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May 17, 2004

AEP:NRC:4034-04
10 CFR 51.53(c)

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
Response to Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) Regarding Severe Accident Mitigation Alternatives for the Donald C. Cook Nuclear Plant Units 1 and 2 (TAC Nos. MC1221 and MC1222)

REFERENCE: Letter from Robert G. Schaaf, U. S. NRC, to Mano K. Nazar, Indiana Michigan Power Company (I&M), "Request for Additional Information (RAI) Regarding Severe Accident Mitigation Alternatives for the Donald C. Cook Nuclear Plant, Units 1 and 2," dated March 18, 2004.

Dear Sir or Madam:

By letter dated October 31, 2003, I&M submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. Appendix E to this license renewal application contained the Applicant's Environmental Report - Operating License Renewal Stage, which is also referred to as the ER. Section 4.20 and Appendix F of the ER included an analysis of CNP Units 1 and 2 Severe Accident Mitigation Alternatives (SAMA) candidates.

During the conduct of its review, the NRC Staff identified areas where additional information is needed to complete its review of the CNP SAMA analysis as documented in the above Reference. This letter responds to those RAIs.

Enclosure 1 to this letter provides an affirmation pertaining to the statements made in this letter. Attachment 1 provides the additional information requested by the NRC Staff. Attachments 2 and 3 provide reference tables and figures, respectively, referred to in Attachment 1. There are no new commitments contained in this submittal.

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Should you have any questions, please contact Mr. Richard J. Grumbir, Project Manager, License Renewal, at (269) 697-5141.

Sincerely,

M. K. Nazar
Senior Vice President and Chief Nuclear Officer

GEW/rdw

Enclosure:

1 Affirmation

Attachments:

1 Response to Request for Additional Information (RAI) Regarding Analysis of Severe Accident Mitigation Alternatives (SAMAs) for the Donald C. Cook Nuclear Plant (CNP)
2 Reference Tables for Response to Request for Additional Information
3 Reference Figures for Response to Request for Additional Information

c: J. L. Caldwell, NRC Region III
K. D. Curry, AEP Ft. Wayne
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J. G. Lamb, NRC Washington DC
MDEQ – WHMD/HWRPS
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R. G. Schaaf, NRC Washington DC

Enclosure 1 to AEP:NRC:4034-04

AFFIRMATION

I, Mano K. Nazar, being duly sworn, state that I am Senior Vice President and Chief Nuclear Officer of American Electric Power Service Corporation and Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

American Electric Power Service Corporation

M. K. Nazar
Senior Vice President and Chief Nuclear Officer

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 17th DAY OF May, 2004

Notary Public

My Commission Expires 6/10/2007

BRIDGET TAYLOR
Notary Public, Berrien County, MI
My Commission Expires Jun. 10, 2007

Response to Request for Additional Information (RAI) Regarding Analysis of Severe Accident Mitigation Alternatives (SAMAs) for the Donald C. Cook Nuclear Plant (CNP)

By letter dated October 31, 2003, Indiana Michigan Power Company (I&M) submitted an application to renew the operating licenses for CNP Units 1 and 2. Appendix E to this license renewal application contained the Applicant's Environmental Report - Operating License Renewal Stage, which is also referred to as the ER. Section 4.20 and Appendix F of the ER included an analysis of CNP Units 1 and 2 SAMA candidates. By letter dated March 18, 2004, the Nuclear Regulatory Commission (NRC) identified areas where additional information is needed to complete its review of the CNP SAMA analysis. This attachment provides the additional information requested by the NRC. Attachments 2 and 3 provide the tables and figures, respectively, that are referred to in this attachment.

SAMA RAI 1:

The Severe Accident Mitigation Alternatives (SAMA) analysis is based on the most recent version of the CNP Probabilistic Risk Assessment (PRA) for internal events (i.e., August 2001 Level 1 model and October 2003 Level 2 model), which is a modification to the revised individual plant examination (IPE) submittal transmitted to the U.S. Nuclear Regulatory Commission (NRC) in October 1995. Please provide the following information regarding this PRA model:

a. *A description and the results of the internal and external peer reviews of the Level 2 and 3 portions of the PRA that have been performed since the IPE. This should include a description of the internal and external peer reviews of the MELCOR Accident Consequences Code System (MACCS2) and MAAP analyses (see also SECY-03-0222).*

b. *An assessment of the impact of the weaknesses/areas for improvement identified in the Westinghouse Owners Group (WOG) peer review on the SAMA identification and evaluation process, (e.g., since one identified weakness is common cause analysis, how would SAMAs be impacted by improving the common cause analysis?).*

c. *A description of the major differences from the Revised IPE submittal, Level 1 PRA modeling changes that have resulted in the new core damage frequency (CDF) and large early release frequency (LERF). Also provide the revised importance measures for basic events (both risk increase and risk decrease), and the importance measures formulations. Please include a discussion of the reasons for the difference in the station blackout (SBO) fraction of CDF from the IPE (1.8%) and current reported value (22.8%) in Table F.2-2.*

d. *A description of the changes in the PRA Level 2 methodology since the IPE submittal, including major modeling assumptions, plant response tree (PRT)/containment event tree (CET) structure. Please confirm that the large early release frequency (LERF) values in*

the Indiana Michigan Power Company's (I&M) SAMA analyses are based on the October 2003 Level 2 PRA update.

e. *A description of the methodology and criteria for binning endstates into the 8 accident sequences/release categories shown in Table F.2-6 and used in the current Level 3 analysis.*

f. *The specific source terms used to represent each of the 8 accident sequence/release categories, and a containment matrix describing the mapping of Level 3 results into the various accident sequences/release categories.*

g. *A description of the accident sequences used to represent each of the 8 accident sequences/release categories shown in Table F.2-6, and how each sequence was chosen to represent a bin.*

h. *A breakdown of the population dose (person-rem per year within 50 miles) by containment release mode, such as steam generator tube rupture (SGTR), interfacing systems loss-of-coolant accident (ISLOCA), containment isolation failure, early containment failure, late containment failure, and no containment failure.*

CNP Response to SAMA RAI 1:

a. The current CNP Units 1 and 2 Level 2 PRA models are a significant revision of the IPE models, and were prepared by an outside consulting firm that has significant experience with issues surrounding Level 2 PRA development. As part of the Level 2 PRA model development, the analysis files were prepared under the consultant's quality assurance (QA) program and subjected to an independent review prior to release to I&M. The Level 2 PRA models were further subjected to an extensive owner acceptance review by I&M. The source term analyses performed in September 2003 as part of the CNP SAMA analysis were based on the Modular Accident Analysis Program (MAAP) computer code model used for performing the most recent CNP containment recirculation sump inventory analyses. The CNP containment recirculation sump inventory analyses were provided to the NRC for review by letter from R. P. Powers (I&M) to J. F. Stang, Jr. (NRC), "Technical Specification Change Request - Containment Recirculation Sump Water Inventory," dated October 1, 1999. NRC review and approval of the CNP containment recirculation sump inventory analyses was documented in the NRC Safety Evaluation Report issued on December 13, 1999, for CNP Unit 1 Amendment 234 and CNP Unit 2 Amendment 217.

The MAAP computer code model was updated as part of the 2003 SAMA analysis to reflect plant changes that had occurred since the submittal of the CNP containment recirculation sump inventory analyses, and to incorporate best-estimate initial conditions and boundary conditions. The MAAP computer code model changes are documented in

evaluations that were prepared, reviewed, and approved in accordance with the governing CNP design control procedures.

The CNP Level 3 PRA and MACCS2 computer code models were prepared by an outside consulting firm that has significant experience with development of Level 3 PRA models using the MACCS2 computer code. As part of the Level 3 PRA model development, the analysis files were prepared and reviewed under the consultant's QA program and subjected to a review by I&M.

b. The WOG Peer Review of the CNP Units 1 and 2 Level 1 PRA model was performed in September 2001 and the final report issued in December 2002. The report contained a total of three Level A Facts and Observations (F&Os) and 24 Level B F&Os. A project to resolve Level A and Level B F&Os, including the resolution of F&Os related to common cause, was completed in April 2004. All Level A and Level B F&Os were addressed with the exception of one Level A finding that identified weaknesses in the current flooding analysis, which was completed in conjunction with the IPE. Internal flooding events contributed 2.0E-07, or less than 1% to overall CDF in the IPE analysis, and it is expected that resolution of the F&O related to internal flooding would not result in a significant change in the contribution to CDF. Therefore, it is concluded that the SAMA analysis would not be changed with the resolution of this F&O.

Calculations using the upgraded model result in a total CDF of 4.3E-05 per year, about 10% less than the value of 4.9E-05 per year from the August 2001 Level 1 PRA model used in the CNP SAMA analysis. Overall, the distribution of events that lead to core damage has changed only slightly. With the upgraded model, 74% of CDF is from sequences involving a loss of reactor coolant pump (RCP) seal cooling, similar to the 71% contribution from such sequences in the August 2001 model used for the CNP SAMA analysis. Sequences that involve loss of RCP seal cooling include loss of all essential service water (ESW) to both units, loss of ESW to a single unit, loss of component cooling water (CCW), and SBO events.

As shown in Section F.2.3.5 of the ER, about 40% of the maximum benefit for the base case evaluation are from onsite costs, driven almost entirely by onsite economic costs, and 60% of the maximum benefits are from offsite costs. As shown in Sections F.2.3.3 and F.2.3.4 of the ER, onsite costs change linearly with changes in CDF. Therefore, use of the recently upgraded model would result in lower maximum benefit for the base case evaluation, and the lower CDF would result in lower onsite costs.

Offsite costs, consisting of offsite radiological exposure costs and offsite economic costs, are a function of both CDF and radionuclide release. Overall distribution of events that contribute to CDF has changed only slightly, and the contribution of loss of RCP seal sequences is about the same for the upgraded model as for the August 2001 model used

for the CNP SAMA analysis. Therefore, it is expected that offsite costs would not be significantly different if the upgraded model was used for the CNP SAMA analysis.

One source used to identify the plant-specific SAMA candidates presented in Table F.4-1 of the ER was the basic event importance measures from the August 2001 PRA model. All basic events with a contribution to CDF of greater than 0.5% were reviewed in the process of identifying SAMA candidates. The basic event importance measures from the upgraded model, while very similar, are not identical to those from the August 2001 PRA model. Therefore, basic event importance measures from the upgraded model were reviewed. As a result of the review, one additional plant-specific SAMA candidate would have been identified if the new model had been used. This new candidate is related to electrical switchgear room ventilation, and could be grouped with SAMA Numbers 25, 26, and 27 in Table F.4-1 of the ER. As shown in Table F.4-2 and discussed in Section F.7 of the ER, these SAMA candidates were considered potentially cost-beneficial for mitigating the consequences of a severe accident using the August 2001 PRA model. Therefore, it is expected that the conclusions would be the same had the upgraded model been used in the CNP SAMA analysis.

c. Three revisions have been made to the CNP PRA model since the revised IPE was submitted to the NRC in October 1995. The first, completed in May 1996, was a minor update involving test and maintenance unavailability. In the May 1996 update, the quantification process was changed to eliminate cutsets that included basic events representing maintenance on opposite trains because maintenance on opposite trains is precluded by the CNP Technical Specifications, and because such cutsets are eliminated by convention in PRA analyses. The update also ensured that test and maintenance data from Revision 1 were used when data from both Revision 0 and Revision 1 were available. Lastly, maintenance unavailability was added for control air system dryers. These changes resulted in a reduction of CDF from 7.14E-05 per year to 6.36E-05 per year. LERF was not calculated for this model.

