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NLS2009048
July 1, 2009

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to Request for Additional Information for License Renewal Application – Severe Accident Mitigation Alternatives
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:

1. Letter from Tam Tran, U.S. Nuclear Regulatory Commission, to Stewart B. Minahan, Nebraska Public Power District, dated June 8, 2009, "Request for Additional Information for the Review of the Cooper Nuclear Station License Renewal Application (TAC No. MD9763 and MD9737)."

2. Letter from Stewart B. Minahan, Nebraska Public Power District, to U.S. Nuclear Regulatory Commission, dated September 24, 2008, "License Renewal Application."

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to respond to Section A of the Nuclear Regulatory Commission Request for Additional Information (RAI) (Reference 1) related to the Cooper Nuclear Station License Renewal Application (LRA) Environmental Report severe accident mitigation alternatives. These responses are provided in Attachment 1. Certain changes to the LRA (Reference 2) have been made to reflect these RAI responses. These changes are provided in Attachment 2.

Should you have any questions regarding this submittal, please contact David Bremer, License Renewal Project Manager, at (402) 825-5673.

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NRR

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

Executed on July 1 2009
(Date)

Sincerely,

Stewart B. Minahan
Vice President – Nuclear and
Chief Nuclear Officer

/wv

Attachments

cc: Regional Administrator w/ attachments
USNRC - Region IV

Cooper Project Manager w/ attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/ attachments
USNRC - CNS

Nebraska Health and Human Services w/ attachments
Department of Regulation and Licensure

NPG Distribution w/o attachments

CNS Records w/ attachments

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©[4]

Correspondence Number: NLS2009048

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITMENT NUMBER	COMMITTED DATE OR OUTAGE
None		

Attachment 1

Response to Request for Additional Information
for License Renewal Application – Environmental Report
Severe Accident Mitigation Alternatives Analysis
Cooper Nuclear Station, Docket No. 50-298, DPR-46

The Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) regarding the License Renewal Application Environmental Report severe accident mitigation alternatives (SAMA) analysis are shown in italics. The Nebraska Public Power District's (NPPD) response to each RAI is shown in block font.

NRC Request: *ENV-SAMA-1*

1. *Provide the following information regarding the probabilistic safety analysis (PSA) used for the severe accident mitigation alternatives (SAMA) analysis:*

   a. *There are three core damage frequency (CDF)-related estimates reported in Attachment E to the environmental report (ER); i.e., 9.27E-06 per year in Table E.1-1 (CDF by Major Initiators), 9.23E-06 per year in Table E.1-8 (CDF by Plant Damage State), and 1.16E-05 per year in Table E.1-9 (Release Frequency by Release Category). Explain the reasons for the differences in these values, and the rationale for selecting the total release frequency as the baseline frequency for evaluating SAMA benefits.*

   b. *Section E.1.4.6 describes the May 2008 Boiler Water Reactor Owners Group (BWROG) peer review of the 2007TM model, Revision 1. Cooper Nuclear Station's (CNS's) review of the preliminary peer review findings determined that resolution of the findings would not result in a significant impact on the probabilistic risk analysis (PRA) results, and that the areas considered "not met" or capability category I have a negligible effect on the baseline CDF. For each peer review finding, provide a summary of the finding and an assessment of the impact of resolution of the finding on the SAMA identification and analysis results. The response should also address each of the supporting requirements having a capability category considered "not met" or capability category I, and discuss its potential impact on the SAMA identification and analysis results.*

NPPD Response:

1.a. The difference between the CDF in Table E.1-1 (CDF by Major Initiators) and the CDF in Table E.1-8 (CDF by Plant Damage State) is due to the method of quantification. The former utilized a "one top" quantification whereas the latter utilized a sequence

quantification. These quantification methods yield slightly different results due to variations similar to rounding differences in algebraic calculations.
The value of 1.16E-05/rx-yr in Table E.1-9 (Release Frequency by Release Category) is higher than the Level 1 CDF due to the manner in which the Level 2 containment event tree (CET) model was constructed and quantified. The Level 2 CET model used to evaluate the SAMAs was a single top model linked with the Level 1 fault tree. The CET model included numerous split fractions with failure paths that did not meet the rare event approximation (i.e., greater than 5E-2). The associated success paths were modeled with complementary events that approximated the expected success path probabilities (i.e., 1.0 failure probability). Thus, the sum of the split fraction events was slightly larger than 1.0. The cumulative effect of these success path approximations on the end state frequencies caused the difference between the Level 1 CDF frequency and the total Level 2 end state frequency.

The total release frequency is appropriate for use as the baseline frequency for evaluating SAMAs because the SAMA evaluation requires Level 2 analysis. To evaluate the benefit of each SAMA, the Level 1 or Level 2 portion of the model was modified as appropriate and the change in the total release frequency was used to provide a bounding estimate of the change in risk that could be achieved by implementation of the SAMA. The model that was used to analyze the SAMAs is the same model that provided the baseline total release frequency value of 1.16E-05/rx-yr. Since the SAMA analysis is based on the difference between the baseline model frequency and the "SAMA model" frequency, it is appropriate to use the same fault tree and CET model to calculate both frequencies.

The success path approximations in the Level 2 CET model have negligible impact on the results of the SAMA analysis because the effects tend to be cancelled out by the delta-CDF calculation and are further mitigated by bounding analyses and by use of the factor of 3 to account for uncertainties in the analysis.

1.b. The May 2008 BWROG peer review identified 22 findings, each against a different supporting requirement (SR). The SRs with findings were classified as follows:

a) Ten (10) SRs classified as not met,
b) Two (2) SRs classified as met (cat. I), and
c) Ten (10) SRs classified either as met (cat. II), met (cat. I/II), met (cat III), met (cat. II/III), or met (cat. I/II/III).

The following tables contain summaries of these SRs and the associated peer review finding. It also includes an assessment of the impact of resolving each finding on the SAMA identification and analysis results. Table 1 contains the findings for SRs classified as not met, Table 2 contains the findings for SRs classified as met (cat. I), and Table 3 contains the remaining findings.

Table 1: Findings for SRs Classified as Not Met

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA
HR-G7	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including: (a) time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel) [Note (1)]	Dependencies between post-initiator actions have been accounted for in general and appear adequate. However, there are some dependencies which do not appear to have been evaluated (or at least documented). In particular, the use of two different "floors" for joint HEPs is a little questionable, as is its application. The Cooper HRA identifies cutsets with multiple HFEs and provides a method for assessing the degree of those dependencies. May want to consider using more recent dependency models. The peer review referred to the following five dependencies sets of examples: • %FLSWRBM * FLD-XHE-FO-MSWRB * FLD-XHE-FO-SWRS1 * SWS-XHE-FO-SWNHP • FPS-XHE-FO-RPVIN * HVC-XHE-FO-ALTQC • ADS-XHE-FO-TRANS * HVC-XHE-FO-CB7A • ECS-XHE-FO-TRANS * SWS-XHE-FO-SWBPS • ADS-XHE-FO-3ALEG * SWS-XHE-FO-SWBPS	Each of the identified human error probability (HEP) combinations identified in the finding were reexamined. These combinations of HEPs represent combinations of HEPs that were evaluated after the original dependent HEP cutset calculation. These were evaluated during the final model review and determined to be composed of independent HEPs that led to combined probabilities above the applicable floor. As a result, no additional dependent combinations were developed. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
HR-I3	DOCUMENT the key assumptions and key sources of uncertainty associated with the human reliability analysis.	The Cooper PSA generally provides very detailed documentation, but, there is no discussion of sources of uncertainty regarding HRA consistent with the intent of SR HR-I3. Given the sensitivity to the issue of sources of uncertainty (as evidenced by NRC Memorandum, "Notice of Clarification to Rev. 1 of Regulatory Guide 1.200", July 27, 2007, NRC ADAMS Accession number ML071170054), and the ASME Standard highlighting this specific issue in all Technical Elements, the intent of SR HR-I3 is judged not met by the current Cooper PSA documentation.	Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA
IE-D3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in QU-E1 and QU-E2) associated with the initiating event analysis.	The requirement is to document the key assumptions and key sources of uncertainty. The assumptions used for initiating events were scattered throughout the document (CNS PSA-001). Uncertainty bounds were established, but sources of uncertainty were not discussed.	Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
IF-B2	For each potential source of flooding, IDENTIFY the flooding mechanisms that would result in a fluid release. INCLUDE: (a) failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. (b) human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system (c) other events resulting in a release into the flood area	PSA-012 Appendix E identifies failure modes of pipes and components for each source. The components are not specifically identified, but are included in the totals. The only failure mode of components identified is rupture. Other failure modes are not discussed. Human induced floods are dismissed in section 2.2.9.1. The main argument is that the generic pipe rupture frequencies already included these types of failures. This seems reasonable for pipe and component ruptures; however it does not include other types of spill scenarios (such as tank overfills). It is likely that these types of releases can also be screened due to alarms or other process parameters, but it is not in the documentation. Maintenance induced spills are dismissed in part by saying that personnel are available to detect the spill because they are the ones doing maintenance. This probably covers most maintenance activities, but it is not necessarily true for operational events that are performed remotely.	Subsequent to the PRA peer review, a supplemental evaluation was performed to estimate the potential contribution for human induced flooding. This evaluation concluded that the contribution of internal flooding events due to maintenance errors is quantitatively included in existing PRA quantifications. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA
IF-F3	DOCUMENT the key assumptions and key sources of uncertainty associated with the internal flooding analysis.	Significant sources of uncertainty have been identified. However, there is a lack of treatment in the uncertainty analysis regarding modeling assumptions and structure. For internal flooding, this may involve the assumptions regarding doors terminating flood propagation or varying the flood target population. For these reasons this SR is "Not Met".	Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
LE-G5	IDENTIFY limitations in the LERF analysis that would impact applications.	There is a requirement to discuss limitations in the LERF analysis that would impact applications. This was performed in the Level 1 analysis, but there is no evidence in the Level 2 analysis of a limitations discussion.	Assumptions associated with the CETs are summarized by individual CET node in the applicable Appendix C section of the PRA Summary Notebook. These assumptions may be viewed as introducing uncertainties on the use of the Level 2. However, there are no limitations that would impact projected applications that are not identified as part of the uncertainty evaluation in the PRA Summary Notebook (Appendices A, B, E). Accordingly, no changes were made to the model. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
MU-B2	PRA Configuration Control - Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are incorporated as soon as practical.	CNS Procedure ESPD-13 (PSA Model Maintenance and Update Procedure) mentions examples of some applications that need to be addressed or updated. However, there is no discussion of prioritization or urgency. Prioritization seems to be focused on base model and future applications rather than past applications and risk-informed decisions. Also, a timetable should be established to state when the application impacts need to be incorporated (e.g., six months after the PSA update is officially released).	Procedure (ESPD-13) was re-written to detail a new model maintenance and update process. The new process meets the requirements of the latest ASME PRA standard. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA
MU-E1	The PRA configuration control process shall include a process for maintaining control of computer codes used to support PRA quantification.	Software (including versions) should be specifically in SQA program, and the versions used should be consistent with the SQA program.	Revised the PRA configuration control procedure to ensure that the software control procedures are now used to control PRA software. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
MU-F1	The PRA configuration control process shall be documented. Documentation typically includes: (a) Description of the process used to monitor PRA inputs and collect new information (b) Evidence that the aforementioned process is active (c) Descriptions of proposed changes (d) Descriptions of changes in PRA due to each Update or Upgrade (e) Record of the performance and result of the appropriate PRA reviews (f) Record of the process and results used to address the cumulative impact of pending changes (g) Record of the process and results used to evaluate changes on previously implemented risk-informed decisions (pursuant to MU-D1) (h) Description of the process used to maintain software configuration control.	Process exists to review past PRA applications and determine if an update to the risk informed application is required when the PRA model is updated. Cannot see evidence that the aforementioned process is active. List of applications is not up-to-date.	No finding presented. (Accordingly, no changes were made to the model.) The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
QU-F5	DOCUMENT limitations in the quantification process that would impact applications.	No discussion of limitations for applications in the documentation. A limitation is that not including IE fault trees in the main model yields incorrect importance measures for events/components in the IE fault tree. (QU-F5-01)	The PRA Maintenance and Update procedure, and application-specific guidelines will be updated to ensure limitations in the quantification are documented. (Accordingly, no changes were made to the model.) The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

