	1.042E-07	6.034E-10	0.00E+00	3.540E-07
LOOP plant	20.954	5.070E-06	0.000E+00	3.191E-06	7.778E-08	3.009E-07	1.492E-06	8.642E-09	0.00E+00	5.070E-06
LOOP grid	5.951	1.440E-06	0.000E+00	9.062E-07	2.209E-08	8.545E-08	4.238E-07	2.455E-09	0.00E+00	1.440E-06
LOOP-Weather	5.993	1.450E-06	0.000E+00	9.125E-07	2.224E-08	8.605E-08	4.267E-07	2.472E-09	0.00E+00	1.450E-06
Seismic	4.712	1.140E-06	0.000E+00	7.174E-07	1.749E-08	6.765E-08	3.355E-07	1.943E-09	0.00E+00	1.140E-06
other	4.629	1.120E-06	0.000E+00	7.048E-07	1.718E-08	6.646E-08	3.296E-07	1.909E-09	0.00E+00	1.120E-06
Flood-internal	5.290	1.280E-06	0.000E+00	8.055E-07	1.964E-08	7.596E-08	3.767E-07	2.182E-09	0.00E+00	1.280E-06
LOCCW	2.608	6.310E-07	0.000E+00	3.971E-07	9.680E-09	3.745E-08	1.857E-07	1.076E-09	0.00E+00	6.310E-07
RT/TT/PLR*	6.104	1.477E-06	0.000E+00	9.295E-07	2.266E-08	8.765E-08	4.347E-07	2.518E-09	0.00E+00	1.477E-06
ISLOCA	3.951	9.560E-07	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	9.560E-07	9.560E-07
LOCA	26.256	6.353E-06	0.000E+00	3.998E-06	1.083E-07	3.770E-07	1.870E-06	0.000E+00	0.00E+00	6.353E-06
TOTAL	100.000	2.420E-05	2.30E-07	1.32E-05	3.32E-07	1.24E-06	6.16E-06	2.09E-06	9.56E-07	2.42E-05

<u>Reference for Response 4:</u>

Letter from W.G. Gates (OPPD) to Document Control Desk (NRC). "NRC Generic Letter 88-20 Submittal for Fort Calhoun Station 'Individual Plant Examination for Severe Accident Vulnerabilities.'" December 1, 1993.

5 *Provide the following information concerning the application of meteorological data to the MACCS2 code:*

a. *Describe the meteorological sampling approach and specifications used in the MACCS2 calculations for FCS, as described in Sections 5.13 through 5.17 of the code manual for MACCS2 (NUREG/CR-6613, Volume 1).*

<u>Response</u>

Meteorological conditions for MACCS2 were established using the METCOD=2 (Meteorological Bin Sampling) with a total of 36 sampling bins and with the variable NSMPLS set to 4. Boundary weather data was set to typical values to give a more realistic consequence assessment. Boundary weather data could have been adjusted to "rain out" the remaining source term within the 50-mile zone. This would only have an effect on the long-term pathways (ingestion, groundshine), which are the most uncertain of the consequences analyzed.

b. *In Section 5 2.1.1 of the application, it states that 131 radionuclides are used to represent the core inventory. However, in Section 5.2.1.7, it states that the dose conversion factors (DCFs) for early fatalities and injuries were taken from DOE/EH-0070, and are limited to the original MACCS set of 60 radionuclides. Clarify the number of radionuclide groups used in the MACCS2 calculations. If 60 groups were used, explain how the additional 71 groups were accounted for in the analysis. Also, please describe and justify the assumed deposition velocity in the MACCS2 analysis (indicated to be 0.03 meters per second in Section 5.2.1 11), and discuss the sensitivity of consequence results to changes in this parameter.*

Response

The calculation of early fatality effects requires a dose conversion factor library with acute effect parameters. As noted in Section 5.2.1.7 of the application, the DOSED825 set of dose conversion factors does not contain acute effect parameters. The DOSD825 dose conversion factors are equivalent to the FGR 11 and FGR 12 reports (Federal Guidance Reports) which include dose conversion factors for latent effects only. Therefore, the DOSDATA dose conversion factor library was used for the acute effect analyses. The DOSDATA dose conversion factor library is restricted to the reduced number of nuclides, whereas, the DOSD825 library allows a much larger number of nuclides. The set of 60 nuclides could have been used for the latent effect analysis; however, it was conservatively determined that the larger set of nuclides would give a more realistic magnitude of the consequences for latent effects.