The second revision, completed in August 1997, converted the logic models from the original GRAFTER computer code to the Computer Aided Fault Tree Analysis (CAFTA) computer code, and increased CDF from 6.36E-05 per year to 7.09E-05 per year. LERF was not calculated for this model. The increase in CDF was a result of using truncation limits for the revised model that were lower than those used for the previous model. Software limitations of the GRAFTER computer code precluded using the lower truncation limits.

The 2001 update to the CNP PRA model was a major update that incorporated changes to the design and operation of the plant implemented since completion of the IPE. The overall purpose of the update was to develop a PRA model to support compliance with 10 CFR 50.65 Section (a)(4) for management of risk during maintenance activities, and to support the new risk-informed, performance-based regulatory environment.

The following general changes were incorporated in the 2001 PRA model update:

- The existing CAFTA computer code top logic model was converted to a linked fault tree WinNUPRA computer code model to better support implementation of Safety Monitor for on-line and shutdown risk evaluation.

- The PRA was updated to include new plant-specific data, procedure and/or design changes, revision of the treatment of common cause failures to comply with the latest methodology, and removal of conservative assumptions and simplifications.

- The IPE was a single unit model and applied only to an operating unit. The 2001 PRA model update created a dual unit model including inter-unit dependencies and spanned all modes of operation (operating and shutdown). This effort included the development of Safety Monitor full power models based on the updated PRA, and development of a shutdown risk model, which can be used to support assessment and management of shutdown risk.

The following specific changes were incorporated in the 2001 PRA model update:

- Initiating events were modified as follows:

  - Large break and medium break loss of coolant accidents (LOCAs), SGTRs, and steam line breaks were subdivided into the individual contributions from each loop, and four separate initiating events were evaluated for each of these categories.

  - Initiators for loss of a single DC electrical power train were added for each train separately.

  - The loss of offsite power (LOSP) initiator was divided into LOSP to a single unit and LOSP to both units (dual unit LOSP) to improve modeling of the unit cross-ties.

  - Similarly, loss of ESW was split to consider the loss of a single unit ESW system separately from a total (dual unit) loss of ESW to improve modeling of the unit cross-ties.

- Initiating event frequencies were reassessed based on updated plant-specific data and new generic data. In addition, a number of the frequencies were obtained from models built into the overall PRA as transfers from other initiators. The initiators included:

  - Consequential medium break and small break LOCAs resulting from a reactor coolant system (RCS) power operated relief valve or safety relief valve failing to reclose.
  - SBOs.
  - Anticipated Transients Without Scram (ATWS) events.

- Also, several initiating event frequencies were obtained from detailed system models:

  - Loss of ESW to a single unit.
  - Loss of ESW to both units.
  - Loss of CCW.
  - Loss of 250 VDC electrical power busses.

• Fault trees were modified as follows:

- The fault tree models were revised to incorporate design changes and operational changes.

- Individual component common cause groups were identified for Multiple Greek Letter method common cause analysis.

- The models were revised to support the implementation of Safety Monitor.

- The heat removal function was removed from the recirculation model, and this function was included in a separate long term cooling model.

- Extensive changes were made in the ESW system model, properly accounting for interactions between units for this shared system.

- The 4160 VAC electrical power system model was changed to address a reconfiguration of the reserve auxiliary transformers.

- Reliability and unavailability data were modified as follows:

  - Revision of component failure data analysis included collecting and analyzing more recent CNP failure data for the time period since the previous update, and the enhancement of common cause failure data for all components.

- Human Reliability Analysis (HRA) was modified as follows:

  - Evaluation of human error probabilities was limited to those operator actions affected by changes in procedures, or to those operator actions that were new to the updated model. The principal change evaluated involved the revised Emergency Operating Procedure for switching to cold leg recirculation.

  - The revised procedure for a loss of CCW was also used to update the associated human error probabilities.

  - The net result was to add or revise 30 human error probabilities (20% of the total human interaction events).

The final results of the 2001 PRA model update include the following:

- The CDF calculated with the updated model is 4.9E-05 per year, which is less than that from the 1995 update of 7.14E-05 per year. This can be attributed to a number of factors including a reduction in LOCA-initiating event frequencies, the removal of conservative assumptions, and the more detailed and complete modeling of ESW cross-ties between units.

- The Unit 2 results are almost identical to those for Unit 1, with the differences being due to minor differences in electrical power supply arrangements to support systems and ATWS unfavorable exposure times.

- The distribution of the contributions to the results with respect to CDF has changed from the 1995 update. The SBO contribution, which includes SBOs initiated by both single unit and dual unit LOSP events, is now about 37% of the total CDF and is higher than the 1995 result. The increase in SBO contribution is caused mainly by two changes from the IPE model. The first cause of the increase in the contribution of SBO events was a change in the definition of a SBO that resulted in increasing the mission time for the emergency diesel generators (EDGs). The second cause that increased the contribution of SBO events was an increase in the failure rate for the EDGs. The IPE model used a failure rate of 9.03E-04 per hour, while the updated model used a value of 4.73E-03 per hour for EDGs failing to run.

- Sequences related to a loss of all ESW contribute approximately 24% of the total CDF. The most significant contributors are loss of ESW, either as the initiator or following a normal transient initiator, with subsequent loss of ESW combined with failure to recover ESW.

- Small break LOCA is still an important contributor (17%) to CDF. The importance of small break LOCA has decreased from the 1995 evaluation due to the reduced initiator frequency. The contribution to the total of SGTRs has been reduced due to more detailed modeling, while the contribution from steamline breaks has gone up because of an increase in assessed secondary side pipe break frequency.

- LERF is calculated to be 5.6E-06 per year using the updated model. LERF was not calculated in the IPE models. The dominant contributors to LERF are LOSP initiated sequences, which make up approximately 50% of the total. SGTRs, loss of ESW, and small break LOCAs each contribute about 10% to the total LERF.

Tables 1 and 2 of Attachment 2 provide the results of the importance analysis of the total CDF and LERF, respectively. Basic events having a Fussell-Vesely importance above 1% are shown in the tables. It should be noted that the basic events that are missing (as indicated by a missing sequential rank number) are items such as complement events and constants, which have no physical meaning but are generated or used as part of the quantification process. As shown in the tables, the value shown in the "Point Estimate" column is the mean failure probability value used in the quantification process. The value shown in the "F-V Importance" column is the Fussell-Vesely importance value of the basic event and is calculated as follows:

$$I_i^{FV} = \frac{\sum_{j=1}^{J} S_{i,j}}{\sum_{k=1}^{K} S_k}$$

Where:

I_i^{FV} = Fussell-Vesely Importance for basic event *i*
$S_{i,j}$ = Probability estimate for cutset j containing event *i*
S_k = Probability estimate for a minimal cutset of the outcome of interest
K = Total number of cutsets in equation
J = Total number of cutsets containing event *i*

The value shown in the "Risk Achievement" column is the Risk Achievement Worth importance value of the basic event, and is calculated as follows:

$$RAW_i = 1 + I_i^{FV} (p_{i.o}^{-1} - 1)$$

Where:

$p_{i.o}^{-1}$ = basic event unavailability for event *i*

The value shown in the "Risk Reduction" column is the Risk Reduction Worth importance value of the basic event and is calculated as follows:

$$RRW_i = (1 - I_i^{FV})^{-1}$$

The importance analysis reflects the results discussed above for contribution to CDF and LERF. That is, failure to recover a loss of ESW is the dominant contribution to core damage and the dominant contribution to LERF is SBO (caused by failure of the EDG to run) and where power is not recovered until core damage results from RCP seal leakage.

d. The IPE Level 2 analysis characterized containment performance for each of the CET end states by assessing containment loading. The CETs were quantified and identified probable containment failure mechanisms. However, some potential containment failure modes were not included in the CET, but were addressed using phenomenology reports prepared by the IPE consultant. In the IPE analysis, phenomenology reports were used to conclude that the following phenomena would not cause containment failure: direct containment heating (DCH), thrust forces at reactor pressure vessel (RPV) failure, in-vessel steam explosions, molten core concrete interactions, thermal degradation of containment penetrations, and hydrogen detonation.

For the October 2003 Level 2 PRA update, a new CET, shown in Figure 1 of Attachment 3, was constructed based on the methodology given in NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events." In the updated Level 2 PRA model, the five phenomena discussed above that were dismissed in the IPE as incapable of causing containment failure are treated probabilistically.

A discussion of the CET headings is provided below.

- Containment Not Bypassed (C-BYP)

  The top event "Containment Not Bypassed (C-BYP)" questions whether or not the core fission products are released inside an intact containment. Reasons for the failure branch of this node could be an ISLOCA or SGTR initiating event. Containment bypass due to induced SGTR is addressed under the ISGTR heading.

- Steam Line Break Secondary Side Isolation (SLB-SS)

  Top event "Steam Line Break Secondary Side Isolation (SLB-SS)" evaluates if the initiating event was a steam line break and, if so, then asks if the secondary side of the steam generators was successfully isolated. The status of this top event is obtained from the Level 1 event trees.

- Containment Isolated (C-ISOL)

  The top event "Containment Isolated (C-ISOL)" questions whether or not the core fission products are released inside an intact containment. As summarized in the IPE, the most likely single event that will cause containment isolation failure was identified as a failure to close an administratively controlled manually isolated line. This implies that the conditions for isolation failure existed prior to the accident initiation. The failure probability of this event was calculated to be 2.9E-04 in the containment isolation analysis for the IPE and is used for this top event.

- Hydrogen Igniters Operating Before Core Damage (H2-IGN)

  The top event "Hydrogen Igniters Operating before Core Damage (H2-IGN)" questions the success or failure of the hydrogen igniters. The status of hydrogen igniters is obtained from the Level 1 event trees.

- Reactor Coolant System Depressurized (DEPRE)

  The top event "Reactor Coolant System Depressurized (DEPRE)" questions the RCS pressure at the time of RPV failure. The success branch indicates a low RCS pressure (yielding lower stresses on containment), while the failure branch indicates a high RCS pressure at RPV failure. The success or failure of this branch is determined from the Level 1 event trees.

- Core Damage Arrested Before Vessel Breach (CDA-BB)

  The top event "Core Damage Arrested Before Vessel Breach (CDA-BB)" provides the ability to credit a melted core that does not result in a RPV failure. Typically, this might be credited for LOSP sequences, or other sequences in which equipment might be recovered after core melt but before RPV failure. The model assumes that core damage cannot be arrested in-vessel for SGTR, ISLOCA, and transient events. For LOCAs, except SGTR and ISLOCAs, failure to arrest core damage in-vessel is assigned a probability of 1.0 unless RPV injection is successful. If RPV injection is successful for LOCA events, then it is assumed that core damage is arrested in-vessel. The status of RPV injection is determined from the Level 1 event trees.

- No Induced Steam Generator Tube Rupture (ISGTR)

  The top event "No Induced Steam Generator Tube Rupture (ISGTR)" applies to high-pressure sequences in which there is a potential for creep failure of the steam generator tubes, resulting in a containment bypass. Keeping with the simplified CET intended in NUREG/CR-6595, the probabilities identified for an intact primary system and no depressurized steam generators are used for all cases. From NUREG-1570, the conditional probabilities of thermal-induced SGTR and pressure-induced SGTR in this case are 1.73E-02 and 5.49E-02, respectively. Summing the two, the conditional probability for top event six is 7.22E-02, which is rounded up to 8.0E-02 and used for core damage sequences in which auxiliary feedwater (AFW) has failed thereby causing steam generator dryout. For those sequences in which AFW is successful, the probability of induced SGTRs was set to 1.0E-04 to account for uncertainty. The status of AFW is determined from the Level 1 event trees.