Table 2: Findings for SRs Classified as Met (Cat. I)

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
IE-A7	No requirement for precursor review.	There is no evidence in the notebook that a precursor review was performed. Response to the question was Table 2.3-3 and LER review performed. The items in the table were all plant scrams. The LER review would contain non-scram precursors however. A question was asked to the CNS team, and the response pointed back to the support system initiator development, which is covered by another SR. This SR of Category I (no requirement for precursor review).	Extensive plant-specific review of operational experience to identify precursors is documented in the Initiating Event Notebook. (Accordingly, no changes were made to the model.) The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
QU-E3	ESTIMATE the uncertainty interval of the overall CDF interval of the overall CDF results. Provide a basis for the estimate consistent with the characterization of parameter uncertainties (DA-D3, HR-D6, HR-G9, IE-C13).	This is described in Appendix A of the Quantification Notebook. Type-code database deals with "state of knowledge" correlation. But, for many components with plant-specific data, even though there are multiple identical components with the same failure probability, type codes were not used. This means that the "state-of-knowledge" correlation is not correctly taken into account. Therefore, only Category 1 is met.	This modified database for UNCERT was available and was used in the CNS uncertainty calculations reported in the PRA; however, it was not provided to the PRA Peer Review Team as part of their evaluation. There is no impact on applications or the base model. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

Table 3: Findings for SRs Classified as Met (Cat. II), Met (Cat. I/II), Met (Cat III), Met (Cat. II/III), or Met (Cat. I/II/III)

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
AS-A2 (MET I/II/III)	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage	Some event sequences are terminated when core damage has not occurred within 24 hours. In some, a stable state has not been reached based on the associated supporting thermal hydraulic calculation. For example, in 1A-L1-HPCI (which supports sequence GTR-002), containment temperature is still increasing at the end of 24 hours and has reached 250 F. The operators will be directed to depressurize when temp reaches 280 F. Another node appears to be needed in the tree to get to a stable state.	Based on the high marks received from the peer review on both SRs AS-A2 and AS-A9, the low significance of the finding as provided by the peer reviewers, and reexamination of the results of the example provided, CNS sees no merit to further analysis. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
SC-B3 (MET I/II/III)	When defining success criteria, USE thermal/hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping (HLR-IE-B) and accident sequence modeling (HLR-AS-A and HLR-AS-B).	Appendix F of PSA-003 dismisses the need for long term core spray in large LOCA scenarios based on MAAP calculations. While consistent with existing PRAs, this needs to be addressed further. MAAP does not treat steaming in the low power bundles precisely. It is OK if recovery is imminent or if the core is going to a melt state, however for long term steady state at low water level it will over-predict the two phase level in the low power bundles. MAAP calculates an overall steaming rate and applies it evenly across all bundles. This provides an adequate collapsed level in each bundle, but the two-phase will be too high in the low power bundles. MAAP also does not behave as expected when calculating the individual node core power. Due to the way it handles the uranium group, the power shape calculated is flatter than expected. This could affect the two phase level as well.	General Electric calculations are the basis for the success criteria. The CNS success criteria are consistent with all boiling water reactor (BWR) PRAs reviewed under the BWROG certification program and NUREG-1150. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
SC-C3 (MET I/II/III)	DOCUMENT the key assumptions and key sources of uncertainty associated with the development of success criteria.	It is possible that the success criteria uncertainty is addressed implicitly in the other elements; however, the treatment of the uncertainty is characterized as an increase or decrease in reliability. Changes to success criteria would be a logic change and is more difficult to deal with in sensitivity analyses.	Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
SY-A4 (MET II/III)	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	IPE system notebook and internal flooding walkdowns are listed in the self-assessment as references for SR SY-A4. However, the IPE walkdowns were not recently performed and the internal flooding walkdowns were performed with different goals in mind. Also, operator interviews were conducted for the HRA analysis and accident sequence modeling, but were not performed for the system analysis and documented.	There is no requirement in SR SY-A4 that system walkdowns be performed at each update. Numerous system walkdowns have been performed over the years in support of the CNS PRA and its applications. The value of the walkdowns and information gained must be balanced against the dose received performing the walkdowns. Some areas are inaccessible for routine walkdowns. The level of effort required to continually perform new walkdowns is inconsistent with the usefulness of such effort. Accordingly, no changes were made to the model. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
SY-A14 (MET I/II/III)	In meeting SY-A12 and SY-A13, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met: (a) A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. (b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same.	SY-A13 imposes the requirement to explicitly include in the model failure modes such as "Fails to Remain Open/Closed". SA-A14 provides criteria for excluding these failures modes. The failure modes were generally excluded from the CNS system fault trees, but no documented assessment of criteria in SY-A14 was found.	There is no documentation requirement for the basis for the exclusion of low probability failure modes. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
DA-E1 (MET I/II/III)	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.	The Maintenance Rule program processes and procedures and other plant data sources are relied upon to meet many of the aspects of the Data Analysis section. The processes that are used to screen and incorporate data collected in the Maintenance Rule program into the PRA plant specific data used in the model are not found in the PRA documentation.	The basis for significance of this item is that the documentation doesn't facilitate peer review. Although addressing this item will help facilitate peer review, documentation issues such as this should be considered suggestions rather than findings. This documentation issue does not impact the technical adequacy of the PRA or its capability. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
DA-E3 (MET I/II/III)	DOCUMENT the key assumptions and key sources of uncertainty associated with the data analysis.	A source of uncertainty not treated relates to failure modeling for certain equipment failure modes. A normally open MOV, which is required to remain open, can spuriously close. This can happen during the 24-hr mission time after the initiator. It is also possible that it could happen during plant operation and not be detected. This should be discussed for all equipment failure modes that are modeled and which are subject to failure either prior to the initiator or after the initiator. Even though one or the other contribution to failure may be evaluated to be negligible, this should be addressed. The negligible contributor still makes a contribution to uncertainty.	Findings related to PRA base model uncertainty characterization have been resolved subsequent to the peer review. These peer review findings are identified as DA-E3-02, HR-I3-01, IE-D3-01, IF-F3-01, and SC-C3-01. Resolution of the findings involved validation of the CNS PRA against applicable industry guidelines and required no changes to the PRA or the PRA documentation. Resolution was completed through validation that the guidance provided by NUREG-1855 and EPRI TR-1016737 was appropriately used to characterize the uncertainty relevant to the base CNS PRA. This guidance was in draft form during the peer review and therefore not available. The lack of final guidance on characterizing uncertainty in the base PRA model resulted in the CNS peer review findings in the supporting requirements subject to uncertainty evaluation. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
HR-G6 (MET I/II/III)	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	The HRA documentation mentions that the resulting HEPs were reviewed against each other. However, it isn't clear how this was done. For example, the HEP for the operator action to initiate drywell spray (RHR-XHE-FO-SPRAY) is about a factor of 3 times lower than the action to initiate torus cooling (RHR-XHE-FO-RHRE & RHR-XHEFO-RHRL). It's not obvious why this is logical given that the action to align torus cooling should be one of the most reliable actions given that it is performed fairly routinely (i.e., following any plant trip or manual shutdown). Also, according to the HRA calculator worksheets for these HFEs, the time available to align torus cooling is 886 minutes, while it's only 290 minutes for drywell spray. In addition, the operator action to perform emergency depressurization (ADS-XHE-FOTRANS) has a slightly lower HEP than the action to perform torus cooling, even though the time available for performing emergency depressurization is less than 30 minutes and under higher stress conditions.	The HEPs were reviewed for consistency and reasonableness. No changes to the HEPs or the model were necessary. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

SR Designator	Supporting Requirement (SR) Description	Peer Review Finding/Observation Summary	Impact on PRA Module
IF-B1 (MET I/II/III)	For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, feedwater system, condensate and steam systems) (b) plant internal sources of flooding (e.g., tanks or pools) located in the flood area (c) plant external sources of flooding (e.g., reservoirs or rivers) that are connected to the area through some system or structure (d) in-leakage from other flood areas (e.g., back flow through drains, doorways, etc.)	Components evaluated as flood initiators should be specifically identified in PSA-012 Appendix E. The components should be identified similar to the way that the pipes are.	The equipment that could be sources of flooding in each area are identified in the walkdown sheets located in the Internal Flood Walkdown Notebook. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.
QU-D4 (MET I/II/III)	REVIEW a sampling of no significant accident cutsets or sequences to determine they are reasonable and have physical meaning.	Could not find evidence of review of non-significant cutsets to determine if they are reasonable. Documentation is available that shows review of high level cutsets (top 100-200).	No documentation requirement exists in the ASME PRA Standard for this item. It appears that the review team is interpreting that detailed write-ups are required for every detailed step taken in the development of the PRA. This is inappropriate and not feasible. The subject review of non-significant cutsets was performed multiple times during draft quantifications of the model, as well as the final documented dominant sequence and cutset discussions in the PRA Summary Notebook. The PRA does not maintain hand mark-ups of draft quantifications and associated fixes. To assess this SR as "Not Met" because the review team requires detailed write-ups of more cutsets and sequences, and write-ups of hand mark-ups and corrections in draft quantifications is judged by NPPD to be beyond the intent of this SR. The resolution of the peer review finding did not result in changes to the PRA model and hence the finding has no impact on the SAMA identification and analysis results.

NRC Request: *ENV-SAMA-2*

2. *Provide the following information relative to the Level 2 analysis:*

   a. *Table E.1-10 identifies the release fractions for each release mode. The release fractions for iodine and cesium for the LL/I release mode are significantly lower than the corresponding release fractions for both the LL/E and LL/L release modes. Provide an explanation for this apparent anomaly.*

   b. *In the discussion of the Level 2 analysis (Section E.1.2), the process used to develop and group the source terms into containment event tree (CET) end states is not clear.*

      i. *Clarify whether a single CET was used for the grouping of source terms, or whether a single CET was used for each accident class or for each plant damage state (PDS). Provide a typical CET showing release categories assigned to each end state.*

      ii. *For each CET sequence, mass fractions obtained from the representative MAAP calculations were "weighed according to the contribution of that sequence to the sum of the sequences in the end state bin" (page E.1-68). Identify and describe the number of MAAP calculations or runs made to obtain the mass fractions. Provide an example of the weighting calculation for a representative CET sequence.*

   c. *Table E.1-5 shows three events (CNT-SMP-FF-MLTOF, RPV-DWV-FO-BARIS, and CNT-MDL-FF-WTRCV) having a risk reduction worth (RRW) value of 2.181. Provide details on which portions of the large early release frequency (LERF) model are affected in the computation of the RRW.*

NPPD Response:

2.a. As described in ER Section E.1.2.2.6, for each CET sequence, a value for each of the release-to-environment mass fractions was obtained from the representative MAAP calculation. These mass fractions were then weighted according to the contribution of that sequence to the sum of the sequences in the end state bin. The final mass fraction representing the end state bin was the sum of these individual weighted mass fractions for each species. An example of this calculation is provided in the response to Question 2b.