Several reference documents, including NUREG-1150, MACCS manuals, and an NRC MACCS parameter assessment, were reviewed. The deposition velocity of 0.03 m/s is within the range of typical values for the Fort Calhoun location. Increasing the deposition velocity depletes the plume of material which decreases the direct short-term doses (e.g. cloudshine, inhalation) and increases the long-term doses (e.g. ingestion, groundshine). However, the direct doses usually overwhelm the long-term doses. For example, if the deposition velocity is set to 0 (no deposition), the total dose would be expected to be higher than if deposition is included. In fact, when bounding (conservative) analyses are performed, deposition is set to 0 and the long-term effects are ignored.

c. *Describe the differences, if any, in the meteorological sampling used with the 1988 data and the data for the 1994-1998 period.*

Response

The 1988 meteorological data was the only year which included precipitation, so it was run for the consequence analyses. An investigation was performed for all six years (without rainfall). The comparison of doses for sample MACCS cases run with the various meteorological years indicated that 1988 produced somewhat more conservative results. The source of the 1988 data was the National Oceanic and Atmospheric Administration (NOAA).

The mixing height data was taken from Holzworth's compilation of isoplethed maps of the contiguous United States showing morning and afternoon mixing heights for each of the four seasons. The source of the 1994 through 1998 data was from the OPPD weather tower data.

d. *Confirm that the release was limited to particles of a single size, and non-depositing gases. If some or all gases were assumed to deposit, describe and justify the choice of deposition parameters for various classes of gases.*

Response

The release was assumed to be limited to particles of a single size and non-depositing gases.

e. *Confirm that precipitation was included in the meteorological data set. If it was included, was wet deposition modeled? If wet deposition was modeled, describe and justify the wet deposition model parameters used (Section 5.5 of NUREG/CR-6613, Vol. 1). If not, explain why it was not modeled.*

Response

Precipitation was included as noted in the meteorological data input files. Wet deposition was included and used the Jon Helton 1986 coefficients

6. *Discuss the implications of any extended power uprate under consideration on the SAMA study, in particular, on the SAMA selection process and benefit (averted risk) determination.*

Response

Although OPPD is considering extended power uprate as an option, no final decision has been made. From a risk assessment perspective, it appears that the two major impacts to plant risk from a power uprate are an increase in the initial radiological source term and an increase in decay heat affecting available operator response times. The actual impact on the plant would depend upon the extent of the power uprate and the supporting plant design changes. OPPD expects to include PRA considerations in any power uprate design process.

7. *OPPD notes, "... SAMAs that affect structures, systems, and components that may enhance mitigation functions during both at-power and shutdown conditions are addressed." (Page 4-36) Please identify and discuss any SAMAs that might significantly enhance mitigation functions during shutdown. Are any in the group evaluated in Section 5.4?*

Response

As stated in the application, shutdown conditions were not explicitly addressed. However, several "at power" improvements may provide benefits during shutdown conditions. These improvements include those SAMAs that would either add an alternate generator to the FCS site or utilize existing generators in alternate capacities (SAMAs 182, 183 and 184). Other SAMAs potentially providing benefits during shutdown include SAMA 41 (develop a procedure for the fire water system to be used as a backup source for the containment spray system) and SAMA 187 (extend hot standby operation using diesel-driven Auxiliary Feedwater Pump FW-54).

8. *Based on a review of the SAMAs considered by OPPD, the staff requests the following additional information regarding specific SAMAs. Also, the source references provided in Table 5.3-1 for the various SAMAs do not appear to be consistently indexed with the references in Section 5 5. Please provide corrected source references for Table 5.3-1.*

Response

OPPD has reviewed the reference sources provided in Table 5.3-1. The reference list supporting Section 5.3 has been revised to correlate with the reference sources listed in Table 5.3-1 (see list below). The original Reference 1 was inadvertently omitted, and with that correction the reference list was renumbered accordingly. In addition, two line items (SAMAs 34 and 116) required corrections to the reference sources. The corrected references sources are provided below.