- No Containment Failure at Vessel Breach (NCF-VB)

  The top event "No Containment Failure at Vessel Breach (NCF-VB)" provides a conditional probability of containment failure given the response to the previous top events and is based on the analyses presented in NUREG/CR-6595. For this top event, determination of the failure probability is based on RCS pressure and the operability of hydrogen igniters at the time of RPV failure. For low-pressure sequences, the conditional probability of containment failure is 1.0E-02 if the hydrogen igniters are operating, and 1.0E-01 if the hydrogen igniters are not operating. For high-pressure sequences, the conditional probability of containment failure is taken to be 5.0E-02 if the hydrogen igniters are operating, and 2.0E-01 if the hydrogen igniters are not operating. In addition, there is the potential for containment failure due to hydrogen combustion with core damage arrested in the RPV. The assessed conditional probability of containment failure before RPV breach is 0.0 if

the hydrogen igniters are operating, and 4.0E-02 if the hydrogen igniters are not operating. The status of hydrogen igniters and RCS pressure is determined from the Level 1 event trees.

- RWST Injection (RWST-INJ)

  Top event "RWST Injection (RWST-INJ)" determines if injection of the refueling water storage tank (RWST) to the containment is successful prior to RPV failure. RWST injection could either be through the emergency core cooling system (ECCS) or containment sprays and is determined from the Level 1 event trees.

- Containment Spray Injection (CS-INJ)

  Top event "Containment Spray Injection (CS-INJ)" questions whether or not containment spray taking suction from the RWST is successful. This top event does not include long-term cooling by the heat exchangers or containment spray using the residual heat removal (RHR) system. The status of containment spray injection is determined from the Level 1 event trees.

- Containment Spray Recirculation (CS-REC)

  Top event "Containment Spray Recirculation (CS-REC)" questions whether or not containment spray in recirculation mode is successful and includes long-term cooling with the heat exchangers. The containment spray recirculation node accounts for the containment response due to the success or failure of containment heat removal systems. Failure of this node will result in a failure of the containment due to overpressure while success of this node will result in maintaining pressure within the capacity of the containment. The status of containment spray recirculation is determined from the Level 1 event trees.

- Recovery of Containment Heat Removal (CHR-R)

  Top event "Recovery of Containment Heat Removal (CHR-R)" is used to evaluate core damage sequences where core damage occurs following events involving a loss of key support systems, i.e., SBO, loss of ESW, and loss of CCW events. For these events, the Level 1 event trees evaluate whether or not the system failures that caused the event are recovered in sufficient time to prevent core damage. However, if the system failures are not recovered prior to core damage, the Level 1 event trees do not question the status of containment heat removal. NUREG/CR-6427 shows that there would likely be a significant time between core damage and RPV failure and between RPV failure and failure of the containment on overpressure. Recovery of containment heat removal systems after core damage could prevent containment

failure. Recovery of containment heat removal systems is assumed to occur one-half hour after recovery of the system failures that initiated the event. The time available to recover containment heat removal is based on specifics of the accident sequence given in the Level 1 event trees with consideration given to the availability of AFW and RCS cooldown. Time available to recover containment heat removal impacts the recovery probability, which is calculated based on the same failure data used in the Level 1 PRA analysis. These failure probability values are used to quantify the CET.

- Size of Release (RELSIZE)

  Top event "Size of Release (RELSIZE)" addresses the potential size of a release for ISLOCA and SGTR events with consideration given to closure of isolation valves. If the Level 1 event trees show that the steam generators are isolated following a SGTR event, then a small release is assumed, otherwise, a large release is assumed. For ISLOCA events, a small release is assumed if the leak is isolated. Otherwise, a large release is assumed.

All source term category (STC) frequency values used by and calculated in the SAMA analysis, including those STCs that contribute to LERF, are based on the October 2003 Level 2 PRA update that is described above.

e. Table F.2-6 of the ER relates the radionuclide release categories required as input to the Level 3 PRA analysis to the radionuclide release categories for which data was obtained from the Level 2 PRA source term analysis. Table F.2-8 of the ER provides a summary of off-site consequence results based upon eight STCs that are defined by the characteristics provided. Each of these eight STCs is associated with one or more CET endstates, based on a set of binning rules. These binning rules evaluate the containment top events, each of which represents a major possible event in the containment response to an accident sequence. The rules used to determine the CET endstate(s) to be included in each STC are summarized below.

First, the rules determine if the conditions represented by the CET endstate would result in successful containment of the accident, i.e., STC-8. For successful containment, the accident sequence must not be a SGTR or ISLOCA event and containment isolation must be successful. Given that the previous statement is true, requirements for successful containment depend on the status of hydrogen igniters. If the hydrogen igniters were successful, then either containment spray recirculation for heat removal, core damage arrest in-vessel, or recovery of containment heat removal in time to prevent over-pressure failure must be successful. If hydrogen igniters were not successful, then the containment must not have failed at RPV breach (no early containment failure) and either heat removal via containment spray recirculation or recovery of containment heat removal in time to prevent over-pressure failure must be successful. Any CET endstate

that meets these conditions is binned into STC-8. Any endstate that does not meet these conditions is evaluated further.

For the remaining CET endstates, those that are either an ISLOCA or SGTR and result in a small release are binned into STC-1. Those CET endstates that are either an ISLOCA or SGTR and result in a large release are binned into STC-2. Endstates that involve an induced SGTR are also binned into STC-2. Any CET endstate not binned into either STC-1 or STC-2 is then evaluated further.

For the remaining CET endstates, the status of containment isolation is questioned. If containment isolation is failed, the endstate is binned into one of three STCs as follows. First, RWST injection is questioned. If RWST injection failed, then the endstate is binned into STC-4. If RWST injection was successful along with success of both containment spray injection and containment spray recirculation, then the endstate is placed into STC-3. Any remaining CET endstate with failure of containment isolation and success of RWST injection is binned into STC-5. All remaining endstates are evaluated further.

Any remaining CET endstate involving containment failure at RPV breach is binned into STC-6. In addition, any remaining endstate where containment spray injection fails is included in STC-6. Any remaining endstate is evaluated further.

Any CET endstate remaining is binned into STC-7, because the endstate would result in late containment failure due to several reasons. One reason is that the sequence involved relocation of the core debris into a dry reactor cavity, and the containment did not fail at the time of vessel failure. Other endstates involved an initial failure of containment heat removal and failure to recover heat removal prior to containment failure.

f. Table 3 of Attachment 2 provides descriptions of the specific radionuclide release histories used to represent the eight STCs shown in Table F.2-8 of the ER. The fission product groups listed are those used as input to the MACCS2 computer code, and the values given are the fraction of initial inventory that is released over the duration of the associated plume.

The first link between the Level 1 accident sequences and the STCs is the CET. As described in the response to SAMA RAI 1.d, the success or failure of systems and operator actions that impact phenomenology important to containment performance are obtained from the Level 1 event trees. The one exception is containment isolation, which uses the probability of failure taken from the IPE fault tree analysis. Probability values are assigned for various containment failure phenomena based on system availability obtained from the Level 1 event trees. The rules used to bin the CET endstates into each STC are described in the response to SAMA RAI 1.e.

g. Table F.2-8 of the ER provides a summary of off-site consequence results based upon eight STCs. One accident sequence was chosen to represent each of the eight STCs shown in Table F.2-8 of the ER. The sequence chosen to represent each STC was selected using the importance analysis from the Level 2 PRA. Typically, the sequence with the highest importance to the STC was chosen. However, a sequence with a lower importance may have been chosen if it was determined that the sequence would better represent the phenomenology significant to the STC. When such determinations were made, the sequence was still one of the most important to the STC, and it was expected that the source terms that would result from the sequence would not be lower than the most important sequence. A description of each of the analyzed accident sequences is given below.

- STC-1

  The sequence analyzed for this STC is a SGTR, and is analyzed using the Level 1 PRA accident sequence that is most important to the STC. The break area for this event is the equivalent of twice the cross sectional area of a single tube. For this sequence, AFW is successful, but operator actions fail to terminate flow from the ruptured tube. Therefore, overfill of the steam generator occurs. As a result, the relief valve on the faulted steam generator sticks open. Containment spray, ECCS, and hydrogen igniters are successful.

- STC-2

  The sequence analyzed for this STC is an ISLOCA event that occurs in the RHR shutdown cooling suction piping. The sequence analyzed is the second in importance to the STC. However, it was selected to represent the source term because it would provide higher source terms than the dominant sequence. For this STC, the dominant sequence is a small break LOCA that results in an induced SGTR. For such an event, much of the fission product inventory would be initially released to containment so it would be expected that releases would be less than the ISLOCA event. In this sequence, the break size is in the lower range of a large break LOCA. High pressure injection is successful. However, efforts to isolate the break fail. Therefore, all RWST inventory is lost out the break and to the auxiliary building. On depletion of the RWST, injection to the reactor fails and core damage occurs.

- STC-3

  The sequence analyzed for this STC is the Level 1 sequence most important to the STC. This sequence is a small break LOCA with failure of all injection, and is analyzed assuming that the break is in the cold leg with an equivalent diameter of two

inches. Containment spray, containment heat removal, and hydrogen igniters are successful. Containment isolation fails.

- STC-4

  The sequence analyzed for this STC is the Level 1 sequence most important to the STC, a loss of all ESW. The loss of ESW results in failure of all ECCS injection and recirculation capability. AFW and hydrogen igniters are successful. However, RCS inventory loss from the RCP seals results in uncovering the core. Containment isolation fails.

- STC-5

  The sequence analyzed for this STC is the Level 1 sequence most important to the STC, a small break LOCA. AFW, ECCS injection, containment spray injection and hydrogen igniters are successful for this sequence. However, ECCS recirculation and containment spray recirculation fail on depletion of RWST inventory. Containment isolation also fails.

- STC-6

  The sequence analyzed for this STC is an SBO, and is the Level 1 sequence most important to the STC. AFW is successful for this sequence, as are operator actions to depressurize the RCS. Offsite power is not recovered prior to core damage. Consequently, hydrogen igniters are not available. Containment failure occurs at the time of RPV failure.

- STC-7

  The sequence analyzed for this STC is the Level 1 sequence most important to the STC. The Level 1 sequence analyzed is identical to the Level 1 sequence analyzed for STC-6. However, the CET evaluates the sequence by having the containment remain intact when RPV failure occurs.

- STC-8

  The sequence analyzed for this STC is the Level 1 sequence most important to the STC. The Level 1 sequence analyzed is identical to the Level 1 sequence analyzed for STC-4. However, containment isolation is successful for this STC.

h. The following provides a breakdown, by containment release mode, of the population dose risk in person-REM per year within 50 miles of the CNP.