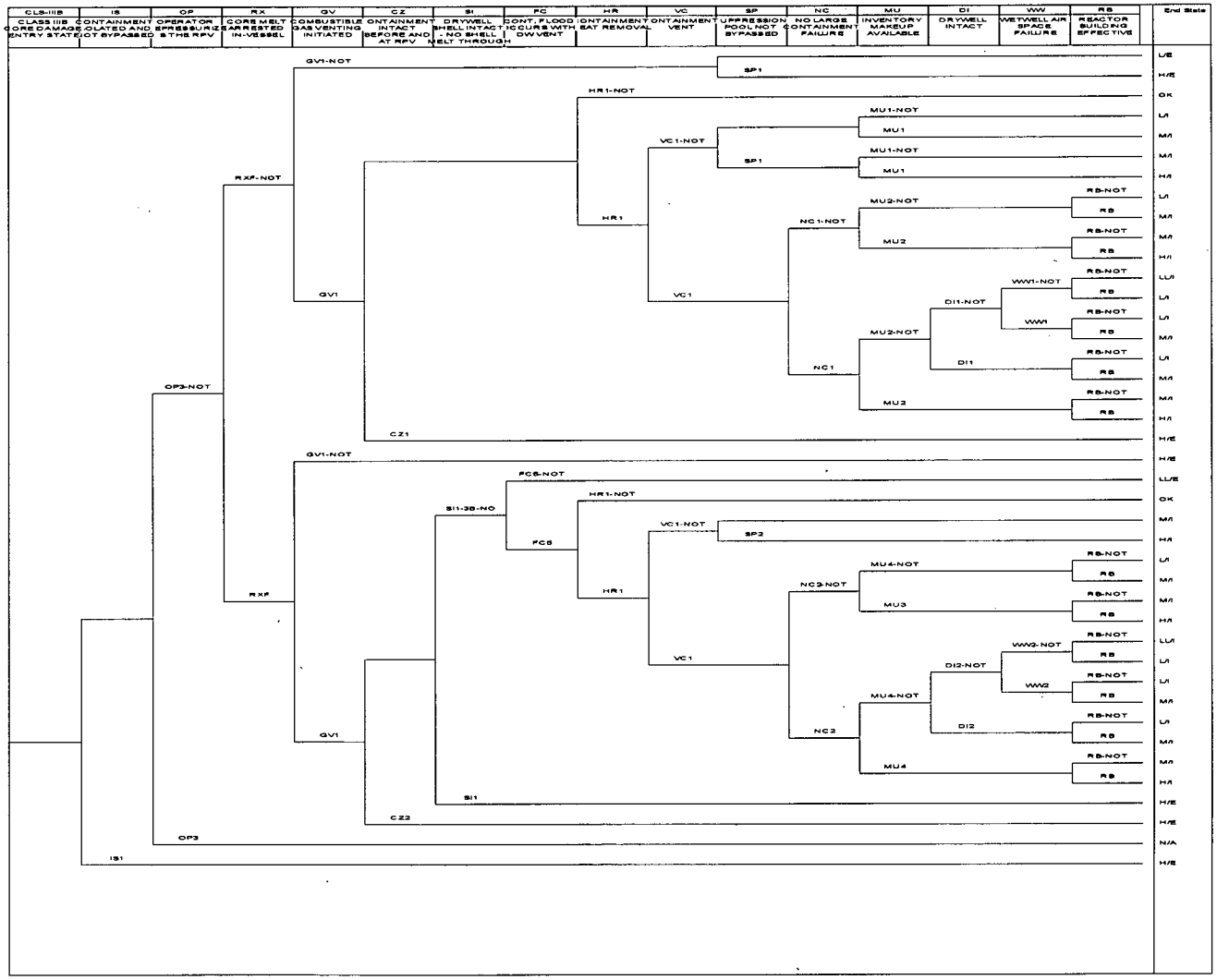
For the LL/I end state, the dominant sequences involve offsite release via containment venting through an intact suppression pool. This path results in very effective fission product scrubbing resulting in very low offsite releases. The MAAP run representing the

dominant sequences for the LL/I end state has a cesium iodide (CsI) release fraction of 1.79E-6.

For the LL/E and LL/L end states, the dominant sequences involve release paths from the primary containment that bypass the suppression pool with less effective fission product scrubbing. These sequences resulted in low-low offsite releases, but not as low as the LL/I sequences. The MAAP run representing the dominant sequences for the LL/L end state has a CsI release fraction of 8.48E-4. Also, the MAAP run representing the dominant sequences for the LL/E end state has a CsI release fraction of 1.7E-4.

Since the dominant sequences in the LL/I end state involve more effective fission product scrubbing through the suppression pool, the release fractions for iodine and cesium for the LL/I release mode are lower than the corresponding release fractions for both the LL/E and LL/L release modes. Although the difference appears significant, the impact is minimal because the LL end state grouping consists of those sequences expected to have a CsI offsite release magnitude of less than 0.1%.

2.b.i A single CET was developed for each PDS. A typical CET showing the release category assigned to each Level 2 end state in the CET is provided below.


CLS-IIIB
CLASS IIIB CORE DAMAGE ENTRY STATE
IS
CONTAINMENT ISOLATED AND NOT BYPASSED
OP
OPERATOR EPRESSURIZES THE RPV
RX
CORE MELT ARRESTED IN-VESSEL
GV
COMBUSTIBLE GAS VENTING INITIATED
CZ
CONTAINMENT INTACT BEFORE AND AT RPV
SI
DRYWELL SHELL INTACT - NO SHELL MELT THROUGH
FC
CONT. FLOOD IC CURS WITH DW VENT
HR
CONTAINMENT EAT REMOVAL
VC
CONTAINMENT VENT
SP
SUPPRESSION POOL NOT BYPASSED
NC
NO LARGE CONTAINMENT FAILURE
MU
INVENTORY MAKEUP AVAILABLE
DI
DRYWELL INTACT
WW
WETWELL AIR SPACE FAILURE
RS
REACTOR BUILDING EFFECTIVE
End State
GV1-NOT
SP1
HR1-NOT
MU1-NOT
MU1
VC1-NOT
SP1
MU1-NOT
MU1
RXF-NOT
MU2-NOT
RS-NOT
RS
HR1
NC1-NOT
MU2
RS-NOT
RS
GV1
VC1
WW1-NOT
RS-NOT
RS
DI1-NOT
WW1
RS-NOT
RS
MU2-NOT
DI1
RS-NOT
RS
NC1
OP3-NOT
MU3
RS-NOT
RS
CZ1
GV1-NOT
FC6-NOT
HR1-NOT
SI1-3B-NO
VC1-NOT
SP2
FC6
MU4-NOT
RS-NOT
RS
NC2-NOT
HR1
MU3
RS-NOT
RS
RXF
WW2-NOT
RS-NOT
RS
DI2-NOT
VC1
WW2
RS-NOT
RS
MU4-NOT
DI2
RS-NOT
RS
NC2
GV1
MU4
RS-NOT
RS
SI1
CZ2
OP3
IS1
U/E
H/E
OK
L/I
M/I
M/I
H/I
L/I
M/I
M/I
H/I
LL/I
L/I
L/I
M/I
L/I
M/I
M/I
H/I
H/E
H/E
LL/E
OK
M/I
H/I
L/I
M/I
M/I
H/I
LL/I
L/I
L/I
M/I
L/I
M/I
M/I
H/I
H/E
H/E
N/A
H/E

2.b.ii The following table identifies the 46 representative MAAP calculations used to obtain the mass fractions.

Case	Description	Comments
CN060500	Loss of Makeup at High Pressure - No Injection - No Drywell (DW) sprays	None
CN060500A	Loss of Makeup at High Pressure - Failure to Isolate Large Containment Penetration - No Injection - No DW sprays	A large 24 inch diameter path to the environment is opened at event initiation to simulate worst case failure of containment isolation. Containment failure time is reported when DW liner melt may be expected following Reactor Pressure Vessel (RPV) breach.
CN060501	Loss of Makeup at High Pressure - Open 1 Safety Relief Valve (SRV) at incipient core damage - No Injection - No DW sprays	This run demonstrates the impact of RPV pressure at time of vessel breach. The pressure in the RPV is reduced to approximately 65 psia at vessel breach, by opening one SRV at incipient core damage. DW pressure just prior to vessel breach is approximately 33 psia.
CN060502	Loss of Makeup at High Pressure - DW sprays - No Injection	DW sprays assumed to fail at containment failure for this run.
CN060503	Loss of Makeup at High Pressure - DW sprays operate - No Injection	DW sprays assumed to fail at containment failure for this run.
CN060504	Loss of Makeup at High Pressure - DW sprays operate - No Injection	DW sprays assumed to continue operating after containment failure for this run (required adjustment of Low Pressure Coolant Injection (LPCI) Net Positive Suction Head (NPSH) curve). CsI release is reduced to negligible fraction, however the case with DW spray failure at the time of containment failure is relatively small also (<1%).
CN060505	Loss of Makeup - RPV at Low Pressure – Emergency Depressurization (ED) at Minimum Steam Cooling Reactor Pressure Vessel Water Level (MSCRWL) - No DW sprays - No Injection	Without DW spray and no injection a large DW liner melt results in worst case radionuclide release for RPV at low pressure.

Case	Description	Comments
CN060505A	Loss of Makeup -RPV at Low Pressure - Failure to Isolate Large Containment Penetration - ED at MSCRWL - No DW sprays - No Injection	A large 24 inch diameter path to the environment is opened at event initiation to simulate worst case failure of containment isolation. Containment failure time is reported when DW liner melt may be expected following RPV breach.
CN060506	Loss of Makeup - RPV at Low Pressure - ED at MSCRWL - DW sprays operate - No Injection	DW sprays assumed to fail at containment failure for this run. Liner melt through occurs for this run after DW sprays fail on containment failure and water level in DW boils off to less than one foot.
CN060507	Loss of Makeup -RPV at Low Pressure - ED at MSCRWL - DW sprays operate - No Injection	Same as CN060506, except containment failure is assumed to be small for this run. DW sprays assumed to fail at containment failure for this run. Liner melt through occurs for this run after DW sprays fail on containment failure and water level in DW boils off to less than 1 foot. Liner melt is delayed slightly compared to CN060506 due to smaller containment breach.
CN060507A	Loss of Makeup -RPV at Low Pressure - ED at MSCRWL - DW sprays operate - No Injection	Same as CN060506, except containment failure is assumed to be small for this run and DW sprays assumed to continue operating after containment failure (required adjustment of LPCI NPSH curve). This run was specified to investigate the impacts of small vs. large containment failure, without a large DW liner melt.
CN060508	Loss of Makeup -RPV at Low Pressure - ED at MSCRWL - DW sprays operate - No Injection	DW sprays assumed to continue operating after containment failure for this run (required adjustment of LPCI NPSH curve).
CN060509	Loss of Makeup -RPV at Low Pressure - ED at MSCRWL - DW sprays operate – Suppression Pool Cooling (SPC) at four hours - No Injection	Turning on one loop of SPC at four hours avoids exceeding Primary Containment Pressure Limit (PCPL) and containment failure curve.

Case	Description	Comments
CN060510	Loss of containment heat removal - ED at MSCRWL - Core Spray (CS) and Control Rod Drive (CRD) (scram flow) available initially	CS pump fails as containment pressure rises above pressure to maintain SRVs open and shutoff head conditions met. Core damage occurs after containment failure at the Wet Well (WW). DW liner melt through occurs after RPV breach.
CN060510T	Loss of containment heat removal - ED at MSCRWL - CS available initially	CS pump fails as containment pressure rises above pressure to maintain SRVs open and shutoff head conditions met. Core damage occurs before containment failure at the WW. Core damage occurs before containment failure in this run because no high pressure injection exists (e.g. CRD) at the time the SRVs go closed. DW liner melt through occurs after RPV breach.
CN060511	Loss of containment heat removal - ED at MSCRWL - CS and CRD (scram flow) available initially	CS pump fails as containment pressure rises above pressure to maintain SRVs open and shutoff head conditions met. Core damage occurs after containment failure at the DW. DW liner melt through occurs after RPV breach. This run is the same as CN060510 except containment failure occurs at the DW head, which shortens the time 10% CsI release. CRD makes up for decay heat losses following CS pump shutoff, until containment failure.
CN060511T	Loss of containment heat removal - ED at MSCRWL - CS available initially	Same as case CN060510T except large containment failure occurs at the DW head instead of WW. Note that for this case 10% CsI is released to the environment prior to liner melt due to containment failure location being at the DW head prior to RPV breach.
CN060512	Loss of containment heat removal - Large LOCA (LLOCA)- CS available	CS prevents core damage until containment failure on failure curve.
CN060513	Loss of containment heat removal - LLOCA - CS available	Same as case CN060512 except containment failure location is DW instead of WW. This run shows the significant contribution of suppression pool scrubbing on CsI release. CS prevents core damage until containment failure on failure curve.