SAMA No.	Potential Enhancement	Reference Source
34	Create a molten core debris containment system with heat removal capabilities under the basemat or other enhancements to prevent melt-through, such as thicker basemat.	3, 4, 6, 8, 11, 16, 17, 20
116	Provide capability for diesel-driven, low-pressure vessel makeup.	4, 5, 13

As a result of renumbering the references, the reference callout for the CCNPP cost estimates given in the disposition for SAMAs 30, 51, 64, 67, 68, 77, 80, 102, 117, 118, and 121 also needs to be revised from 5.3-25 to 5.3-26.

Ref. 5.3-1 Letter from Mr. R. E. Denton (BGE) to Document Control Desk (NRC). "Summary Report of Individual Plant Examination Results (Generic Letter 88-20) (TAC Nos. M74392 and M74393)." December 30, 1993.

Ref. 5.3-2 Letter from Mr. M. O. Medford (TVA) to Document Control Desk (NRC). "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Generic Letter (GL) 88-20 - Individual Plant Examination (IPE) for Severe Accident Vulnerabilities - Response (TAC M74488)." September 1, 1992.

Ref. 5.3-3 *Cost Estimate for Severe Accident Mitigation Design Alternatives, Limerick Generating Station for Philadelphia Electric Company.* Bechtel Power Corporation. June 22, 1989.

Ref. 5.3-4 U.S. Nuclear Regulatory Commission. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants.* Volume 1, Table 5.35, "Listing of SAMDAs considered for the Limerick Generating Station." NUREG-1437. Office of Nuclear Regulatory Research. Washington, D.C., May 1996.

Ref. 5.3-5 U.S. Nuclear Regulatory Commission. *Generic Environmental Impact Statement for License Renewal of Nuclear Plants.* Volume 1, Table 5.36. "Listing of SAMDAs considered for the Comanche Peak Steam Electric Station." NUREG-1437. Office of Nuclear Regulatory Research. Washington, D.C., May 1996.

Ref. 5.3-6 Letter from Mr. W. J. Museler (TVA) to Document Control Desk (NRC). "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Severe Accident Mitigation Design Alternatives (SAMDA) - (TAC Nos. M77222 and M77223)." June 5, 1993.

Ref. 5.3-7 Letter from Mr. D. E. Nunn (TVA) to Document Control Desk (NRC). "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Severe Accident Mitigation Design Alternatives (SAMDA) - Response to Request for Additional Information (RAI) - (TAC Nos M77222 and M77223)." October 7, 1994.

Ref. 5.3-8 Letter from N. J. Liparulo (Westinghouse Electric Corporation) to Document Control Desk (NRC). "Submittal of Material Pertinent to the AP600 Design Certification Review." December 15, 1992.

Ref. 5.3-9 Brookhaven National Laboratory, Department of Advanced Technology, Technical Report FIN W-6449. *NRC - IPE Workshop Summary/ Held in Austin, Texas; April 7-9, 1997.*" Appendix F - Industry Presentation Material, Contribution by Swedish Nuclear Power Inspectorate (SKI) and Safety Assessment Consulting (SAC). "Insights from PSAs for European Nuclear Power Plants," presented by Wolfgang Werner, SAC. July 17, 1997.

Ref. 5.3-10 Brookhaven National Laboratory, Department of Advanced Technology, Technical Report FIN W-6449. *NRC - IPE Workshop Summary/ Held in Austin, Texas, April 7-9, 1997.* Appendix D - NRC Presentation Material on Draft NUREG-1560. July 17, 1997.

Ref. 5.3-11 U.S. Nuclear Regulatory Commission. *Final Environmental Statement related to the operation of Watts Bar Nuclear Plant, Units 1 and 2*." NUREG-0498, Supplement No. 1. Associate Director for Advanced Reactors & License Renewal. Washington, D.C., April 1995.