Containment Release Mode	Population Dose (Person-REM per Year)
SGTR	2.70
ISLOCA	2.43
Containment Isolation Failure	7.80E-03
Early Containment Failure	9.60
Late Containment Failure	19.75
No Containment Failure	0.0

SAMA RAI 2:

In Section F.2.1, the CDF for internal events is given as $5.0x10^{-5}$ and the CDF for internal fires and seismic events are given as, $3.8x10^{-6}$ and $3.2x10^{-6}$ respectively. These internal fires and seismic CDFs, and the individual plant examination of external events (IPEEE) models, were not used in the identification and screening of SAMAs. Also, it is not clear whether/how SAMAs to address internal flooding events were considered in the SAMA analysis. In this regard, the following information is needed:

a. NUREG-1742 ("Perspectives Gained from the IPEEE Program", Final Report, April 2002), lists the significant fire area CDFs for CNP (page 3-14 of Volume 2). For each fire area, please explain what measures were taken to further reduce the CDF, and explain why these CDFs cannot be further reduced in a cost effective manner.

b. Please identify those SAMAs from Table F.4-2 that could provide a significant risk benefit in the important seismic, internal fire, and internal flood events at CNP. For each of these SAMAs, provide an estimate of the additional benefit that these SAMAs would provide in the respective events.

c. The SAMA analysis for Catawba Nuclear Station identified a cost beneficial enhancement involving installation of a watertight wall around a 6900/4160V transformer in the turbine building basement (to reduce the risk from flooding events). Please discuss whether a similar modification was evaluated or would be applicable for CNP.

CNP Response to SAMA RAI 2:

a. The fire areas listed on page 3-14 of NUREG-1742, Volume 2, are those that were quantitatively analyzed in the IPEEE and that had a potential CDF greater than 1.0E-07. In the IPEEE submittal, these fire areas were not listed as "significant." Therefore, these

fire areas should not be interpreted as significant despite being included in the listing of Table 3-3 in NUREG-1742. The analysis for each of these areas is documented in the revised PRA for fire events, which was submitted for NRC review by letter from E. E. Fitzpatrick (I&M) to the USNRC Document Control Desk, "Donald C. Cook Nuclear Plants Units 1 and 2 Individual Plant Examination of External Events Response to NRC Audit Concerns and Request for Additional Information," dated February 15, 1995. Areas of conservatism in the analysis of the fire areas are discussed in Section 6.0 of this analysis, and it can be expected that removal of these conservatisms would result in a significant reduction of the CDF attributed to fire events.

Since the purpose of the IPEEE was to identify potential plant vulnerabilities, and the fire areas listed in NUREG-1742 were determined not to be vulnerabilities, no further detailed analysis of the areas was performed to remove conservatism.

As discussed in the submittal letter for the IPEEE, and summarized in Table 3.4 of NUREG-1742, no vulnerability to fires was identified. Since no vulnerability to fire events was identified, and fire events were not significant contributors to overall CDF, no plant modifications were planned as a result of the IPEEE analyses. This is summarized in Table 3.5 of NUREG-1742.

It can be concluded that changes to the plant and procedures to reduce risk from the fire areas listed in Table 3.3 of NUREG-1742 would not be cost-effective for two reasons. First, fire events overall are a small contributor to overall CDF (about 7% of overall CDF). Therefore, elimination of fire risk would be expected to result in a small benefit. Second, the analyses for the fire areas contain significant conservatism. If these conservatisms were eliminated, then the CDF attributed to each of the areas would be significantly lowered.

b. The results of the revised seismic PRA (submitted along with the revised fire PRA in the letter referenced in the response to SAMA RAI 2.a) show that seismic events contribute about 6.0% of CDF. Of the total seismic CDF of 3.17E-06 per year, roughly 23% is due to failure of the auxiliary building structure. An item to address this failure was included in the initial list of SAMA candidates (SAMA Item 183), but was screened as too costly and not included in Table F.4-2 of the ER. Failure of two block walls contributed to about an additional 22% of seismic CDF in the revised IPEEE. Two items were included in the initial list of SAMA candidates (SAMA Items 181 and 182) to address these failures, and were screened as implemented since modifications to improve the seismic capacity of these walls were completed subsequent to completion of the IPEEE. Therefore, this failure mode would likely be eliminated in an updated analysis. Other seismic-related failures contributed much less than 5% each to seismic CDF. Given these considerations, and using the importance measures from the seismic PRA, the SAMA candidates in Table F.4-2 of the ER were reviewed and it is concluded that none would result in a significant reduction in risk caused by seismic events.

As discussed in the response to SAMA RAI 2.a, the analysis of risk from fire events performed for the IPEEE contained significant conservatisms, and contributed about 7% of overall CDF. As was discussed further in the response to SAMA RAI 2.a, removal of the conservatisms from the analysis would result in a significant reduction in the fire-attributed CDF. Given these considerations, the SAMA candidates listed in Table F.4-2 of the ER were reviewed, and it is concluded that none of the SAMA candidates would provide a significant risk benefit to the dominant fire scenarios.

Internal flooding events have a contribution of 2.0E-07 per year to CDF in the IPE. The vast majority of this frequency is from only one event, flooding in the turbine building basement. However, overall CDF from flooding events is small. Therefore, elimination of this flooding risk would provide a small benefit. Furthermore, significant conservatisms in this analysis (e.g., including success criteria for primary bleed and feed, and not crediting flood mitigating plant features and operator responses), results in the contribution to CDF from flooding being overstated. The use of more realistic success criteria for this scenario or more realistic modeling of mitigation efforts would result in a significant reduction in the contribution to CDF from this scenario. Given these considerations, the SAMA candidates listed in Table F.4-2 of the ER were reviewed, and it is concluded that none of the SAMA candidates would provide a significant risk benefit to the dominant flood scenario.

c. For CNP Units 1 and 2, grade elevation slopes from 608' 0" on the east side of the auxiliary buildings to 595' 0" on the west side of the turbine buildings located on the shore of Lake Michigan. The main floor of the turbine building basements for both units is located at elevation 591' 0" with the condenser pit floors located at elevation 579' 0". No transformers or electrical switchgear modeled in the PRA are located in the turbine building basements. Electrical switchgear rooms are located at elevation 609' 6" in the auxiliary buildings. Therefore, for a flooding event to fail electrical components important to the PRA, the turbine building must flood to a level approximately fourteen feet above grade elevation. Several doors and numerous ventilation louvers prevent accumulation of water to the point that a flood in either turbine building would allow water to enter the electrical switchgear rooms. Therefore, a modification similar to that identified in SAMA RAI 2.c would be not applicable to CNP.

SAMA RAI 3:

According to Table F.4-1, I&M evaluated 194 SAMA candidates. Of these 194 candidates, 32 were obtained from CNP-specific documents. It is not clear that the set of SAMAs evaluated in the environmental report (ER) addresses the major risk contributors for CNP. In this regard, please provide the following:

a. *A description of how the dominant risk contributors at CNP, including dominant sequences and cut sets from the current PRA and equipment failures and operator actions identified through importance analyses were used to identify potential plant-specific SAMAs for CNP.*

b. *The number of cut sets reviewed/evaluated and what percentage of the total CDF they represent.*

c. *A listing of equipment failures and human actions that have the greatest potential for reducing risk at CNP based on importance analysis and cut set screening.*

d. *A list of the top ten items from "reliability issues" initially considered as SAMA candidates.*

e. *For each dominant contributor identified in the current PRA (August 2001), a cross-reference to the SAMAs evaluated in the ER which addresses that contributor. If a SAMA was not evaluated for a dominant risk contributor, justify why SAMAs to reduce these contributors would not be cost beneficial.*

CNP Response to SAMA RAI 3:

a. Potential improvements identified from the August 2001 PRA model update were based on a review of the conclusions and the basic event importance analysis from the update. The results of the August 2001 PRA concluded that loss of RCP seal cooling is the major contributor to core damage. Loss of RCP seal cooling could occur directly either through a loss of CCW that causes a loss of RCP seal thermal barrier cooling, or by a loss of RCP seal injection through the loss of the charging pumps. Loss of RCP seal cooling also could occur indirectly either through a loss of ESW to the CCW heat exchangers, or during an SBO where motive power is lost to the CCW, charging, and ESW pumps.

   An importance analysis for the CDF cutsets of the at-power PRA model was performed. Each basic event with a Fussell-Vesely importance of greater than 0.5%, for a total of 146 basic events, was reviewed to identify any potential SAMA candidates. The results of the importance analysis review to identify SAMA candidates are detailed in the response to SAMA RAI 3.e.

b. As discussed in the response to SAMA RAI 3.a, identification of SAMA candidates from the PRA model results was based on a review of the conclusions from the updated PRA and a systematic and detailed importance analysis of the CDF cutsets. A specific target number of CDF cutsets were not reviewed, because the use of importance analysis results was judged to yield satisfactory insight into potential plant vulnerabilities.

c. Importance analysis results for the CNP have been provided in the response to SAMA RAI 1.c, and identify the equipment failures and human actions with the greatest potential for reducing risk at the CNP. As described in the response to SAMA RAI 3.b, CDF cutset screening was not used to identify SAMA candidates. The conclusions resulting from the PRA were drawn from a high-level review of accident sequence and cutset results. Furthermore, the use of importance analysis results was determined to be sufficient.

d. The top ten items from reliability issues initially considered as SAMA candidates are as follows:

Top Ten Rank	Issue	SAMA Item Number
1	Circulating Water Screens and Debris	186
2	Power Supplies	187
3	Transformers/Offsite Power	188
4	EDG Governor/Voltage Regulator/Fuel Lines	185
5	Zebra Mussel Strategy	189
6	Main Feedpump Health	190
7	Expansion/Bellows/Boots	191
8	AFW Pump/Valves	192
9	Main Steam Isolation Valve Vulnerability	193
10	Main Turbine Controls Upgrade	194

e. Potential improvements identified from the current (August 2001) PRA were based on a review of the conclusions and the basic event importance analysis (See the listing of basic event importance values from SAMA RAI 1.c). Using the results of the current PRA, it was concluded that loss of RCP seal cooling is the major contributor to core damage. Loss of RCP seal cooling could occur directly either through a loss of CCW that causes a loss of RCP seal thermal barrier cooling, or by a loss of RCP seal injection through the loss of the charging pumps. Loss of RCP seal cooling also could occur indirectly either through a loss of ESW to the CCW heat exchangers, or during an SBO where motive power is lost to the CCW, charging, and ESW pumps. SAMA Item 184 was added to the list of SAMA candidates to consider improving RCP seal cooling to address this issue. Numerous options to reduce the contribution of RCP seal cooling to CDF were also identified from the generic industry sources, and are listed as SAMA Items 1 through 23.

A detailed importance analysis for the CDF cutsets of the August 2001 PRA model was performed. Each basic event with a Fussell-Vesely importance of greater than 0.5%, for a total of 146 basic events, was reviewed to identify any potential SAMA candidates. Thirty-four of the basic events, such as complement events or constants, have no physical meaning and can be excluded. Of the remaining 112 basic events, 27 are used to

represent failure of operator actions and are listed together as SAMA Item 172. The disposition of the remainder of the basic events is presented below:

- Thirty of the basic events are related to a loss of RCP seal cooling, primarily during SBO events. The important basic events related to a loss of RCP seal cooling are addressed by SAMA Item 184, which also addresses the conclusion of the PRA update that a loss of RCP seal cooling is the major contributor to core damage.

- Failure of the EDGs is represented by eleven of the basic events. Because SBO is an important contributor to CDF, improving reliability of the EDGs is listed as SAMA Item 185 to address these basic events.

- Five of the basic events represent failure of EDG room ventilation fans. Minimizing the potential for EDG failure caused by failure of the room ventilation fans is listed as SAMA Item 28, which was also identified from generic industry sources.

- Failure of the check valves that isolate the RHR system from the safety injection (SI) system are represented by two basic events. Consideration of these basic events is listed as SAMA Item 126.

- Four of the basic events represent failure of air compressors. SAMA Item 141 is listed to evaluate the potential to improve reliability of the air system. This SAMA candidate was also identified from generic industry sources.