Case	Description	Comments
CN060514	LLOCA with two vacuum breakers failed open - No Injection - No Sprays	Containment failure at DW assumed to occur one minute following LOCA due to vapor suppression bypass.
CN060515	LLOCA with two vacuum breakers failed open - No Injection - DW Sprays operate	Containment failure at DW assumed to occur one minute following LOCA due to vapor suppression bypass. DW sprays operate within five minutes of Emergency Operating Procedure (EOP) trigger and eventually fail at approximately 4.6 hours due to assumed NPSH issues. This run is similar to CN060514 except DW sprays are used and it demonstrates their ability to keep CsI release relatively low.
CN060516	LLOCA with two vacuum breakers failed open- No Injection - No Sprays	Containment failure at WW assumed to occur one minute following LOCA due to vapor suppression bypass. This run shows the extended time and reduction in CsI release for a WW airspace failure compared to run CN060514 (DW failure).
CN060517	LLOCA - No Injection - No Sprays - DW venting at PCPL	Although DW liner melt may occur during this run, it was not allowed, in order to capture the impact of venting. Venting via a one inch path commenced at PCPL (12.15 hours) and slowly reduces containment pressure, until a pressure spike requires opening two inch path at 21.23 hours. Vents were never closed once opened.
CN060517A	LLOCA - No Injection - No Sprays - WW venting at PCPL	Although DW liner melt may occur during this run, it was not allowed, in order to capture the impact of venting. Venting via a 20-inch WW path commenced at PCPL (12.15 hours) and quickly reduces containment pressure. This is considered a bounding Hard Pipe Vent (HPV) continuous release from the WW to environment.

Case	Description	Comments
CN060518	Anticipated Transient Without Scram (ATWS) with Standby Liquid Control (SLC) Failure, RPV depressurized, DW head failure location, High Pressure Coolant Injection (HPCI) and LPCI utilized with level control	SPC fails following containment failure at 260°F pool temperature. LPCI NPSH required adjusted to 60% of manufacturer.
CN060519	ATWS with SLC Failure, RPV not depressurized, DW head failure location, HPCI and LPCI utilized with level control	Due to run not allowing depressurization, HPCI assumed failure on high suppression pool temperature results in loss of injection and early core damage. LPCI NPSH required adjusted to 60% of manufacturer.
CN060520	ATWS with SLC Failure, RPV depressurized, WW failure location, HPCI and LPCI utilized with level control	SPC fails following containment failure at 260°F pool temperature. LPCI NPSH required adjusted to 60% of manufacturer.
CN060521	ATWS with SLC Failure, RPV not depressurized, WW failure location below water line, HPCI and LPCI utilized with level control	Due to run not allowing depressurization, HPCI assumed failure on high suppression pool temperature results in loss of injection and early core damage. LPCI NPSH required adjusted to 60% of manufacturer.
CN060522	ATWS with SLC Failure, RPV depressurized, WW failure location below water line, HPCI and LPCI utilized with level control	SPC fails following containment failure at 260°F pool temperature. LPCI NPSH required adjusted to 60% of manufacturer. Decontamination Factor for Reactor Building (DFRB) is higher for this run due to suppression pool draining into torus room, resulting in code seeing increased CsI in the Reactor Building (RB) not making it to the environment.
CN060523	ATWS with SLC Failure, RPV not depressurized, DW head failure location, HPCI and LPCI utilized with level control	Due to run not allowing depressurization, HPCI assumed failure on high suppression pool temperature results in loss of injection and early core damage. LPCI NPSH required adjusted to 60% of manufacturer.
CN060524	Break Outside Containment - No Injection	None

Case	Description	Comments
CN060525	Station Blackout (SBO) - ED at MSCRWL - No DW sprays - No Injection	No real Direct Current (DC) power limitations impact for this run, due to assumption of no injection. RPV failure occurs before DC power is challenged, resulting in failure at low RPV pressure.
CN060525A	SBO - ED at MSCRWL - No DW sprays Reactor Core Isolation Cooling (RCIC) Injection	RCIC trips on high exhaust back-pressure after approximately six hours. ED on MSCRWL occurs within an hour of RCIC loss. Assumed battery depletion at 11 hours results in RPV repressurization and RPV fails at high pressure.
CN060525B	SBO - ED at MSCRWL - No DW sprays - HPCI Injection	HPCI is assumed failed at five hours for this run due to assumed battery depletion. RPV is depressurized by manually opening SRVs at MSCRWL. All batteries are assumed depleted at 11 hours, resulting in RPV pressure rising prior to vessel failure at high pressure due to potential battery impact on SRVs.
CN060525C	SBO - ED at MSCRWL - No DW sprays - RCIC Injection	RCIC trips on high exhaust back-pressure after approximately one hour. ED on MSCRWL occurs within an hour of RCIC loss. RPV breach occurs prior to 11-hour assumed battery depletion time, therefore, RPV pressure is low at breach.
CN060525D	SBO - ED at MSCRWL - No DW sprays - HPCI Injection	HPCI is assumed failed at five hours for this run due to assumed battery depletion. RPV is depressurized by manually opening SRVs at MSCRWL. All batteries are assumed depleted at 11 hours, resulting in RPV pressure rising prior to vessel failure. Vessel failure occurs with RPV pressure approximately 500 psia due to potential battery impact on SRVs.
CN060526	Loss of Makeup at High Press - DW sprays - SPC - No Injection	No containment melt or failure when sprays and SPC are successful.
CN060527	Loss of Makeup at Low Press - ED at MSCRWL - DW sprays - SPC - No Injection	No containment melt or failure when sprays and SPC are successful.

Case	Description	Comments
CN060528	LLOCA with two vacuum breakers failed open - DW sprays - SPC - No Injection	No containment melt or failure when sprays and SPC are successful. This run assumed two vacuum breakers were failed open, but did not fail containment automatically as was assumed in CN060516.
CN060529	Loss of Makeup at High Press - No Injection - No DW sprays - Flood containment with service water (SW) cross tie following RPV breach - Venting Throttled	Containment venting via WW until full, then switched to DW. Vent path size and duration used to control containment pressure within PCPL and 5 psi below PCPL. If largest vents were needed the code opens and remains open. DW liner melt was not allowed for SW cross tie runs. Depressurization via the RPV breach allows injection rates in excess of Minimum Debris Retention Injection Rate (MDRIR) via SW cross tie.
CN060529A	Loss of Makeup at High Press - No Injection - No DW sprays - Flood containment with SW cross tie following RPV breach - Venting Remains Open	Containment venting via WW until full, then switched to DW. Vent path size determined by ability to reduce/maintain pressure below PCPL. Vents remain open once opened.
CN060530	Loss of Makeup -RPV at Low Press - ED at MSCRWL - No Injection - DW sprays (within five minutes of RPV breach) - Flood containment with SW cross tie following RPV breach - Venting Throttled	Containment venting does not occur until DW is flooded to an elevation near the bottom of fuel. At which time the largest DW vent path is opened and remains open (arbitrarily assumed). PCPL is never exceeded.
CN060530A	Loss of Makeup -RPV at Low Press - ED at MSCRWL - No Injection - DW sprays (within five minutes of RPV breach) - Flood containment with SW cross tie following RPV breach - Venting Remains Open	This run result is the same as CN060530, since the only difference is vent remains open in this run. Since, PCPL was not exceeded until flood-up was near complete, the results are the same.

Case	Description	Comments
CN060531	LLOCA - No Injection - DW sprays (within five minutes of RPV breach) - Flood containment with SW cross tie following RPV breach - Venting Throttled	No throttling of venting was required as PCPL not exceeded until flood-up was near completion. Largest DW vent was opened and remained open at this time.
CN060532	Loss of Makeup at High Press - No Injection - No DW sprays Stuck Open Relief Valve (SORV) at time = 0	None
CN060533	Loss of Makeup at High Press - No Injection - No DW sprays - SORV at hot node temperature of 3000°F	None

The following example of the weighing process is provided for the IBL1-40 Level 2 sequence:

Sequence: IBL1-40
Sequence Probability: 1.0651E-7
CET end state for sequence: M/I
Total probability for all significant M/I sequences: 1.8327E-7
Fraction of sequence probability to total end state probability: 1.0651E-7/1.8327E-7 = 0.581
MAAP run representative of Sequence: CN060525C
CsI mass fraction obtained from MAAP results for CN060525C: 8.75E-2
Sequence CsI mass fraction contribution to total M/I end state mass fraction: 0.581 x 8.75E-2 = 5.08E-2
The mass fraction contribution from all M/I sequences was summed resulting in an end state mass fraction of: 1E-1

2.c. The three basic events CNT-SMP-FF-MLTOF, RPV-DWV-FO-BARIS, and CNT-MDL-FF-WTRCV all involve failure of the drywell shell (melt-through).

Basic event CNT-SMP-FF-MLTOF models the probability that the amount of core melt debris is greater than the sump volumes. If the sumps do not overflow, no debris comes into contact with the drywell shell. The drywell pedestal sumps are able to hold approximately 22% of the core debris; however, eventually more than 80% of the debris may be released from the RPV causing the sumps to overflow leaving a large fraction of

the core that could migrate to the drywell shell. Therefore, the probability of this event is set to 1.0 in the CNS PRA model.

Basic event RPV-DWV-FO-BARIS models the probability that barriers fail to block the debris from reaching the steel shell. This event is a result of the proposed Mark I containment modification to install a curb to prevent debris from spreading across the floor and contacting the shell. Since such a curb does not exist at CNS, the probability of this event is set to 1.0 in the CNS PRA model.

Basic event CNT-MDL-FF-WTRCV models the probability that the operator fails to align alternate injection sources (i.e., SW crosstie or fire protection system crosstie) in the Level 2 analysis. Following failure to recover injection systems in the Level 1 analysis, if the RPV is successfully depressurized, an additional time window is available (approximately three hours) for system recovery before RPV breach and the potential for shell attack occurs. Given that the operator has failed to align alternate injection in Level 1, the HEP accounts for recovery of injection sources within the extended time before RPV breach and the attendant containment challenges that would occur at RPV breach. The probability of this event is set to 0.9 in the CNS PRA model.

NRC Request: *ENV-SAMA-3*

3. *For each of the dominant fire areas, explain what measures, if any, have already been taken (since the individual plant examination of external events (IPEEE)) to reduce fire risk. Include in the response specific improvements to fire detection systems, enhancements to fire suppression capabilities, changes that would improve cable separation and drain separation, and improvements to processes/procedures for monitoring and controlling the quantity of combustible materials in critical areas.*

NPPD Response:

3. The IPEEE listed compartments 3A (switchgear room 1F), 3B (switchgear room 1G), 10B (control room and security access system corridor), and 20A (service water pump room) as the dominant fire areas.

   Fire protection improvements to monitor and control the quantity of combustible materials in critical process areas, and to monitor and control pre-staging of outage materials have been implemented since the IPEEE was submitted on October 30 1996, which would reduce CDF values for all of the dominant zones.

   The following discussion for each zone explains additional measures, if any, taken to reduce risk in that zone and explains why the fire CDF cannot be further reduced in a cost-effective manner.

Switchgear rooms 1F and 1G (Compartments 3A and 3B)

No design changes have been made in the switchgear rooms to reduce fire risk since the IPEEE. The switchgear rooms have a fire detection system which provides an alarm in the control room. Phase II SAMA 63 (not cost-beneficial) was evaluated to determine the benefit from adding automatic suppression systems to the switchgear rooms. Thus, no further cost-effective changes in this zone were identified to reduce CDF.