Ref. 5.3-12 U.S. Nuclear Regulatory Commission. *PWR Dry Containment Issue Characterization*. NUREG/CR-5567. (BNL-NUREG-52234). Brookhaven National Laboratory. Upton, New York, August 1990.

Ref. 5.3-13 U.S. Nuclear Regulatory Commission. *Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance*. NUREG-1560, Volume 2. Division of Systems Technology. Washington, D.C., December 1997.

Ref. 5.3-14 U.S. Nuclear Regulatory Commission. *PWR Dry Containment Parametric Studies*. NUREG/CR-5630. (SAND90-2339). Sandia National Laboratories. Albuquerque, New Mexico, April 1991.

Ref. 5.3-15 U.S. Nuclear Regulatory Commission. *Quantitative Analysis of Potential Performance Improvements for the Dry PWR Containment*. NUREG/CR-5575. (EGG-2602). EG&G Idaho, Inc. Idaho Falls, Idaho, August 1990.

Ref. 5.3-16 *CESSAR Design Certification*. Appendix U, Section 19.15.5. "Use of PRA in the Design Process." December 31, 1993.

Ref. 5.3-17 U.S. Nuclear Regulatory Commission. *Final Safety Evaluation Report Related to the Certification of the System 80+ Design*. NUREG-1462. Associate Director for Advanced Reactors & License Renewal. Washington, D.C., August 1994.

Ref. 5.3-18 Forsberg, C. W., E. C. Beahm, and G. W. Parker, "Core-Melt Source Reduction System (COMSORS) to Terminate LWR Core-Melt Accidents," Second International Conference on Nuclear Engineering (ICONE-2). San Francisco, California, March 21-24, 1993.

Ref. 5.3-19 Letter from Mr. D. E. Nunn (TVA) to Document Control Desk (NRC). "Watts Bar Nuclear Plant (WBN) Unit 1 and 2 - Severe Accident Mitigation Design Alternatives (SAMDAs) Evaluation from Updated Individual Plant Evaluation (IPE) (TAC Nos. M77222 and M77223)." June 30, 1994.

Ref. 5.3-20 Entergy Arkansas. *Arkansas Nuclear One Unit 1 Probabilistic Risk Assessment Summary Report*. April 1993.

Ref. 5.3-21 Entergy Arkansas. "Summary Report of Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities for Arkansas Nuclear One, Unit 1." May 1996.

Ref. 5.3-22 Florida Power & Light Company. *Applicant's Environmental Report, Operating License Renewal Stage, Turkey Point Units 3 & 4*, Appendix F. "Severe Accident Mitigation Alternatives Analysis." September 11, 2000.

Ref. 5.3-23 Duke Power Company. *Applicant's Environmental Report, Operating License Renewal Stage*, Attachment K. "Oconee Nuclear Station Severe Accident Mitigation Alternatives (SAMAs) Analysis." Rev. 0. June 1998.

Ref. 5.3-24 Letter from Mr. H. L. Sumner, Jr. (SNC) to Document Control Desk (NRC). "Edwin I. Hatch Nuclear Plant Application for Renewed Operating License." February 29, 2000.

Ref. 5.3-25 Letter from W.G. Gates (OPPD) to Document Control Desk (NRC). "NRC Generic Letter 88-20 Submittal for Fort Calhoun Station 'Individual Plant Examination for Severe Accident Vulnerabilities.'" December 1, 1993.

Ref. 5.3-26 Baltimore Gas and Electric. Applicant's Environmental Report – Operating License Renewal Stage – Calvert Cliffs Nuclear Power Plant Units 1 & 2. April 10, 1998.

a. *SAMA 60 - Please provide a brief description of the four basic events that were set to zero (Page 5-52).*

Response

The four basic events set to zero are ECBD1A11, ECBD1A31, ECBD1A22, and ECBD1A42. These events represent failures of the associated breakers that automatically open/close to fast transfer the 161 Kv offsite power source to onsite equipment. The description for each event is as follows.