- Failure of 4kV AC circuit breakers to properly transfer electrical power from the normal source of power to alternate and emergency sources of power is represented by four basic events. Failure of these breakers contributes to SBO, so SAMA Item 73, which is in common with a similar SAMA candidate from a generic industry source, is listed to evaluate these basic events.

- Four of the basic events represent failure to recover offsite power. Because several SAMA candidates identified from the generic industry sources relate to power recovery strategy, and because the failure probability values for these basic events are developed from generic industry sources, no plant-specific SAMA candidate is included for these basic events.

- Two basic events represent failures of the RHR system such that it would not be capable of providing injection of water to the RCS, or supplying a source of water to the high head SI pumps. Two SAMA candidates on the list evaluate improving the failure probability of these basic events. SAMA Item 117 evaluates adding an alternate water injection source. SAMA Item 24 evaluates adding another means to

provide containment heat removal. These two SAMA candidates are also used to evaluate the importance of preventing failures of the RHR system.

- Failure of cooling water to the EDGs is represented by one basic event. SAMA Item 79 is included to evaluate removal of cooling water as a dependency for the EDGs. This SAMA candidate was also identified from generic industry sources.

- Common cause failure of all four main steam isolation valves (MSIVs) to close appears in the importance listing. SAMA Item 193 was added to evaluate improving reliability of the MSIVs.

- Initiating events developed using generic industry data are represented by fourteen of the basic events, and were not included as potential SAMA candidates. Any means to reduce the contribution of an initiating event would likely be identified in the generic industry sources used to identify potential SAMA candidates. Therefore, no plant-specific SAMA candidates were added to address these basic events.

- Transient initiating events are represented by one basic event. Improving the transient initiating event frequency is included in SAMA Item 171. This potential SAMA candidate was identified from generic industry sources. Therefore, no plant-specific SAMA candidate is included for this basic event.

- One basic event represents failure of automatic actuation of ECCS equipment. Because additional modeling of manual, backup procedure actions would likely eliminate this item from the importance list, no SAMA candidate was added as a result of this basic event appearing in the importance analysis.

- Failure of the SI pumps is represented by three of the basic events considered from the importance analysis. These events appear because of the conservative success criteria used for LOCAs and primary feed and bleed. Since it is expected and likely that an analysis of system success criteria and subsequent system modeling would eliminate these events from the basic event importance listing, no SAMA candidate is included as a result of these basic events appearing in the importance analysis.

- One of the basic events represents failure of the reactor protection system to insert control rods following a reactor scram signal. The failure probability for this basic event is taken from a generic industry source, and several SAMA candidates (e.g., SAMA Items 144 and 145) identified from generic industry sources evaluate improvements to reduce the contribution from ATWS. Therefore, no plant-specific SAMA candidate is included as a result of this basic event appearing in the importance analysis.

- Failure of the turbine-driven AFW pump to start is represented by one of the basic events in the importance analysis. Because this is the only basic event representing hardware failure of the AFW system, and because the operator action to provide AFW from the opposite unit is included in the list of operator actions considered for evaluation as a SAMA candidate, the benefit of improving the operator action to cross-tie the AFW systems would bound the benefit of improving the turbine-driven AFW pump. Therefore, no SAMA candidate is included as a result of this basic event appearing in the importance analysis.

SAMA RAI 4:

According to Section F.5, the I&M analysis was performed based on a single unit implementation. It is not clear which SAMAs would benefit both units, and how the single unit cost for such SAMAs were estimated (i.e., were the implementation costs divided by 2 to arrive at the single unit implementation costs?). Please provide a list of those SAMAs (both procedural and hardware based) where both units would benefit, and confirm that the reported costs and benefits were developed on a consistent basis (i.e., a single-unit basis).

CNP Response to SAMA RAI 4:

Table 4 of Attachment 2 lists those SAMA candidates that would benefit both units, and that were analyzed in the cost-benefit analyses. Where implementing a SAMA candidate would benefit both units, the costs were shared between both units (i.e., costs were developed on a single unit basis). For example, the cost estimates to provide an independent electrical power supply for distributed ignition system hydrogen igniters (SAMA Item 39) assumed that the costs would be shared between the two units. Other examples include SAMA Items 68 and 115 where one backup electrical power supply could potentially be shared between the units.

SAMA RAI 5:

From the SAMAs in Table F.4-2, 16 SAMAs, grouped into five areas, were identified as cost beneficial. Even though the cost beneficial SAMAs are not aging-related, they appear to warrant further consideration for implementation under the current operating license. Please provide a further evaluation of the most cost-effective means of reducing risk in each of the risk improvement areas. This evaluation should include a consideration of the costs and benefits associated with each of the 16 SAMAs, and the potential to achieve a large portion of the risk reduction at a minimum cost by implementation of a carefully selected subset of the SAMAs. The result of this discussion should be an assessment of which SAMAs, if implemented as a set, would offer a significant cost-beneficial risk reduction. Also, please discuss I&M's plans and schedules for implementing cost beneficial plant improvements.

CNP Response to SAMA RAI 5:

Section F.7 of the ER discussed five potential areas for risk improvement. These areas include minimize consequences of RCP seal LOCAs, minimize consequences of loss of HVAC, remove dependence of distributed ignition system on AC power, minimize consequences of AC bus failures, and improve recovery from ISLOCA. The specific SAMA candidates associated with these five areas are listed in Table F.4-1 of the ER. In each of these categories, with the exception of minimize consequences of AC bus failures, multiple SAMA candidates are included and addressed in a group. Each of the five categories corresponds to a broad issue relating to the risk profile at CNP, and each of the SAMA candidates included in the category represents one specific means of mitigating the risk associated with the broader issue. A discussion of each of the five areas for potential risk reduction is provided below.

- Minimize Consequences of RCP Seal LOCAs

  The results of the August 2001 PRA model update show that accident sequences involving loss of RCP seal cooling contribute to about 70% of total CDF at the CNP. Because loss of RCP seal cooling events are such a large contributor to CDF, a plant-specific SAMA candidate, SAMA Item 184, was added to the initial list of potential SAMA candidates. SAMA Item 184 did not propose a specific plant change to effect lowering the risk from RCP seal LOCAs, since evaluation of benefits should explore a variety of methods to ensure that reasonable cost options are investigated. In addition to the plant-specific SAMA Item 184, numerous SAMA candidates that would reduce the risk from a loss of RCP seal cooling were identified from generic sources and are delineated as SAMA Items 1 through 23 of Table F.4-1 of the ER. It was determined that six of these SAMA candidates, SAMA Items 5, 9, 10, 12, 13, and 160, could potentially be cost-beneficial, and each of the SAMA candidates proposed a specific and unique means of mitigating risk. The benefits of SAMA Item 184 could be considered obtained by implementing any of the six generic SAMA candidates.

  Each of the six SAMA candidates that were evaluated represents one option to minimize the risk of RCP seal LOCA events. The potential to achieve a substantial reduction in risk at a minimum cost was considered by evaluating the six identified SAMA candidates. Each of the options represents a method to eliminate a portion of the risk associated with RCP seal LOCAs. For example, SAMA Item 12 proposes the addition of a completely independent and diverse RCP seal injection system, and is shown to provide a substantial benefit for loss of cooling water events as well as SBO events. SAMA Item 13 proposes a diverse seal injection system. However, because the system would not be independent of station AC electrical power, no benefit would be provided for SBO events. SAMA Item 5 proposes adding the capability to provide cooling to the charging pumps (which provide RCP seal cooling) using temporary connections from a source diverse to the CCW system. This alternative would provide some benefit for loss of cooling water events, but because charging pumps rely on AC electrical power, no benefit would be provided for SBO events.

SAMA Item 10 proposes removing the dependence of RCP thermal barrier cooling on CCW. Implementation of this alternative could be by providing temporary connections that allow a water system diverse to the CCW system to be aligned to the thermal barrier heat exchangers. This alternative could reduce the risk of RCP seal LOCAs for loss of cooling water events as well as SBO events. However, this alternative would not address RCS inventory control issues.

Evaluation of the benefits for each SAMA candidate begins with the CNP PRA model. To calculate the probability of RCP seal failure following a loss of cooling, the CNP PRA uses the "Rhodes" model which concludes that seal heatup begins very soon after the loss of cooling. Therefore, to prevent seal failure, actions to restore seal cooling must be completed quickly. Because of the short time frame available to provide RCP seal cooling, actions taken must be simple and few in number or systems that actuate automatically must be installed. If the actions to provide RCP seal cooling require significant time to perform, then it is unlikely that the actions would be completed before RCP seal failure has occurred.

The cost evaluations for the SAMA candidates ruled out installation of automatically actuated backup systems because plant experience shows that the high costs associated with the installation of such systems would likely exceed the maximum benefit shown in Section F.2.3.5 of the ER. Because automatically actuated systems would not be economically feasible to implement the SAMA candidates, plant modifications would be required so that RCP seal cooling can be restored quickly with a few simple actions. For example, SAMA Item 10 proposed installing piping and connections so that a temporary, alternate cooling water supply could be aligned to the thermal barrier seals by means of a few simple valve manipulations. These valve manipulations would be performed outside the control room. Because of the short time available to prevent RCP seal failure, if initiation of one of the alternatives fail, then it is unlikely that sufficient time would be available to attempt use of a diverse system. In addition, because all proposed alternatives involve manual actions, there would be a high dependence of operator action failures causing failure of the second system.

The six SAMA candidates described above cannot be viewed as independent. Each represents one means to eliminate a different portion of the risk due to RCP seal failures with each alternative addressing the issue in a different manner with differing costs and benefits. By evaluating the six alternatives, the potential to achieve a substantial portion of the risk at the lowest cost was evaluated with the analyses presented in the submittal. Because each of the alternatives has a dependence on the others, and because of the limited time available to restore RCP seal cooling, implementing more than one of the proposed alternatives would provide no more benefit than implementing a single alternative.

- Minimize Consequences of Loss of HVAC

  Four SAMA candidates were in this area of risk reduction. Three of the SAMA candidates, SAMA Items 25, 26, and 27, were identified from generic sources. The fourth, SAMA Item 28, was based on plant-specific, as well as generic, sources. As discussed in Section F.7 of the ER, SAMA Item 27 is a general proposal for providing temporary ventilation. Since only EDG room ventilation was identified as important based on the CNP PRA, and because only electrical switchgear room ventilation systems were specifically identified from generic sources, SAMA Item 27 was applied to only those two areas.

  Each unit of the CNP has two electrical switchgear rooms and two EDG rooms that are located on separate elevations of the auxiliary building. Each of the rooms is cooled by a dedicated, once-through ventilation system. A significant distance separates access to the switchgear rooms from access to the EDG rooms. Furthermore, the EDG rooms are normally supplied with about ten times the flow that is supplied to the switchgear rooms. Therefore, it is unlikely that a single temporary system could be used to implement both SAMA candidates, and there would be no benefit to combining the implementation of these two SAMA candidates.

  Installation of a permanent, backup system for these two areas was eliminated from consideration because such a modification would likely cost much more than the calculated benefit based on experience at CNP. The option to provide cooling to the switchgear rooms was evaluated considering that cooling would be provided using pre-staged, temporary fans and ducting. It was also assumed that power for the fans could be provided from existing electrical power sources located near the rooms.

  For the EDG rooms, use of temporary, pre-staged fans and ducting was the option evaluated for implementation. The flows required for the EDG rooms are about ten times those required for the switchgear rooms. Therefore, much larger and costlier fans would be required. In addition, if the EDGs were required, then existing AC electrical power sources located near the rooms would not be energized. Therefore, a plant modification to provide an electrical power source for the fans from a diesel-backed bus would be required.