Main Control Room and Security Access System Corridor (Compartment 10B)

No design changes have been made in the control room and security access system corridor to reduce fire risk since the IPEEE. For this compartment, the sensing devices used for fires include both fuse elements (that melt given high temperature) and smoke detectors. The main control room is also continuously manned. Therefore, a fire in this compartment will result in prompt fire brigade response and manual extinguishment. An automatic suppression system would not provide a significant safety benefit. An automatic suppression system based on Halon or CO_2 would asphyxiate any personnel remaining in this compartment, and thus require evacuation. A water-based automatic suppression system would damage the control equipment.

Phase II SAMA 65 (not cost-beneficial) was evaluated to determine the benefit from upgrading the alternate shutdown panel to reduce the fire CDF from scenarios that result in main control room evacuation. Since the main control room is always inhabited ensuring prompt fire detection and manual suppression, no further cost-effective changes in this zone were identified to reduce CDF.

Service water pump room (compartment 20A)

No changes have been made in the service water pump room to reduce fire risk since the IPEEE. Since the service water pump room is equipped with a detection system that alarms in the main control room and a total flooding halon suppression system, no further cost-effective changes in this zone were identified to reduce CDF.

NRC Request: *ENV-SAMA-4*

4. *Provide the following information concerning the MACCS2 analyses:*

   a. *Specify the fraction of the public that was assumed to participate in an evacuation at CNS. NUREG-1150 assumed a 99.5 percent evacuation within the emergency planning zone (EPZ); previous SAMA analyses have assumed a 95 percent evacuation. If a 95 percent evacuation was not assumed at CNS, address the*

potential impact on the off-site exposure risk and averted public exposure cost if 5 percent of the population fails to evacuate the EPZ.

b. *In Section E.1.5.2.8, it is stated that the core inventory is based on a bounding reload core immediately following shutdown. Provide the core enrichment and burnup used in the MAACS2 analyses. Confirm that this core inventory reflects the expected fuel management/burnup during the renewal period.*

NPPD Response:

4.a. The fraction of the public that was assumed to participate in an evacuation at CNS is 1 (or 100 percent) in the SAMA analyses.

If only 95 percent of the population had been assumed to evacuate the EPZ, then the off-site exposure risk would have been 2.15 person-rem/yr for the baseline severe accident consequences. Similarly, the public exposure cost would have been $46,280. These numbers represent an increase of less than one percent of the values reported in Section 4.21.5.1.4 of the License Renewal Application Environmental Report (LRA-ER) assuming 100 percent evacuation. The fact that the site is in a low population area is the reason for the results. Consequently, the impact on the offsite exposure risk and averted public exposure cost would be insignificant if five percent of the population failed to evacuate the EPZ.

4.b. Core enrichment and burnup are not explicitly used in the MACCS2 analysis; rather the core inventory of key nuclides is used directly. The core enrichment and burnup assumed for determining the core inventory for the MACCS2 analysis are consistent with values used to determine the fission product inventory utilized in the current licensing basis dose calculations reflected in the CNS Updated Safety Analysis Report (USAR). The fission product activity inventory utilized a "bounding" core assumed to consist of an initial enrichment of 3.908 Wt % U-235 and high burnup GE14 fuel undergoing 1300 effective full power days (EFPD) of continuous irradiation to achieve an end of cycle core average exposure (CAVEX) of 35.8 Gwd/MT, which is more than six percent above the equilibrium core analyzed. Activity inventories were calculated with the Oak Ridge National Laboratory isotope generation and depletion code ORIGEN2, incorporating the BWR extended burnup library BWRUE. The core inventory utilized in the MACCS2 analysis assumed a thermal power level of 2429 MW(t), which encompasses the licensed maximum power level of 2419 MW (t), following the CNS Measurement Uncertainty Recapture (MUR) power uprate. This core inventory reflects the expected fuel management/burnup for the renewal period.

NRC Request: *ENV-SAMA-5*

5. *Provide the following with regard to the SAMA identification and screening process:*

   a. *Table E.1-5 identifies the RRW for events that contribute to LERF. The table appears to contain success events and associated SAMAs. For example, CGS-PHE-FF-INERT represents successful containment inerting and has the highest RRW value in the table (3.417). Another example is RPV-MDL-SC-C1A1E, successful depressurization (Class IA, IE), which has an RRW value of 1.335. Discuss the rationale for identifying SAMAs for success events.*

   b. *There appear to be events in Table E.1-5 that are complementary (e.g., as the probability of event A approaches zero, the probability of event B approaches 1.0. Evaluation of RRW for these events would need to consider this relationship. To examine how this relationship between events was addressed in Table E.1-5, provide the value used for CGS-PHE-SC-INERT (containment not inerted; venting required) when calculating the RRW for CGS-PHE-FF-INERT (containment inerted; venting not required). If the value of this and other complementary events were not directly coupled in the computation of the RRWs, provide a revised Table E.1-5 with these events appropriately addressed (i.e., the probability of event CGS-PHE-SC-INERT is set equal to 1.0 when the probability of event CGSPHE-FF-INERT is set equal to 0.0). Provide an assessment of the results on the SAMA identification and evaluation.*

   c. *From Table E.1-3, event TDCA (loss of 125 VDC A) and event EDC-XHE-FO-RSTA (failure to restore DC power within 30 minutes) have RRW values of 1.19, which would imply a potential CDF reduction for DC power improvements of about 20 percent. These events are addressed by SAMAs 1, 2, 3, 13, 14, 15, 19, and 21, which in turn are covered by analysis cases 1 through 5 and 14. However, these analysis cases show a CDF reduction (in Table E.2-2) of about 3 percent or less (except for analysis case 14). Explain why there are not additional DC power-related SAMAs for these basic events that have potentially greater CDF reduction impacts than 3 percent. For example, provide the rationale for why a procedural SAMA addressing event EDC-XHE-FO-RSTA with a potential CDF reduction of about 20 percent was not considered.*

   d. *Event PCI-CNT-FF-PREEX (pre-existing containment failure) has an RRW of 1.056. All of the SAMAs considered for this event involve major hardware modifications. Provide an assessment of the costs and benefits of lower cost SAMAs for this event (e.g., periodic monitoring of containment integrity during normal operation or procedures to isolate the containment following an event).*

e. *ER Section E.2.2, identified three criteria used to screen Phase I SAMAs, all of which are qualitative. However, information provided in the "Screening Results" column of Table E.2-1 suggests that other criteria were used, including "Small CDF Reduction" (SAMAs 209, 227, and 229), and "Outliers Were Resolved Analytically" (SAMAs 217, 220, 221, 224, 226, and 227). Clarify the criteria used to perform the Phase I screening.*

f. *Table E.2-2, describes SAMA 21 as being part of SAMA 13 and no separate evaluation is provided. Explain the rationale for including SAMA 21 as a unique SAMA if it cannot be implemented as an independent SAMA.*

g. *In Table E.2-1, SAMA 232 (protect the diesel exhaust from tornado generated missiles) is described as being resolved by a modification completed in 1998. However, in the NRC safety evaluation report (SER) on the CNS IPEEE dated April 2001, the issue is described as yet to be addressed in the IPEEE Issue Resolution Plan. Clarify this apparent discrepancy.*

NPPD Response:

5.a. As described in ER Section E.1.2.1, Table E.1-5 provides a correlation between the Level 2 RRW risk significant events (severe accident phenomenon, initiating events, component failures and operator actions) down to 1.005 identified from the CNS 2007TM Revision 1 PSA LERF model and the SAMAs evaluated in Section E.2.

Table E.1-5 includes basic events that are not failure events. Rather, they are "success events" used to identify the conditions under which the failure events in a particular sequence occur. These events were included in Table E.1-5 since they provide information about the model, but the SAMAs identified for these events do not have the same meaning as those identified for the component and human failure events. SAMAs identified for component and human failure events are intended to decrease the risk contribution from that failure event by decreasing the probability of the failure event itself. However, since decreasing the probability of a success event is not desirable, SAMAs identified for success events are intended to decrease the risk contribution from the cutsets or events related to the success event. The SAMAs identified for a success event in Table E.1-5 are also listed in Table E.1-5 for the specific component and human failure events in Table E.1-5 that are related to the success event.

For example, RPV-MDL-SC-C1A1E represents successful RPV depressurization. If the probability of successful depressurization increases, then the LERF contribution is decreased. The primary benefits associated with the ability to depressurize the RPV include the following:

- Enables injection to the RPV from low pressure injection systems, such as, LPCI, CS, condensate, and fire water.
- The stresses on primary system components are reduced; thereby increasing the likelihood that the primary system will remain intact.
- The likelihood that high pressure blowdown coupled with inadequate vapor suppression would lead to immediate containment failure is reduced.
- The likelihood that molten material will be finely dispersed in the containment atmosphere leading to a direct containment heating failure mode is reduced.

Thus, Phase II SAMAs 26, 27, 43, and 44, to enhance depressurization (or decrease the risk contribution from failure of depressurization), were listed for event RPV-MDL-SC-C1A1E.

Also, CGS-PHE-FF-INERT and CGS-PHE-SC-INERT are split fractions which indicate whether the containment is inerted. The failures that result in release with containment inerted are different than those with containment not inerted.

CGS-PHE-FF-INERT represents the containment being inerted, so combustible gas venting is not required. Since containment failure must occur to have a release in the absence of containment gas venting, SAMAs that decrease the probability of containment failure due to severe accident phenomena will decrease the LERF impact of cutsets containing CGS-PHE-FF-INERT. Thus, Phase II SAMA 70, to install a curb to prevent debris from spreading across the floor and contacting the shell, was identified to decrease risk from cutsets containing CGS-PHE-FF-INERT.

CGS-PHE-SC-INERT represents the containment not inerted and venting is required. Since release can occur without containment failure in cutsets with CGS-PHE-SC-INERT, Phase II SAMAs 48, 49, 50, and 52, to improve venting and fission product scrubbing and to provide passive overpressure relief, were identified to reduce risk from cutsets containing CGS-PHE-SC-INERT.

5.b. The value of CGS-PHE-SC-INERT and CGS-PHE-FF-INERT when calculating RRW were 0.01 and 0.99, respectively.

The split fraction events are utilized for sequence identification. As described in response to RAI 5.a., the LERF impact of cutsets containing CGS-PHE-FF-INERT or CGS-PHE-SC-INERT was reduced by decreasing the probability of individual events in the cutsets. The values of other split-fraction events were also directly coupled in the computation of RRW.

Since split fraction events were directly coupled in the computation of RRW, revision of table E.1-5 is not necessary and there is no impact on SAMA identification and evaluation.

5.c. The list of 244 Phase I SAMA candidates includes SAMAs from a number of industry documents and from previous SAMA submittals. This list included several DC power enhancements which would improve events TDCA (loss of 125 VDC bus A) and EDC-XHE-FO-RSTA (failure to restore DC power within 30 minutes), but these had already been implemented at CNS. The remaining applicable enhancements were evaluated as Phase II SAMAs 1, 2, 3, 13, 14, 15, 19, and 21. While each of these Phase II SAMAs is capable of improving the DC power system, none is capable of mitigating a complete loss of 125 VDC bus A. Thus, all postulated DC power-related SAMAs for these basic events have been implemented or evaluated.

Since DC power is a support system, SAMAs that mitigate failure of the supported systems would also reduce the contribution from failure of TDCA. TDCA is a dominant CDF contributor for two primary reasons.

1. Loss of 125 VDC bus A power results in loss of division 1 remote breaker and logic controls for division 1 low pressure emergency core cooling systems (ECCS), diesel generator 1, and reactor core isolation cooling. Additionally, loss of 125 VDC bus A results in loss of the low pressure injection source provided by the main condensate pumps because all three condensate pumps are lost. Pumps A and C are lost due to the inability of DC logic to transfer the alternating current (AC) buses to the startup transformer. Pump B is lost due to inadequate cooling caused by DC logic failure of two of the three turbine equipment cooling (TEC) pumps.