- ECBD1A11 – Fast transfer AC breaker 1A11 fails to trip
- ECBD1A31 – Fast transfer AC breaker 1A31 fails to close
- ECBD1A22 – Fast transfer AC breaker 1A22 fails to trip
- ECBD1A42 – Fast transfer AC breaker 1A42 fails to close.

b. *SAMA 70 - It is not clear why this SAMA was considered to not apply since 2-out-of-4 logic is used at both FCS and the plant for which this SAMA was originally identified. Please provide a discussion of the significance of spurious safety system actuation events at FCS, and why this SAMA would not be effective for FCS.*

Response

This SAMA was misclassified. The SAMA was evaluated and found not to be risk significant, and should have been classified as meeting screening criterion D. The SAMA is focused on the potential change in the safety system and reactor trip actuation logic from its current design basis of 2 out of 4, to 3 out of 4. The 2 out of 4 logic allows for highly reliable safety system actuation at the expense of a greater spurious actuation and plant trip potential. In fact, the actuation logic was designed as a 2 out of 3 system with the fourth channel being an installed spare. This feature provides significant operational flexibility to the plant. Changing to 3 out of 4 logic would decrease the actuation system reliability and operational flexibility associated with plant operation with one channel inoperable. Hence, due to great expense and low (or negative) risk benefit associated with this change, SAMA 70 is screened out due to minimal risk (screening criterion D). Experience at FCS confirms that the 2 out of 4 logic has not resulted in operational problems.

c. *SAMA 84 - Provide a description of which penetrations constitute the dominant contributors to ISLOCA risk, and whether some subset of these lines can be tested at an increased frequency without the need for significant hardware modifications, thereby deriving some benefit without the large cost of adding or modifying test lines and instrumentation.*

Response

The dominant contributor to ISLOCA risk involves a failure of the reactor coolant pump seal cooler(s) and the inability to close the associated penetration isolation valve. The inability to close the valve includes operator error, mechanical failure, and failure of the supporting system(s) for the isolation valve. A plant modification was made to reduce the risk associated with the reactor coolant pump seal cooler to approximately 9.6E-07 per year. This modification involved reversing one of the containment isolation valves so that the pressure from the ISLOCA would force it closed rather than open.

Other less significant penetrations that contribute to ISLOCA risk include the safety injection lines, shutdown cooling suction line, and letdown line. The lines in these penetrations are equipped with multiple valves in series, each of which must fail in order to expose the low pressure piping to normal RCS pressure. The overall contribution to ISLOCA risk for the four safety injection lines is approximately 3.9E-8 per year. Likewise, the ISLOCA risk contribution for the shutdown cooling suction line and letdown line is approximately 3.0E-7 per year and 1.4E-9 per year, respectively. The overall ISLOCA risk for all of the above line penetrations is 1.3E-6 per year.

The line penetration for the reactor coolant pump seal coolers contributes approximately 74% to the overall ISLOCA risk and cannot be tested with the reactor coolant pumps operating. Each of the line penetrations for safety injection or letdown contributes less than 1.0E-8 per year (or less than 1%) to the overall ISLOCA risk. The ISLOCA risk for any of these line penetrations is considered small and no significant benefit would be derived by increasing the test frequency for these line penetrations.

The line penetration for shutdown cooling contributes approximately 23% to the overall ISLOCA risk. A relief valve is located in the high pressure piping on the isolated portion of the SDC line (SI-188). Any leakage into this volume would be expected to first lead to an overpressure condition and observable leakage. The second isolation valve will prevent overpressure failure of the low pressure RCS piping. Hence, no significant benefit would be derived by increasing the test frequency for the shutdown cooling line penetration.

d. *SAMAs 182, 183, and 184 - All three of these SAMAs call for a portable generator to be used for various applications What would be the power ratings of these portable sources for the various SAMA applications? Are such generators already available on the FCS site, and would these generators be dedicated for the purpose of the SAMA or installed on an ad hoc basis, if needed?*

<u>Response</u>

The power demands on the generators referred to in SAMAs 182, 183, and 184 are as follows:

<u>SAMA 182: Add Capability for SG Level indication during an SBO</u>

The proposed generator would have to supply about 5 amps at 125 VAC.