  Because the switchgear rooms are located a significant distance from the EDG rooms, a single system or implementation option would not be able to be applied to both areas. In addition, ventilation failures in one area are independent of failures in the other area. Therefore, providing temporary ventilation for one area should be viewed as independent of the other area.

- Remove Dependence of Distributed Ignition System on AC Power

  The intent of each of the SAMA candidates in this area is to ensure that, following a severe accident initiated by an SBO, uncontrolled hydrogen burns do not threaten containment

integrity. SAMA Items 39 and 40 were included in this area. Implementation of either of the SAMA candidates would effectively control hydrogen following SBO events so there would be no additional benefit in combining the two SAMA candidates.

The cost evaluations considered a pre-staged, diesel-powered electrical generator sized to power one train of igniters. Temporary cables, switches, and circuit breakers were postulated to connect the new diesel-powered electrical generator to the hydrogen igniters.

- Minimize Consequences of AC Bus Failures

  The goal of this SAMA candidate is to provide a means to supply electrical power from one emergency bus to another emergency bus within a unit. The cross-tie capability could be either permanently installed or a temporary alternative with equipment staged for use and procedures in place. Implementation of a permanent cross-tie was not evaluated in detail, because the costs of such a modification would greatly exceed the potential benefits. Rather, costs for implementation of the SAMA candidate considered that the capability would be provided to selectively cross-tie individual loads using pre-staged, temporary cabling.

- Improve Recovery from ISLOCA

  Two SAMA candidates, SAMA Items 101 and 172-27, were included in this area. Each of the SAMA candidates was identified from a separate CNP-specific source, but each represents the identical action. Therefore, implementing one of the SAMA candidates implements the other.

During performance of the SAMA analysis, I&M evaluated the various synergies between these potentially cost-beneficial SAMA candidates. For example, a combined implementation of SAMA Items 39, 68, and 115 was not considered for further cost-benefit evaluation for several reasons. First, the power requirements for each of the loads are significantly different. Hydrogen igniters operate on 120 VAC and require about 5kW of power to operate. Each battery charger for the plant DC power system operates on 600 VAC and requires about 78kW. The turbine-driven AFW pump operates from a DC power system separate from other plant DC power systems and uses battery chargers that operate on 600 VAC power and require 25kW of power.

The cost estimates for providing backup power to the hydrogen igniters were taken from an internal NRC memorandum from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "RES Proposed Recommendation for Resolving Generic Safety Issue 189: 'Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident'," dated May 13, 2002 (Accession Number ML021340114). Because the estimates assumed that only hydrogen igniters were being powered, only one generator operating at 120 VAC with a capacity of about 5kW

was considered in the cost-benefit analysis. Furthermore, because of the small capacity, the fuel consumption was assumed to be minimal so costs to ensure an adequate fuel supply were small.

To power one battery charger for plant DC systems, one battery charger for the turbine-driven AFW pump, and one train of hydrogen igniters, a diesel-powered electrical generator with a capacity of 108kW would be needed and would require a much larger fuel source than that considered in the NRC cost estimates for a backup power supply for the hydrogen igniters alone. The physical size of a diesel-powered electrical generator with a capacity of 108 kW in conjunction with the fuel supply requirements would limit the portability of such a unit. In addition, since the battery chargers operate on 600 VAC and the hydrogen igniters operate on 120 VAC, a transformer would be required to provide the proper voltage to the hydrogen igniters. Also, since three systems would need to be supplied individually, individual circuit breakers for each load would be required and some form of switchgear to contain the circuit breakers would need to be provided between the generator and the loads. Given the size of the diesel-powered electrical generator required, the need for a transformer to power the hydrogen igniters, and the need for switchgear, the implementation costs used to provide a backup power supply to the hydrogen igniters would not apply.

Another factor that complicates using one diesel-powered electrical generator to supply the three systems is the location of the loads to be supplied. Hydrogen igniters are supplied from power panels located in the EDG rooms. Battery chargers for plant DC systems are supplied power from sources located in the electrical switchgear rooms. The turbine-driven AFW pump battery chargers and associated power supplies are located in the auxiliary building. Separate access to each of the areas is necessary so one set of cabling could not be used to supply the three loads.

Given the factors described above, it is concluded that little synergy could be obtained by simultaneously implementing the three SAMA candidates. Furthermore, the three SAMA candidates impact different aspects of the CNP risk profile so calculated benefits would not increase if implementation of the three SAMA candidates was combined into one project.

The benefits of implementing more than one SAMA candidate with a single project were considered when calculating benefits. The benefits of implementing multiple means of minimizing the impact of a loss of RCP seal cooling are discussed above where the conclusion is that time constraints prohibit success of more than one alternative. As discussed above, it is also concluded that there is no synergy in providing backup HVAC to both the electrical switchgear rooms and the EDG rooms with a single, temporary system.

SAMA Item 12 proposed that an independent RCP seal injection pump with a dedicated diesel-powered electrical generator be provided. A diesel-powered electrical generator used to implement this SAMA candidate could be used as a backup power supply for hydrogen igniters. The benefits calculated for SAMA Item 12 result mainly from a reduction in CDF to less than half the base value (i.e., a significant reduction in the contribution to CDF due to loss of seal cooling events). Providing a backup power supply for the hydrogen igniters at the same time

would reduce the chance of containment failure given that core damage has occurred and power is not available. For accident sequences other than SBO, power is likely available, so a backup power supply would provide very little benefit. For SBO sequences, the backup seal cooling system eliminates a large portion of CDF risk so the benefits of an additional power supply for the hydrogen igniters would be less than the benefits of only providing backup power to the igniters. However, the costs to implement both SAMA candidates with one system increase significantly, as is discussed above for the DC power systems.

Providing backup power to DC systems with a diesel-powered electrical generator used for RCP seal cooling would provide little synergy. The risk profile for the CNP shows that failure of plant DC power systems is a small contributor to overall CDF. Failure of the turbine-driven AFW pump also contributes very little to plant risk for two reasons. First, supplying AFW from the opposite unit in included in the emergency operating procedures. If only one train of AC power is available on the opposite unit, AFW system design ensures that sufficient flow is available to remove decay heat from both units. Second, current emergency operating procedures include steps to take manual control of the turbine-driven AFW pump when battery capacity is exhausted, which will not occur for at least four hours at the CNP.

For the other SAMA candidates, the changes needed to implement each of the SAMA candidates were so diverse that no synergy would be obtained to combining the SAMA candidates into one project. Furthermore, for the remaining SAMA candidates, the potential benefits were so small that the conclusions reached that the SAMA candidates would not have a positive net present value would not be changed.

CNP is tracking these potentially cost-beneficial SAMA candidates and evaluating implementation options in accordance with the CNP Corrective Action Program and business planning process. Detailed plans and schedules are not available at this time. The evaluation, design, and implementation (if appropriate) of these alternatives will be performed in accordance with CNP procedures for design changes.

SAMA RAI 6:

For certain SAMAs considered in the ER, there may be lower cost alternatives that could achieve much of the risk reduction. Please confirm that low cost SAMAs were considered, and provide a brief discussion of these low cost alternatives.

CNP Response to SAMA RAI 6:

Low cost methods were considered when determining the costs of implementing each of the SAMA candidates. For example, implementation of SAMA Item 67, discussed in the response to SAMA RAI 5, considered the use of temporary cabling and pre-staged equipment to power

selected loads, rather than installation of a permanent cross-tie. Other examples of how low-cost options were considered are given in the response to SAMA RAI 5.

When estimating costs, the use of automatically actuated, permanently-installed equipment was generally not considered unless timing constraints precluded taking manual operator action. For most of the SAMA candidates, implementation considered options such as using temporary hose connections and operator actions from outside the control room.

SAMA RAI 7:

SAMA candidates were considered potentially cost-beneficial if the cost of implementation was estimated to be less than two times the calculated benefit, so as to account for "other risk contributions not specifically quantified by the CNP PRA models." The staff is not convinced that this factor of two is sufficient to encompass the collective impact of several potentially non-conservative assumptions in the baseline analysis, and the added impact of uncertainties in the analysis on the SAMA evaluation process and results. In this regard, please address the following:

a. Provide a list and brief description of these "other risk contributions" that were not quantified, and an estimate of the contribution of each to the factor of two. Examples identified by the staff, that should be addressed in the response, include:

i. The total bounding benefit estimated for each of the SAMAs only accounts for the benefits obtained during the 20 year period of the proposed life extension. This could underestimate the total benefit by about 15 percent since CNP has more than 10 years of operation remaining on its existing license.

ii. The estimates of the benefits for each SAMA are made in base years that are 5-10 years earlier that the base years for the estimates of the costs of implementation. This could underestimate the total benefit by about 20 to 50 percent assuming an average inflation rate of 4% per year.

iii. Sensitivity analyses performed as part of previous SAMA evaluations for MACCS2 inputs such as evacuation and population assumptions could yield variations in population dose of about 20 percent.

iv. The use of a reference pressurized water reactor (PWR) inventory scaled only for power (as opposed to a bounding operating cycle), could result in a significant underestimate of the fission product inventory of important long lived radionuclides that dominant population dose (e.g., an underestimate of about 50 percent for Sr-90 and Cs-137).

b. *The SAMA analysis did not include an assessment of the impact of PRA uncertainties. Please provide the following information to address these concerns:*

   i. *An estimate of the uncertainties associated with the calculated core damage frequency (e.g., the mean and median internal events CDF estimates and the 5th and 95th percentile values of the uncertainty distribution),*

   ii. *An assessment of the impact on the Phase 1 screening if risk reduction estimates are increased to account for uncertainties in the risk assessment, and*

   iii. *An assessment of the impact on the Phase 2 evaluation if risk reduction estimates are increased to account for uncertainties in the risk assessment. Please consider the uncertainties due to both the averted cost-risk and the cost of implementation to determine changes in the net value for these SAMAs.*

CNP Response to SAMA RAI 7:

a. A factor of two was used to bound several sources of uncertainty in the CNP SAMA analysis. The first source of uncertainty was the contribution of external event initiators, including internal flooding, fire, and seismic initiators, which were not explicitly quantified. Internal flooding events had a frequency of 2.00E-07 per year in the IPE analysis. Fire events had a CDF of 3.76E-06 per year, and seismic events had a CDF of 3.17E-06 per year, in the revised IPEEE. Other external events, such as high winds and aircraft accidents, were determined by the IPEEE analyses to be insignificant contributors to plant risk. The total CDF contribution of internal flooding, fire, and seismic events is the sum of the above frequency values, or 7.13E-06 per year.

   Modeling of external events and internal flooding events was based on the IPE models, which produced a CDF of 6.26E-05 per year. The CNP SAMA analysis cost-benefit analyses were based on the August 2001 PRA update, which calculates a CDF of about 4.9E-05 per year for each unit. The contribution of unquantified events should be compared to the August 2001 update, because the CDF for the recent update is less than that of the IPE. Therefore, a higher relative contribution from the unquantified events would result from using the recent update. Based on the values given above, unquantified events would add an additional 15% to total CDF.