2. Loss of 125 VDC bus A power results in the loss of instrument air and the inability to open air-operated valves to vent containment using the wetwell or drywell vent systems. This leaves only the hard pipe vent method of venting containment. Instrument air is lost because the A compressor fails due to DC logic failures, and air compressors B and C lose cooling due to DC logic failure of two of three TEC pumps and two of four reactor equipment cooling (REC) pumps.

Therefore, the SAMAs postulated to improve low pressure injection (Phase II SAMAs 28, 29, 32, 47, 64, and 78) and containment venting (Phase II SAMAs 20, 48, 49, 50, 52, and 53) would also lower the contribution due to loss of 125 VDC bus A.

Basic event EDC-XHE-FO-RSTRA (failure to restore DC power within 30 minutes) is a recovery event, similar to loss of offsite power recovery events, with a probability based

upon industry experience. It is not based upon plant-specific human reliability analyses. CNS has detailed, symptom-based procedures which provide adequate guidance for loss of 125 VDC components.

5.d. Event PCI-CNT-FF-PREEX represents a pre-existing containment failure leading to loss of NPSH to the ECCS pumps. Lower cost SAMAs such as periodic monitoring of containment during normal operation and procedures to isolate containment following an event were not evaluated because they already exist at CNS. Containment is inerted during normal operation, so a leak large enough to lead to loss of NPSH to the ECCS pumps would result in the need for an unusual amount of nitrogen and would not go unnoticed.

Also, CNS procedures and programs monitor containment integrity. The containment leak rate program, in accordance with 10 CFR 50, Appendix J, includes local and integrated leak rate testing of primary containment pressure-retaining components. Also, the containment inservice inspection program, in accordance with ASME Section XI Subsection IWE and 10 CFR 50.55a, includes inspections of the primary containment and its integral attachments to ensure that degradation does not exist that would challenge the leak tight barrier.

5.e. Section E.2.2 has been revised to add statements that clarify the screening criteria (see Attachment 2).

5.f. SAMA 21, Modify plant procedures to allow use of a portable power supply for battery chargers, was included as a unique SAMA due to the belief that a suitable portable power supply was available to supply the battery chargers. Further investigation revealed that the available skid mounted portable power supply is not sufficient to supply the battery chargers. (The portable power supply is considered available in the cost estimates for SAMAs 14 and 20.) Since SAMA 13, purchase of a portable power supply, requires the plant procedure revisions, SAMA 21 is included as part of SAMA 13.

5.g. The modification to protect the diesel exhaust from tornado generated missiles was completed in 1998.

In response to the NRC Staff Evaluation Report on the Cooper IPEEE dated April 2001, clarification was provided in CNS letter (NLS2001057) to the NRC dated July 6, 2001, with the subject *"IPEEE Staff Evaluation Report Clarification and Commitment Status Update - Cooper Nuclear Station, NRC Docket 50-298, DPR-46."* The clarification indicated that the modification, which involved elimination of the bypass valves for the diesel generator mufflers and eliminated the failure mode described in the CNS IPEEE submittal, was completed in 1998.

NRC Request: *ENV-SAMA-6*

6. *Provide the following with regard to the Phase II cost-benefit evaluations:*

   a. *For a number of the Phase II SAMAs listed in Table E.2-2, the information provided does not sufficiently describe the associated modifications and what is included in the cost estimate. Provide a more detailed description of both the modifications and the cost estimates for Phase II SAMAs 20, 44, 45, 63, 70, 72, 73, 76, 77, and 80. Also, for SAMA 76 describe what is meant by "group 1 isolations" in the context of both the plant and the PRA model.*

   b. *Analysis case 14 covers SAMA 14 (portable generator for DC power to supply individual panels), SAMA 22 (install independent high-pressure injection (HPI) system), and SAMA 23 (additional HPI pump with independent diesel). For this case, high-pressure coolant injection (HPCI) unavailability was set to zero resulting in a 32 percent reduction in CDF. However, event HCI-SYS-TM-HPCI (HPCI unavailable due to test and maintenance) has an RRW equal to 1.03.*

      i. *Explain the rationale for assigning SAMA 14 to analysis case 14, and for estimating the benefit for this SAMA by setting HPCI unavailability to zero.*

      ii. *Explain the large CDF reduction (32 percent) when the RRW for HPCI unavailability due to test and maintenance would suggest only a 3 percent reduction. Clarify whether loss of DC panel power is the dominant contributor to HPCI unavailability. Identify the other significant contributors to HPCI unavailability.*

   c. *In Table E.2-2, the modeling assumption for SAMA 6 (i.e., change the time available to recover offsite power to 24 hours) is inconsistent with the modeling assumption of analysis case 6 (i.e., set failure to transfer the reactor protection system panels to their alternate power source to zero). Clarify which description is correct. Provide a revised evaluation, if necessary.*

   d. *Table E.2-2 describes the bounding analysis for SAMA 78 as "reducing operator actions that could be improved via training for alternate injection via the fire water system by a factor of 2." The description provided for the corresponding analysis case 23 (page E.2-8) is similar. Provide a clarification of how the evaluation of this SAMA was modeled in the PRA. Identify the associated operator action(s) and the initial and modified human error probability value(s).*

e. *SAMA 41 (modify procedure to provide ability to align diesel power to more air compressors) has an estimated cost of $1.2M. This cost appears high for what appears to be a procedure and training issue. Justify the cost estimate for this SAMA.*

f. *SAMA 69 (upgrade the seismic capacity of the diesel fire pump fuel tank and water supply tank) is intended to increase the reliability of the fire water system in seismic events. The benefit of this SAMA was determined by eliminating failure of the diesel-driven fire pump. Discuss how eliminating failure of the diesel-driven fire pump in the internal event model, in conjunction with the external event multiplier, captures the benefit from this SAMA in seismic events. Justify the benefit and cost estimates for this SAMA.*

g. *SAMA 14 (portable generator for DC power to supply the individual panels) and SAMA 13 (portable generator for DC power to supply the battery chargers) both involve use of a portable generator. SAMA 14 is cost beneficial while SAMA 13 is not. Discuss whether it is feasible for SAMA 13 to use the same portable generator as SAMA 14, and if so, provide a revised evaluation of SAMA 13.*

h. *SAMA 70 (install a curb to prevent debris from spreading across the floor and contacting the shell) has a CDF reduction of 11.6 percent. Explain how this SAMA reduces CDF, and identify the events in Table E.1-3 that are impacted.*

i. *SAMA 75 (implement generation risk assessment into plant activities) has a large benefit owing to the many risk contributors that it impacts. Provide a detailed description of this SAMA, including: (1) a discussion of how it would be implemented at CNS, (2) a more comprehensive description of analysis assumptions, (3) justification for the assumed factor of 2 reduction in initiating event frequency for affected events, and (4) a more detailed discussion of the estimated implementation cost.*

j. *Pages 4-83 and E.2-2 of the ER indicate that the SAMA cost estimates did not include the cost of replacement power during extended outages required to implement the modifications, nor did they account for inflation. Clarify how other cost factors were treated in these estimates, specifically, contingency costs associated with unforeseen implementation obstacles, and maintenance and surveillance costs.*

k. *Explain why a factor of 3 is used to represent uncertainty, given that the ratio of the 95th percentile CDF to the mean CDF is said to be 1.86. Provide the 5th percentile, mean, and 95th percentile CDF values.*

l. The CNS cost-benefit analysis showed that eleven of the SAMA candidates (SAMAs 14, 25, 30, 33, 40, 45, 64, 68, 75, 78, and 79) were potentially cost-beneficial.

i. The ER does not provide any indication of CNS's plans regarding the eleven Phase II SAMAs found to be potentially cost-beneficial. Describe CNS's plans regarding these SAMAs, and any other potentially cost-beneficial SAMAs that may emerge from further analyses in response to these RAIs.

ii. In view of the significant number of potentially cost-beneficial SAMAs, it is likely that several of these SAMAs address the same risk contributors. As such, implementation of an optimal subset of these SAMAs could achieve a large portion of the total risk reduction at a fraction of the cost, and render the remaining SAMAs no longer cost-beneficial. In this regard: (1) identify those SAMAs that Nebraska Public Power District (NPPD) considers highest priority for implementation, (2) provide an assessment of the impact on the remaining SAMAs if these high-priority SAMAs are implemented, and (3) identify those SAMAs that would no longer be cost-beneficial given implementation of the high-priority SAMAs. Also, provide any specific plans/commitments regarding implementation of the high priority SAMAs.

NPPD Response:

6.a. ER Section E.2.3 provides a general description of the cost estimating process. It notes that detailed site-specific cost estimates were not required for all SAMA candidates, but that the cost of each candidate was conceptually estimated to the point where conclusions regarding the economic viability of the proposed modification could be adequately gauged.

SAMA 20 provides redundant power to the direct torus hard pipe vent valves to improve their reliability and enhance containment heat removal capability. CNS already has a skid mounted portable AC generator that was used for the old Technical Support Center. The cost estimate for SAMA 20 assumes that this generator would be used to provide power to the valves; therefore a generator is not included in the cost estimate. The $714,000 cost estimate includes costs to provide electrical wall penetrations in the control building to allow generator hook-ups to the appropriate buses. It also includes costs to provide new or revised operating procedures, and to provide operator training.

Since the CNS SRVs and main steam isolation valves (MSIV) have redundant power supplies and accumulators, SAMA 44 would enhance the reliability of these valves by

replacing them with an improved design. As stated in the ER, the cost estimated for SAMA 44 came from a cost estimate for a similar SAMA at Pilgrim. In an RAI response letter (ADAMS Accession Number ML061930418), Pilgrim indicated that since their SRVs have redundant DC power supplies and back up nitrogen supplies, the only way to enhance SRV reliability would be to replace the valves with an improved design. The estimated cost to replace four SRVs with more reliable SRVs at Pilgrim was $1,500,000. Since CNS has eight SRVs and eight MSIVs, and the Pilgrim estimate to replace four valves is greater than the benefit (with uncertainty) of $574,974 for CNS SAMA 44, a detailed site-specific cost estimate was not needed to determine that SAMA 44 is not cost-beneficial.

The modification for SAMA 45 involves procurement of a portable compressor to be aligned to the supply header to reduce the risk associated with loss of instrument air. CNS recently replaced the three positive-displacement instrument air compressors with screw type compressors. While that modification was being performed, a rented portable compressor was used to supply instrument air. Therefore, the hook-ups remain available to support the SAMA 45 modification and are not included in the cost estimate. The cost estimate of $100,000 includes the cost of a new screw type compressor (approximately $85,000) as well as costs to mount the compressor on a trailer for mobility and necessary procedure changes. SAMA 45 has been retained on the list of SAMAs that is potentially cost-beneficial.

The modification for SAMA 63 is to add automatic gas suppression systems in the switchgear rooms. As stated in the ER, the cost estimate for SAMA 63 came from a cost estimate for a similar SAMA at Brunswick. NUREG-1437, Supplement 25, indicates that the estimated cost of installing an automatic gas suppression system in two reactor building areas was $750,000 at Brunswick. Since Brunswick is a two unit site, the estimate for CNS was reduced accordingly to $375,000. Since the Brunswick and CNS SAMAs both evaluate addition of two automatic gas suppression systems, use of the Brunswick estimate for CNS is appropriate. Since the estimated cost is greater than the benefit (with uncertainty) of $347,908, a detailed site-specific cost estimate was not needed to determine that SAMA 63 is not cost-beneficial.