<u>SAMA 183: Add 480 volt power supply to open the PORVs</u>

A review of the PORV vendor manual indicates that the maximum inrush current for the PORV solenoid is 28.6 A. The recommended fuse is 20 A. Once open the steady state current to maintain the PORV open is 1.6 A. A generator-sizing estimate has not been performed.

<u>SAMA 184: Add Capability to flash EDG field</u>

The proposed generator would have to supply about 5 amps at 125 VDC. It has not been determined if generators on site could be used, or if new generators would have to be purchased. It also has not been determined if the generators would be dedicated, or if they would be installed on an ad hoc basis.

e. *SAMA 4 - In Table 4.16-2, page 4-42, OPPD indicates that the cost of SAMA 4 is >$30K. The staff assumes that this is a typo and that it should read "<$30K, as stated on page 5-45. Please confirm the proper value.*

Response

The proper cost of implementation for SAMA 4 is less than $30K.

f. SAMA 181 - In Table 4.16-2, page 4-44, OPPD indicates that the cost of the modification would "exceed" the estimated benefits. It appears that this should state that the cost (<$30K) is less than the benefit ($78K). Please clarify.

Response

Table 4.16-2 states that the "Cost of hardware modifications would exceed the estimated benefit."(emphasis added) As noted in Appendix 5, page 5-56, the alternative to the hardware modification (enhancing guidance to alert operators on available time before onset of premature RAS) would involve minimal cost and the estimated cost is less than $30K.

9. *In Table 4.16-2, page 4-42, OPPD lists the estimated benefit of each SAMA candidate. It is not clear whether the values presented in the table have been multiplied by two to account for external events (note comment on page 4-38 on the factor of two.) The following SAMAs appear to have a positive or neutral net value if the benefit values in Table 4.16-2 are doubled: SAMAs 54, 56, and 185. Clarify what the values represent.*

   In addition, SAMA 188 has a benefit (including the factor of two) of $32,000 while the cost is indicated as >2xbenefit. From the "description" on page 5-63, it is not clear whether this SAMA includes hardware changes. If procedural changes alone are sufficient for enhancing plant response to external flooding, then a procedural change (costing about $30,000) may be justified. Discuss the cost estimate and role of hardware modifications, if necessary.

Response

The benefit values provided in Table 4.16-2 have not been doubled. The SAMAs whose costs of implementation are in excess of 2 times the benefit are clearly indicated. The results discussion provided in Table 4.16-2 addresses the basis for screening those SAMAs whose cost exceeds the estimated benefit, but is less than 2 times the benefit (SAMAs 4, 54, 56, 185). The basis and screening results for these SAMAs is summarized below.

- SAMA 4 – This SAMA is included in the set of SAMAs OPPD plans to implement.
- SAMA 54 – Implementation of other SAMAs identified as cost beneficial will reduce the benefit of this SAMA; therefore, this SAMA is screened.
- SAMA 56 – The results indicate that this SAMA is potentially cost beneficial when applying a factor of 2 to the benefit results; however, when considering the probability of success the cost greatly exceeds to the benefits.

- SAMA 185 – Although the estimated cost of implementation exceeds marginally the estimated benefits, OPPD states in Table 4.16-2 that we will continue to evaluate this improvement outside of the SAMA process.

SAMA 188 was assumed to include both hardware modifications and procedural modifications. The hardware modifications envisioned included more reliable and robust pumps, perhaps with greater capacity, as well as an improved flow path that does not rely upon hoses and temporary fittings. It was concluded that the combined cost of procedural and hardware improvements was not justified by the calculated risk benefit.

10. *It is indicated on page 4-34 that the net present value of cleanup and decontamination over the life of the plant (U_{cd}) is \$1.61E+10. The correct value appears to be \$1.16E+10 based on the equation presented on page 4-33 of the application. Please confirm the correct value.*

<u>Response</u>

The value of U_{CD} given on page 4-34 of the environmental report is a typographical error. The correct value of 1.16E+10 was used in the calculation.