   Other types of uncertainty can be related to data, analyst assumptions, modeling, and consequences. Uncertainty in data can be estimated by an uncertainty analysis of the PRA results (as discussed in the response to SAMA RAI 7.b), where it can be seen that the mean and point estimate values are in close agreement. The other sources have been identified in this question. For example, if accounting for benefits considers the time remaining in the current operating license, benefits could be approximately 15% higher (SAMA RAI 7.a ii identified that benefits could be underestimated by 20% to 50%

because of the time period assumed for implementation). Two other factors identified in SAMA RAI 7.a are assumptions involving evacuation and population, and the use of a scaled radiological source inventory instead of actual core inventory. Sensitivity analyses were performed to address these last two factors.

As described on Page F-17 of the ER, evacuation speeds used for the CNP SAMA analysis are based on the maximum estimated evacuation time assuming a winter weeknight under adverse weather conditions. Under these worst-case conditions, 370 minutes are needed for evacuation within the 10-mile Emergency Planning Zone (EPZ). The minimum evacuation time shown on Page F-17 of the ER is 210 minutes. Obviously, the best-estimate timing for evacuation would be between these two values. Nonetheless, a sensitivity analysis was performed where the already conservative evacuation speed was reduced by 50%. This sensitivity analysis resulted in an increase in offsite dose from 42.53 person-rem per year to 46.89 person-rem per year. Offsite property damage costs did not change. Since total offsite exposure costs increase in proportion to total offsite dose, and total offsite property damage costs increase proportional to expected offsite property damage costs, a simple ratio demonstrates that total costs for the base case would be expected to increase from $2,700,000 to $2,800,000.

I&M concludes that a reasonable projection of population was made. However, any increase or decrease in population projection will result in a proportional increase or decrease in population dose. As shown in Section F.2.3.5 of the ER, offsite dose contributes approximately 34% to total benefits for the base case. Therefore, for small changes in population, it can be concluded that a change of one percent in population will result in a 0.34% change in calculated benefits.

To estimate the effects of using a bounding core inventory as opposed to scaling a reference PWR inventory, a sensitivity analysis was performed using the core inventory shown in the CNP Updated Final Safety Analysis Report (UFSAR) Table 14.2.1-3. The inventory shown in this table is considered bounding for current and expected plant operation, and resulted in an increase in offsite dose from 42.53 person-rem per year to 51.66 person-rem per year. Expected offsite property damage costs increased from $64,582 per year to $84,022 per year. Since total offsite exposure costs increase in proportion to total offsite dose, and total offsite property damage costs increase proportional to expected offsite property damage costs, a simple ratio will show that total costs for the base case would be expected to increase from $2,700,000 to $3,100,000.

An initial estimate of the total contribution from the factors discussed could be as shown in the table below:

Factor	Contribution (percent)
Unquantified events	15
Consideration of benefits for remaining portion of license (This amount is provided in SAMA RAI 7.a.i.)	15
Benefits estimated for years earlier than costs (This contribution is the maximum stated in SAMA RAI 7.a.ii.)	50
Reduced evacuation speed	4
Use of bounding core inventory	15
Total contribution	**99**

As demonstrated in this table, use of these values would show that a factor of two could be used. However, use of a factor of two could overestimate the benefits as further discussed below.

For seismic events, failure of two block walls contributed 7.08E-07 per year to the seismic CDF. As described on page F-74 of the ER, these two walls were reinforced during the extended plant shutdown that ended in 2000. Therefore, the contribution to CDF from these walls would be reduced. As discussed in the response to SAMA RAI 2, the analysis of fire events contained several significant conservatisms. Therefore, the CDF from fire events is overestimated. Lastly, the contribution of unquantified events is based on comparing values from the IPE analysis to the current CNP PRA model. Since the current CNP PRA results in a lower CDF than the IPE model, it would be expected that the absolute value of CDF from these events would also decrease.

The second factor addressed is consideration of benefits for the remainder of the current operating license. As stated on page 21 of NUREG/BR-0058, "Value and impact estimates are to be incremental best estimates relative to the baseline case, which is the no action alternative." Since none of the SAMA candidates would be required under the current operating license, implementation of the items before expiration of the current license would be voluntary. NUREG/BR-0184 describes on page 5.20 that analysts should give no credit for "voluntary actions" in the base estimates. Therefore, it can be concluded that not including the 15 percent factor to consider implementation in the current license period is in conformance with the regulatory guidance provided by NUREG/BR-0184.

The CNP SAMA analysis used a worst-case evacuation speed rather than a best-estimate. The sensitivity analysis discussed above shows that decreasing evacuation speed further

would result in an increase of only 4% in costs. However, use of the best-estimate evacuation would result in a reduction of maximum benefit in the base case.

Based on the discussions above, it is concluded that the use of a factor of two bounds the risk contributions not specifically quantified by the CNP PRA models, is in conformance with regulatory guidance for performance of cost-benefit analyses, and is appropriate for the CNP SAMA analysis.

b. The uncertainty analysis for the CNP PRA is shown below:

	Unit 1 CDF (per year)
Point Estimate	4.85E-05
95th Percentile	9.73E-05
Mean	4.95E-05
Median	4.27E-05
5th Percentile	2.23E-05

The ratio of the point estimate CDF (used for the CNP SAMA analysis) to the 95th Percentile CDF is (9.73E-05) / (4.85E-05) or 2.01. Assuming that the benefits calculated from the CNP Level 2 PRA results would propagate in the same ratio, the Phase 1 screening would use a maximum benefit value for the base case of $5,400,000. If a value of $5,400,000 was used, only two of the SAMA candidates originally screened in Phase 1 would be considered for further cost-benefit analysis in Phase 2. The first SAMA candidate, SAMA Item 38, would install a hardened containment vent capable of removing decay heat, and referenced a potential cost of $3,100,000. The benefits of implementing such a system would be similar to the benefits calculated for SAMA Item 24 (i.e., $11,437). Therefore, SAMA Item 38 would be excluded by the Phase 2 cost-benefit analysis screening.

If the 95th percentile CDF was used, then the second SAMA candidate that would not have screened in Phase 1 would be SAMA Item 58. This SAMA candidate referenced potential costs of $2,500,000 to $4,700,000. This SAMA candidate proposed adding a RPV exterior cooling system so that core debris can be cooled before causing RPV failure. As with SAMA Item 38, the benefits of implementing such a system would be similar to the benefits calculated for SAMA Item 24. Therefore, SAMA Item 58 would be excluded by the Phase 2 cost-benefit analysis screening.

The benefits and costs calculated for SAMA candidates determined to not be cost-beneficial in the Phase 2 screening are shown in Table 5 of Attachment 2, provided in response to SAMA RAI 8.a. As can be seen in that table, the benefits for many of the SAMA candidates are very small, and doubling of the benefits by using the 95th

percentile CDF value would not change the screening conclusions. Furthermore, as indicated in the response to SAMA RAI 8.a, the costs estimated for many of the SAMA candidates did not include many of the most significant costs. A detailed cost estimate would result in much higher costs for most of the SAMA candidates. Therefore, it is concluded that use of the 95th percentile CDF value would not change the conclusions of the Phase 2 cost-benefit screening.

SAMA RAI 8:

Table F.4-2 does not provide the estimated cost for those SAMAs where the estimated cost is ">2 x Benefit." This precludes an independent assessment of the relative cost-benefit conclusion, especially as it relates to the sensitivity analysis. In this regard please provide the following:

a. An estimated cost (approximate) for all of the screened out SAMAs. Also provide a brief description of the methodology, information sources, major cost elements, and assumptions (i.e., design assumptions, % contingency, unit costs, average hourly labor rates, etc.) used to develop these cost estimates. If no specific cost estimate was developed for a given SAMA because the cost was judged to be much greater than the estimated benefit, please provide the rationale for this conclusion.

b. Justification for the estimated cost for: (1) SAMA 154 - Make procedural changes only for the RCS depressurization option, which has a benefit of <$315,931 and (2) SAMA 171 - Enhanced screen wash, which has a benefit of <$221,837.

CNP Response to SAMA RAI 8:

a. Table 5 of Attachment 2 provides the cost estimates used in the cost-benefit analysis for all of the SAMA candidates that were listed as screened out in Table F.4-2 of the ER. The degree of detail included in the cost estimates was commensurate with the level of benefits that could potentially be achieved by the SAMA candidate. For example, SAMA candidates with low potential benefits were not subjected to detailed cost estimates. Instead, certain minimum costs were addressed until the total for the partial costs considered greatly exceeded the benefits that would be expected. Table 5 of Attachment 2 indicates the SAMA candidates for which significant expenses were not included in the cost estimates.

In determining the costs of implementing each SAMA candidate, standard costs were established for certain implementation activities. First, a minimum cost of $50,000 was considered for a simple design change. A simple design change would include activities such as completing and assembling the design change package paperwork, performing one or two simple calculations, revising fewer than six drawings, and revising fewer than

four procedures. This cost does not include any work associated with procurement of materials, job planning, or installation. Complex design changes would cost considerably more.

Second, a simple procedure change not associated with a design change would have a minimum cost of $10,000 for preparation, review, approval, and implementation. Complex procedure changes or changes involving emergency operating procedures would cost considerably more.

Third, any plant change that requires operator training would have a minimum cost of $10,000 for lesson plan preparation and two hours of instruction for each of six operator crews. If the procedure change requires simulator training, then costs would be higher and $20,000 would be assumed.

These estimated costs were compared to those used in Attachment 1 to the NRC internal memorandum from John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, to Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, dated May 13, 2002. This attachment to the memorandum is titled "Backup Power for PWRs with Ice Condenser Containments and for BWRs with Mark III Containments under SBO Conditions: Impact Assessment," Revision 2. The comparison demonstrates that the estimated costs used for the CNP SAMA analysis are consistent with those in the report. In fact, the estimated costs used in the CNP analysis may be slightly lower in many cases.

For some of the higher benefit SAMA candidates, costs were estimated based on similar plant modifications that may have been completed in the recent past. For example, the cost estimates for SAMA Item 149, which involves installing a new system of RCS relief valves, considered the actual costs associated with the recent replacement of a RHR valve inside containment at CNP.

For certain SAMA candidates, projects are currently being planned that would implement procedure and/or plant modifications related to the purpose of the SAMA candidate. When such situations existed, the costs were taken directly from the project estimates. For example, a project to replace main turbine controls is currently being implemented, and the costs for this project were used as the estimate for SAMA Item 194.

Specific cost estimates were not developed for 18 SAMA candidates listed in Table 5 of Attachment 2. The rationale for not including cost estimates for each of these SAMA candidates is discussed below:

- The goal of SAMA Item 72 is to improve the reliability of the EDGs by providing the capability to replenish the fuel oil day tank using pumps from the opposite unit. The current CNP system has two fuel oil storage tanks. Each tank supplies the EDG for

one train on each unit. Each EDG is supplied from a day tank that provides less than one hour of operation at full load. For operating times approaching one hour, typical of most EDG surveillance tests, fuel oil must be supplied to the day tank from the fuel oil transfer system using one of two pumps for each EDG. While the fuel oil storage tank is shared between two EDGs across units, the fuel oil transfer pumps can only supply the associated EDG. This SAMA candidate would allow the fuel oil transfer pumps on one unit to provide a makeup to the EDG on the opposite unit.

The CNP PRA model includes failure to supply makeup to the day tank as a failure mode for the EDG. Therefore, all past failures to supply fuel oil to the day tank would be included in the EDG fail-to-run data. Implementation of this SAMA candidate would be reflected in the CNP PRA model as reduced fail-to-run probability values for the EDGs.

The EDG failure data from the IPE, the updated IPE, and the recent CNP PRA update were reviewed. No failures of the fuel oil transfer pumps were identified. Implementation of this SAMA candidate would not affect the likelihood of any failure mode that has been observed over a significant portion of CNP operation. Therefore, the benefit of implementing this SAMA candidate is judged to be insignificant, and no further analysis was performed. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for SAMA Item 72.