SAMA 70 is to install a curb to prevent debris from spreading across the floor and contacting the drywell shell. The curb would be steel reinforced concrete with a diameter of approximately 50 feet, six inches high and three feet wide. The curb may not be a perfect circle to allow for routing around installed equipment and existing structures. Drilling into the existing containment floor may be necessary to ensure compatibility between existing and new concrete. Analysis would have to be performed to ensure that new drywell structure is as good as, or better than, the existing design. Details of the cost estimate are as follows:

$50,000	civil modification and 10 CFR 50.59 review
$120,000	3-D analysis of new drywell structure
$160,000	miscellaneous material and labor
$100,000	adder for safety-related modification
$50,000	project management and support
$33,000	mobilization, tools and training
$165,000	contingency for lack of detailed estimate
$100,000	contingency for budget estimate
$66,000	construction management fee, insurance, and performance bond

NUREG-1437, Supplement 26, indicates that the estimated cost of providing a means of automatically preventing drain down of the condensate storage tank to the hot well during an SBO was $230,000 at Monticello. Although the Monticello report does not indicate exactly what this modification would entail and the details of the modification for CNS have not been postulated, it would require a hardware modification. The benefit (with uncertainty) of CNS SAMA 72 is only $62,671. As described in ER Section E.2.3, based on a review of previous SAMA evaluations and an evaluation of the expected implementation costs at CNS, hardware modifications cost from $100K to > $1000K. Since the benefit is less than $100K, a detailed site-specific cost estimate was not needed to determine that SAMA 72 is not cost-beneficial.

Analysis Case 37 in ER Section E.2.3 describes the modification for SAMA 73. This modification entails installation of a chiller in the yard with the associated piping and valves and includes penetrations through the reactor building. A heat exchanger would be required in the reactor building to transfer heat from the torus water to the chiller water. Pipe penetrations would be needed on the torus to allow for water circulation to the heat exchanger. Procedure changes and training would also be required. Details of the cost estimate are as follows:

$70,000	multi-discipline modification and 10 CFR 50.59 review
$40,000	pipe sizing and seismic calculations
$25,000	one new and eight revised procedures
$100,000	commercial grade 150 ton chiller
$250,000	nuclear grade heat exchanger
$40,000	piping and supports
$30,000	valves
$40,000	two penetrations to allow chiller water circulation to and from the torus
$30,000	instrumentation and controls
$30,000	electrical cables and conduits
$100,000	adder for safety-related modification
$70,000	project management and support

$50,000	mobilization, tools and training
$200,000	contingency for lack of detailed estimate
$100,000	contingency for budget estimate
$100,000	construction management fee, insurance, and performance bond

SAMAs 76, 77, and 80 were assumed to cost > $100K since they are hardware modifications. The details of the modifications have not been established. As described in ER Section E.2.3, based on a review of previous SAMA evaluations and an evaluation of the expected implementation costs at CNS, hardware modifications cost from $100K to > $1000K. Since the benefit (with uncertainty) for SAMAs 76, 77, and 80 is less that $100K, the conclusion that these SAMAs are not cost beneficial was made without detailed modification descriptions or detailed cost estimates.

"Group 1 isolations" refers to the primary containment isolation system (PCIS) actions to close the MSIVs and main steam line drain valves. Groups 2 through 7 of PCIS actuate to close other valves such as residual heat removal (RHR), reactor water cleanup, HPCI and RCIC steam lines, and sample valves. Each isolation group is actuated by different isolation signals within PCIS. A Group 1 isolation occurs on low reactor water level, low main steamline pressure , high main steamline area leak detection temperature, low condenser vacuum, or high main steamline flow. The PRA models Group 1 isolation as an initiator which causes closure of the MSIVs, resulting in loss of the main condenser as a heat sink.

SAMA-76 is proposed to improve steam tunnel HVAC reliability. The main steamline leak detection system is composed of temperature switches, some of which are dispersed near the steam lines in the steam tunnel providing assurance that a significant break will be detected rapidly and accurately. However, since PCIS will cause a Group 1 isolation if the temperature in the steam tunnel is > 200°F, failure of steam tunnel HVAC can result in a Group 1 isolation, and closure of the MSIVs, although a main steamline break has not occurred.

6.b.i. Phase II SAMA 14 would provide alternate DC feeds (using a portable generator) to panels supplied by a DC bus. Upon loss of a DC bus, a portable generator could be used to provide power to an individual 125 VDC motor control center. Analysis case 14, which estimates the benefit of setting HPCI unavailability to zero, was selected to conservatively assess the benefit of SAMA 14 because the benefit of providing an alternate DC source to HPCI to support emergency core cooling was judged to be larger than the benefit of providing an alternate DC source to other panels. This is due to the importance of HPCI in intermediate LOCA sequences and because the turbine-driven HPCI pump can continue to run in SBO sequences as long as DC control power is available.

6.b.ii. Event HCI-SYS-TM-HPCI (HPCI unavailable due to test and maintenance) has an RRW equal to 1.03. However, in analysis case 14, HPCI was set to never fail (HPCI top gates were set to zero). Thus, analysis case 14 conservatively results in a 32% reduction in CDF.

Although importance measures for HPCI unavailability have not been quantified, loss of DC power is a significant contributor to HPCI unavailability. Table E.1-3 shows that other significant HPCI failure contributors are failure of the turbine-driven pump to start (HCI-TDP-SS-TP), the test and maintenance event (HCI-SYS-TM-HPCI), failure of hydraulic valve HO10 (HCI-HOV-CC-HO10), failure of the turbine-driven pump to continue to run (HCI-TDP-SR-TP24) and failure to bypass the high temperature trip (HCI-XHE-FO-BYPTP).

Since the implementation cost is not much larger than the benefit (with uncertainty) for Phase II SAMA 14, this SAMA might not be cost-beneficial if the analysis was refined to only fail the DC power contributor to HPCI unavailability. However, the SAMA was conservatively retained on the list of SAMAs for further evaluation.

6.c. The Table E.2-2 description of the modeling assumption for SAMA 6 has been revised to state "Failure to transfer the RPS panels to their alternate power source was set to zero in the level 1 PSA model" (see Attachment 2). No revised evaluation is necessary.

6.d. For SAMA analysis case 23, failure probabilities for operator actions related to injection via the fire water system were reduced by a factor of 2. Specifically, failure probabilities for basic events FPS-XHE-FO-DFPAL, FPS-XHE-FO-DISEL, and FPS-XHE-FO-RPVIN were reduced as described below.

<u>FPS-XHE-FO-DFPAL – CREW FAILS TO ALIGN DFP UNDER SBO CONDITION (>2 HRS. AVAIL)</u>
The failure probability for this operator action was reduced to 5.00E-02 from its original value of 1.00E-01. The manipulation time for this action could be reduced via training. The response time may also be reduced slightly due to heightened awareness provided by training. A factor of two improvement bounds the expected decrease in failure probability attainable by decreasing the response and manipulation time by increasing training on this action.

<u>FPS-XHE-FO-DISEL – NO FUEL OIL MAKEUP PROVIDED WITHIN 8 HOURS</u>
The failure probability for this operator action was reduced to 9.50E-03 from its original value of 1.90E-02. The manipulation time for this action could be reduced via training. The response time may also be reduced slightly due to heightened awareness provided by training. A factor of two improvement bounds the expected decrease in failure

probability attainable by decreasing the response and manipulation time by increasing training on this action.

FPS-XHE-FO-RPVIN - OPERATOR FAILS TO ALIGN FIRE PROTECTION SYSTEM FOR RHR LOOP A INJECTION
The failure probability for this operator action was reduced to 5.00E-02 from its original value of 1.00E-01. The manipulation time for this operator action could be reduced via training. The response time may also be reduced slightly due to heightened awareness provided by training. A factor of two improvement bounds the expected decrease in failure probability attainable by decreasing the response and manipulation time by increasing training on this action.

Operator actions FPS-XHE-FO-DFPAL1, "CREW FAILS TO ALIGN DFP UNDER SBO CONDITION (<30 MIN. AVAIL)," and FPS-XHE-FO-RHR25A, "OPERATOR FAILS TO MANUALLY OPEN RHR-MO-25A LOCALLY," were originally assigned a failure probability of 1.0 because they cannot be performed within the available time. The failure probability for these events was not reduced in the analysis for SAMA 78 because increased training is not expected to reduce manipulation or response times such that these actions can be performed within the available time. Since SAMA 78 is potentially cost-beneficial with the existing analysis, altering this assumption would not change the SAMA results.

6.e. SAMA 41 is intended to increase availability of instrument air after a loss of offsite power. Station air compressor (SAC) A is powered from critical 480V bus 1F, SAC B is powered from critical 480V bus 1G, and SAC C is powered from a non-critical bus. Thus, only SACs A and B will receive diesel power following a loss of offsite power.

Since current design does not allow for power connections between SAC C and either diesel generator (DG)1 or DG2, implementation of SAMA 41 would require electrical, mechanical, and structural hardware modifications to power SAC C from diesel power following a loss of offsite power. The cost estimate of $1,200,000 for this SAMA is based on an estimate for a similar SAMA for James A. FitzPatrick Nuclear Power Plant.

Although procedure changes would also be required to utilize this additional power connection, the description of SAMA 41 in the Tables E.2-2 and E.2-3 has been clarified (see Attachment 2).

6.f. Without a complete fire PRA, complete seismic PRA, and detailed evaluation of seismic-fire interactions, the exact benefit from SAMA 69 for external events cannot be characterized. However, in accordance with the guidelines in NEI 05-01, an external events multiplier has been used to ensure that SAMA benefits account for the benefits of both internal and external events. The external events multiplier for CNS is three,

because the external event CDF contribution was estimated to be twice that of the internal events.

In the internal events PRA, the diesel fire pump is credited as an alternate source of water for injection to prevent core damage. Following a fire or seismic event, the diesel fire pump might also be needed to combat a fire, but its contribution to core damage prevention would come from its use as an alternate source of water for injection. Although this particular SAMA would not increase the availability of the diesel fire pump following an internal event initiator, the benefit of eliminating failure of the diesel-driven fire pump can be, and was, assessed using the internal events model. Since the external event CDF contribution is twice that of the internal events, and SAMA 69 only provides external event benefit, doubling the internal events benefit would provide an appropriate estimate of the benefit of SAMA 69. However, the external events multiplier of three was used for ease and consistency with the other SAMAs.

In summary, since it provides a conservative assessment, the benefit estimate for SAMA 69 is appropriate.

As described in ER Section E.2.3, based on a review of previous SAMA evaluations and an evaluation of expected implementation costs at CNS, hardware modifications were estimated to cost from $100K to > $1000K. Since SAMA 69 would entail a hardware modification of the fuel and water tank structural supports, the implementation cost estimate of > $100k is appropriate.

6.g As indicated in the response to Question 5.f., the available skid mounted portable power supply considered in the cost estimates for SAMAs 14 and 20 is not sufficient to supply the battery chargers as proposed in SAMA 13. Since the SAMA submittal, NPPD has designed, installed and placed in service a new diesel generator that can supply the swing battery charger during an SBO at CNS. Therefore, SAMA 13 has been implemented at CNS.

6.h. The "CDF Reduction" value for each SAMA in Table E.2-2 is the reduction in release frequency obtained by implementation of the SAMA. SAMA 70 reduces the release frequency by providing a barrier between the reactor vessel and the drywell shell which will prevent core debris from contacting the drywell shell, but it does not reduce the Level 1 CDF. Therefore, events in Table E.1-3 are not impacted by SAMA 70. Several events in Table E.1-5 are impacted by SAMA 70.

6.i. (1) The insights gained from the CNS PRA model have identified important systems and components that, if maintained at a high level of reliability, provide assurance that core damage is avoided. NPPD identified those insights by identifying those components whose failure or unavailability would contribute the most to core

damage. Similarly, the Generation Risk Assessment (GRA) relies on a detailed logic model, but its objective is to determine which components are most likely to cause a plant trip or to result in a manual trip. Insights from the GRA model assist in identifying maintenance activities and operational practices that will avoid the most likely causes of plant trips. At CNS, the GRA model is part of an overall Equipment Reliability Excellence Plan. The initial phase, which is complete, was to perform single point vulnerability (SPV) studies on 19 systems to identify components that, if failed, would lead to plant trip. This information can be used to re-evaluate maintenance activities and design dependencies for these systems to reduce the likelihood of a failure that leads to a plant trip. By identifying these SPVs and addressing the insights gained in this manner, the likelihood of plant trips occurring from normally running support systems will be reduced.

(2) The GRA program was piloted by CNS for the Electric Power Research Institute (EPRI) and the results were presented to the industry on numerous occasions, including EPRI's Equipment Reliability Forum and the American Nuclear Society International PSA Conference. The overview of the methodology was made available publically through these venues. The technical details of CNS system-specific equipment performance and loss generation figures of merit are published in EPRI literature, but the material remains subject to copyright laws.

The analysis assumptions made for the GRA are too numerous to be presented here, but are well documented in several EPRI reports including the following:

- Introduction to Simplified Generation Risk Assessment Modeling, TR1007386, Final Report, January 2004.
- Generation Risk Assessment (GRA) at Cooper Nuclear Station, TR1011924, Final Report, December 2005
- Pilot Application of Enterprise Project Prioritization Process at Nebraska Public Power District (NPPD)1012954, Final Report, March 2006
- Comparison of Qualitative (AP-913) and Quantitative (Generation Risk Assessment) Equipment Reliability Assessment Technique, TR1013575, Technical Update, December 2006.

The GRA model is a decision-making tool that focuses resources on achieving safe and reliable plant operation while dealing specifically with uncertainties associated with projecting equipment performance. The method specifically accounts for statistical variability and is part of the decision-making process. The GRA uses state-of-the-art industry accepted risk assessment tools to improve plant performance by further reducing scrams while ensuring nuclear safety at the plant.

(3) A bounding analysis of the benefit of fully implementing GRA was performed for SAMA 75 by reducing initiating event frequencies since the objective of the GRA is to reduce the likelihood of failures that lead to plant trip. The conservative factor of 2 reduction in initiating event frequency is based on recent operating history and expert opinion given NPPD's expertise in the GRA process.

(4) The estimated implementation cost is a best-estimate of the costs associated with the GRA based on NPPD experience to date. A detailed implementation cost estimate has not been developed.

6.j. The cost estimates for SAMA implementation did not include maintenance and surveillance costs. However contingency costs for unforeseen implementation obstacles were included.

Cost estimates for SAMA consideration followed Entergy's standard process for development of project estimates. The process is applied to establish conceptual (+/- 25% to 50% accuracy), preliminary (+/- 15% to 30% accuracy), and definitive (+/- 10% to 20% accuracy) estimates during the study, design, and implementation phases of a design project.

The SAMA cost estimates capture anticipated expenses by identifying the parts of the organization that must support the proposed SAMA modification from the conceptual perspective. Typical expenses associated with project cost estimating include calculations, drawing updates, specification updates, bid evaluations, contract issuance, design package preparation, walkdowns, planning and scheduling, estimating, procurement, configuration management, ALARA, QC/QA, training, simulator, IT, design basis update, construction, multi-discipline and independent review of design concepts and calculations, 10 CFR 50.59 review, USAR update, cost control, contingency, security, procedures, post work testing, and project management and close-out. In addition, the project cost estimates include corporate indirect charges.

In summary, the cost estimates for the subject SAMAs followed Entergy's standard process for development of project estimates. Therefore, these cost estimates are reasonable conceptual level estimates.

6.k. The summary report for the PRA model provides the following CDF values:

5th percentile CDF = 4.41E-06
Mean CDF = 9.63E-06
95th percentile CDF = 1.79E-05

Thus, as indicated in the ER, the ratio of the 95th percentile CDF to the mean CDF is about 1.86. However, the previous revision of the summary report indicated that this ratio was approximately 2.9. When the revised values were published, the uncertainty factor of 3 was retained in the SAMA analysis to prevent the necessity for rework.

Since the potential impact of using of a larger uncertainty factor would be to retain more SAMAs as cost-beneficial, the factor of 3 is conservative.

6.1.i. The potentially cost-beneficial SAMAs do not relate to adequately managing the effects of aging during the period of extended operation; therefore, they need not be implemented as part of license renewal pursuant to 10 CFR Part 54. However, as indicated on page 4-85 of the ER, detailed engineering project cost-benefit analyses were initiated for the 11 potentially cost-beneficial SAMAs. Additional potentially cost-beneficial SAMAs that may emerge from further analyses in response to RAIs would be treated in the same manner.

6.1.ii. (1) Of the 11 potentially cost-beneficial SAMAs, those that have the most cost-effective risk reduction potential are 30, 33, 68, and 79, based on their potential for significant risk reduction and relatively low implementation cost (cost estimate is less than 10% of the benefit with uncertainty).

SAMAs 14, 45, 75, and 78 would have second priority based on their potential for risk reduction and their mitigation of plant risk contributors not addressed by SAMAs 30, 33, 68, or 79.

(2) The benefit of the remaining potentially cost-beneficial SAMAs (25, 40, and 64) is expected to be reduced significantly if the higher priority SAMAs are implemented.

(3) As indicated in Item (2) above, the benefit of SAMAs 25, 40, and 64 is expected to be reduced significantly, although a reanalysis has not been performed that would establish that these SAMAs would be no longer cost-beneficial if the higher priority SAMAs were implemented. Regarding specific plans or commitments to implement the high priority SAMAs, refer to the response to 6.1.i. above.

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7. *For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER, would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be*

potentially cost-beneficial at other plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at CNS:

a. Provide additional space cooling to the residual heat removal service water (RHRSW) booster pump rooms, CS pump rooms, residual heat removal pump rooms, service water pump rooms, and HPCI pump room via the use of portable equipment (in lieu of a redundant train of RHRSW booster pump room ventilation considered in SAMA 35).

b. Improve alternate shutdown training and equipment (in lieu of upgrading the ASDS panel considered in SAMA 65). The intent of this alternative is to reduce the human error probability of required actions by improving training on operating the plant from outside the control room and improving communications equipment and plans for coordination among local operators (see Brunswick Phase II SAMA 31).

c. Enhance dc power availability (provide cables from diesel generators or another source to directly power battery chargers).

d. Develop guidance/procedures for local, manual control of reactor core isolation cooling following loss of dc power.

e. Manual venting of containment using either a local hand wheel or gas bottle supplies (considered for Nine Mile Point Unit 1) as a possible alternative for containment pressure control.

NPPD Response:

As described in ER Section E.2.1, the Phase I list of SAMA candidates included those presented in NEI 05-01, those evaluated by other BWRs, those identified in the CNS Individual Plant Evaluation (IPE) and IPEEE, and those postulated to address risk-significant terms in the current CNS PSA model. Thus, a comprehensive effort was made to identify lower-cost alternatives to those Phase II SAMAs considered in the ER. However, many lower-cost alternatives have already been implemented at CNS and therefore, were not included in the Phase II cost-benefit analysis.

7.a. NPPD has implemented several lower-cost alternatives to provide additional space cooling to the CS pump rooms, RHR pump rooms, RHRSW booster pump room, SW pump room, and HPCI pump room, should the normal ventilation system fail. The specific actions vary depending on the particular room, but include managing heat loads, removing equipment hatches or plugs, opening doors, and using portable fans and ducting

as necessary to prevent damage to the equipment in the room. Therefore, the proposed SAMA has already been implemented at CNS.

7.b. Brunswick Phase II SAMA 31, to improve alternate shutdown training and communication equipment and plans for coordination among operators at local controls, was evaluated as Phase I SAMA 192. This SAMA has already been implemented at CNS because CNS has detailed procedures for alternate shutdown, which include use of communication equipment and coordination among operators. CNS operators are trained in these procedures.

7.c. Phase II SAMA 13 evaluated the use of a portable generator to supply DC power to the battery chargers. As indicated in the response to Question 6.g., NPPD has designed, installed and revised procedures to use a new diesel generator that can supply the swing battery charger during an SBO at CNS. Therefore, SAMA 13 has been implemented at CNS.

7.d. Phase I SAMA 79 was evaluated for developing guidance to allow local, manual control of RCIC operation. Since procedural guidance exists to allow local, manual control of RCIC operation, this SAMA has already been implemented at CNS.

7e. Phase I SAMAs 63 and 151 were evaluated to improve manual venting of containment. Phase I SAMA 63 evaluated revising procedures to allow manual initiation of emergency depressurization. Phase I SAMA 151 evaluated enabling manual operation of containment vent valves via local controls. Both Phase I SAMAs 63 and 151 have already been implemented at CNS.

Attachment 2

Changes to the License Renewal Application Related to
Environmental Report SAMA RAI Responses
Cooper Nuclear Station, Docket No. 50-298, DPR-46

This attachment provides changes to the License Renewal Application based on the responses to the RAIs provided in Attachment 1. The changes are presented in underline/strikeout format.

1. Section E.2.2 (Qualitative Screening of SAMA Candidates (Phase I) states:

"The purpose of the preliminary SAMA screening was to identify the subset of candidate SAMAs that would reduce severe accident risk at CNS and would therefore warrant a detailed cost-benefit evaluation. Potential SAMA candidates were screened out if they modified features not applicable to CNS, if they had already been implemented at CNS, or if they were similar in nature and could be combined with another SAMA candidate to develop a more comprehensive or plant-specific SAMA candidate. During this process, 49 of the Phase I SAMA candidates were screened out because they were not applicable to CNS, 24 of the Phase I SAMA candidates were screened out because they were similar in nature and could be combined with another SAMA candidate, and 91 of the Phase I SAMA candidates were screened out because they had already been implemented at CNS, leaving 80 SAMA candidates for further analysis. The final screening process involved identifying and eliminating those items whose implementation cost would exceed their benefit as described below. Table E.2-2 provides a description of each of the 80 Phase II SAMA candidates."

This paragraph is revised to read:

"The purpose of the preliminary SAMA screening was to identify the subset of candidate SAMAs that would reduce severe accident risk at CNS and would therefore warrant a detailed cost-benefit evaluation. Potential SAMA candidates were screened out if they modified features not applicable to CNS, if they had already been implemented at CNS, or if they were similar in nature and could be combined with another SAMA candidate to develop a more comprehensive or plant-specific SAMA candidate. During this process, 49 of the Phase I SAMA candidates were screened out because they were not applicable to CNS. Of these 49 SAMAs, three SAMAs (209, 227, and 229) had already been analyzed for CNS and were considered not applicable due to low benefit. Also, five SAMAs (217, 220, 221, 224, and 226) were seismic outliers that were resolved during the A-46 program. These SAMAs were included to show that IPEEE vintage improvements had been addressed. However, since the outliers have been resolved, the suggested improvements are no longer applicable to CNS. ~~, 24~~ Twenty-four of the Phase I SAMA candidates were screened out because they were similar in nature and could be combined

with another SAMA candidate, and 91 of the Phase I SAMA candidates were screened out because they had already been implemented at CNS, leaving 80 SAMA candidates for further analysis. The final screening process involved identifying and eliminating those items whose implementation cost would exceed their benefit as described below. Table E.2-2 provides a description of each of the 80 Phase II SAMA candidates."

Reference: Response to RAI ENV-SAMA-5.e.

2. Table E.2-2, Assumptions column for SAMA 6 on Page E.2-31 states:

"The time available to recover offsite power before HPCI and RCIC are lost was changed to 24 hours during station blackout scenarios in the level 1PSA model."

This is revised to read:

"~~The time available to recover offsite power before HPCI and RCIC are lost was changed to 24 hours during station blackout scenarios in the level 1PSA model.~~ Failure to transfer the RPS panels to their alternate power source was set to zero in the level 1 PSA model."

Reference: Response to RAI ENV-SAMA-6.c.

3. Tables E.2-2 and E.2-3, left most columns on Page E.2-49 and E.2-74, respectively, state:

"41- Modify procedure to provide ability to align diesel power to more air compressors."

This is revised to read:

"41- ~~Modify procedure to~~ Provide ability to align diesel power to more air compressors."

Reference: Response to RAI ENV-SAMA-6.e.