- SAMA Item 126 was identified by reviewing the basic event importance analysis from the CNP updated PRA. As modeled in the updated PRA small break LOCA accident sequence analysis, RCS pressure likely would not decrease below the shutoff head of the RHR pumps. During the injection phase of a small break LOCA, the SI pumps would be providing flow to the RCS through the cold leg injection lines, which are shared with the RHR system. Since the RCS pressure would be above RHR shutoff head, flow from the SI system should cause check valves to close, preventing flow from the SI system to RHR system piping.

  If the check valves fail to close, then RHR system piping for the train associated with the failed check valve would be exposed to SI system operating pressure. If flow past the failed check valve is larger than the capacity of the RHR system relief valves, then the RHR system could be subject to pressures exceeding design limits. Should design pressure limits be exceeded, there is a chance that rupture of the RHR piping outside of containment could occur, and coolant needed for ECCS recirculation would be lost.

  This failure mode is important to the updated PRA model because of conservatism and simplification of the CNP PRA model. First, small break LOCA success criteria are based on design basis accident analyses that require one SI and one charging

pump to mitigate small break LOCAs. It is likely that a plant-specific analysis of small break LOCA success criteria would relax the criteria to any one of four high-head pumps (SI or charging pumps) injecting into the RCS.

Second, exposing the RHR piping to SI system operating pressure may not result in rupture of the RHR system. For breaks near the larger end of the small break LOCA spectrum, the SI pumps would be injecting into the RCS, and RCS pressure would be less than RHR system design. If the RHR system is exposed to SI system operating pressure for these types of events, then it is more likely that the system will remain intact. Analyses of similar systems for ISLOCA analyses typically show that the RHR system will remain intact over 90 percent of the time when exposed to full RCS pressure.

Next, not all failures of the check valves would be expected to pass full flow of the SI pumps, but a spectrum of leak flows would be expected. Based on experience and data published in NUREG/CR-5102, it is expected that the greatest portion of failures would occur with the smallest leaks. Should failure of the check valves occur, but in a manner that backflow is less than the capacity of RHR system relief valves, then an overpressure condition would not occur.

Explicit consideration of the factors described above would likely eliminate failure of the SI system check valves as important contributors to CDF. Therefore, it is concluded that implementation of a SAMA candidate to prevent failure of the ECCS caused by the SI system overpressurizing the RHR system would result in an insignificant benefit, and no further evaluation was performed. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for SAMA Item 126.

- SAMA Item 162 would separate the CCW trains so that a fault on one train may not cause failure of the opposite train. The CCW system at CNP consists of two trains, with one normally operating and one in standby. The system is normally operated with the pump suction and discharge cross-connect valves open, and the heat exchanger outlet cross-connect valves open. Operation in such a configuration allows recovery of any single failure in the system with minimal interaction by the operators. As part of the switch to ECCS recirculation, the operators split the CCW trains by closing the cross-tie valves, thus preventing any single passive failure from failing the CCW system.

  Implementation of this SAMA candidate would remove operational flexibility of the CCW system. The benefits analysis determined that implementation of this SAMA candidate would result in an increase in CDF for CNP because of the lost operational flexibility. Since CDF would increase if SAMA Item 162 was implemented, no cost estimates were performed.

- SAMA Item 163 would separate the ESW trains so that a fault on one unit may not cause failure on the opposite unit. The ESW system at CNP consists of two trains for each unit with one train on each unit normally operating and one in standby. Cross-tie valves connect one train on each unit to the opposite unit and all ESW pumps start on a SI signal to either unit. The trains within a unit are physically separated with the exception of a cross-tie provided to the EDGs.

  Implementation of SAMA Item 163 would remove operational flexibility of the ESW system. The benefits analysis determined that implementation of this SAMA candidate would result in an increase in CDF for CNP because of the lost operational flexibility. Since CDF would increase if SAMA Item 163 was implemented, no cost estimates were performed.

- Nine basic events listed under SAMA Item 172 did not have costs evaluated. The goal of SAMA Item 172 is to reduce the probability that failure to correctly perform an operator action will cause or exacerbate a severe accident. Operator actions that are important to preventing core damage were identified from importance analysis of the PRA cutsets.

  Item 1 of SAMA Item 172 represents operator failure to energize the hydrogen igniters in certain large break LOCA scenarios or during an SBO. Large break LOCA is not important to CDF at CNP, so basic event "HI1-FAILURE-HE" appears in the importance listing because of SBO events. Because no power is available to the hydrogen igniters in an SBO, the action is assumed to be a guaranteed failure, and reducing the failure probability of this event would require a modification to supply electrical power to the igniters in an SBO. SAMA Item 39 evaluates the benefit of such a modification. Other operator errors related to failure of hydrogen igniters are not important contributors to CDF or LERF. Therefore, no further consideration of this item is performed.

  Item 2, "OLI---13B-EHHE," is a conditional failure probability and represents failure to depressurize the RCS to allow RHR injection given failure of the high-pressure injection systems. The failure probability of this event was assigned a value of 0.5 because of a high dependence on nearly simultaneous actions to switch to ECCS recirculation. Reviewing the PRA model shows that this action is important to CDF because of the use of overly conservative success criteria. Since the importance of this basic event would be significantly reduced with more detailed modeling and a refinement of success criteria, initiatives to reduce the failure probability of this event are not the appropriate means to address the issues underlying the importance of this human error probability. Therefore, no further analysis of this event was performed, and the benefits of implementation would be negligible. Since the benefits of the

SAMA candidate are judged to be negligible, no cost estimates were performed for Item 2 of SAMA Item 172.

The basic event of Item 10, "EPORVMANOPENHE," represents failure of operator action to take manual control of steam generator power operated relief valves, and appears in the cutsets and importance because of simplifications in the modeling of the system. More detailed modeling of the system would remove this event from the importance listing. Therefore, no further analysis of this event was performed, and the benefits of implementation would be negligible. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for Item 10 of SAMA Item 172.

The basic events of Item 11, "BAMV-ESWWESTHE" and Item 12, "BAMV-ESWEASTHE," represent errors that occur just prior to a loss of CCW initiating event, and represent failure of the operator to place the standby CCW heat exchanger on line after failure of the operating heat exchanger. During normal operation, one CCW heat exchanger is on line and the other is in standby. Each of the basic events represents operator action to open a single motor-operated valve from the control room. These events are modeled as a simple, single action in response to multiple annunciator alarms. It is not likely that initiatives taken to reduce the importance of these events would result in a reduction of the human error probability values given the analysis performed. Therefore, no further evaluation of these events was performed, and the benefits of implementation would be negligible. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for Items 11 and 12 of SAMA Item 172.

The basic event of Item 16, "AFW-OPENDOORHE," represents operator error to open the AFW pump room doors on a loss of room cooling. Room heat-up calculations performed since the PRA was completed show that room cooling is not required for AFW pump operation. Therefore, this event could be removed as a failure mode for the AFW system, and no further analysis is performed. Since the failure mode modeled with this item can be eliminated from the model, no cost estimates were performed for Item 16 of SAMA Item 172.

The basic event of Item 18, "AFW-CROSSTIEHE," represents operator error to provide steam generator makeup from the opposite unit in accordance with the emergency operating procedures. The failure probability of this event was developed using a very simple screening-type analysis, and it is likely that a more detailed analysis would result in a significantly lower human error probability. Because the event appears in the list of important basic events due to the simplified technique used to model the human error probability, and because a lower human error probability is expected with a detailed analysis, any initiatives to improve performance of this action should begin with a more detailed analysis. Therefore, no further analysis of

this event was performed, and the benefits of implementation would be negligible. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for Item 18 of SAMA Item 172.

Items 19 and 20 of SAMA Item 172 represent failure to strip loads from the safety buses to prevent failure of the EDG caused by aligning it to a loaded bus. The EDG is designed to rapidly start and load on the bus. In addition, the operators must complete immediate actions of the emergency operating procedures prior to taking other actions to recover the plant. Given the physical constraints on the time available to strip the buses, reducing the failure probability of this action would not be likely so no further evaluation of these events was considered and the benefits of implementation would be negligible. Since the benefits of the SAMA are judged to be negligible, no cost estimates were performed for Items 19 and 20 of SAMA Item 172.

- The goal of SAMA Item 177 is to provide better assurance that the plant could withstand damage caused by tornado winds. SAMA Item 177 would strengthen the RWST so that it could withstand a higher strength tornado. The CNP IPEEE shows that tornado-induced core damage is an insignificant contributor to core damage. Given that tornadoes are insignificant contributors to core damage, any changes to eliminate or reduce the contribution to CDF from tornado events would provide an insignificant benefit. Therefore, no further analysis of the benefits from this SAMA candidate is performed.

- The goal of SAMA Item 179 is to provide better assurance that the plant could withstand damage caused by tornado winds. SAMA Item 179 would strengthen the 4kV switchgear rooms and condensate storage tank so that they could withstand a higher strength tornado. The CNP IPEEE shows that tornado-induced core damage is an insignificant contributor to core damage. Given that tornadoes are insignificant contributors to core damage, any changes to eliminate or reduce the contribution to CDF from tornado events would provide an insignificant benefit. Therefore, no further analysis of the benefits from this SAMA candidate is performed.

- SAMA Item 191 was identified by a CNP equipment reliability programmatic review, which determined that a preventive maintenance program for expansion joints, bellows, and boots could improve plant operation. Improving the reliability of such components could reduce the frequency of internal flooding events. Currently, flooding events are not identified as significant contributors to core damage. Therefore, it was concluded that implementation of this SAMA candidate would result in a negligible benefit. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for SAMA Item 191.

- SAMA Item 192 was identified by a CNP equipment reliability programmatic review. Performance of the AFW system was identified as an issue that repeatedly challenges plant operation. Improving reliability of the AFW system would reduce the chance of a loss of secondary heat sink following a reactor trip.

  The CNP PRA models failure of the AFW system if three trains of AFW for the affected unit and both motor-driven trains of the opposite unit fail to provide flow to the steam generators. As evidenced by the basic event importance analysis, failure of the AFW system is not an important contributor to overall CDF. Such a result is expected, since five trains of AFW must fail to cause loss of secondary heat sink. Therefore, implementation of this SAMA candidate is expected to result in negligible benefit, and no further evaluation was performed. Since the benefits of the SAMA candidate are judged to be negligible, no cost estimates were performed for SAMA Item 192.

- SAMA Item 193 was identified by a CNP equipment reliability programmatic review, which identified that MSIVs have slowly drifted closed from the full open position during previous reactor startups. When the valves start to drift closed, operator action is required to open the valves thereby posing a distraction to the operators.

  The CNP PRA models only one function for the MSIVs, closure to isolate a faulted steam generator. Since the intent of this SAMA candidate is to improve the reliability of the MSIVs to stay open, there would be no impact on the function modeled in the PRA. Therefore, implementation of this SAMA is expected to result in negligible benefit, and no further evaluation is performed.

b. Although the description of SAMA Item 154 in Tables F.4-1 and F.4-2 of the ER states that it would encompass only procedural changes, Table F.4-2 shows the estimated cost of the enhancement as over $600,000, a value inconsistent with expectations for procedure changes. The high cost is indicated because hardware modifications that add additional depressurization capability would be required to effect any meaningful change in CDF. Success criteria analyses for the CNP PRA show that following a small break LOCA where high-pressure ECCS systems have failed and the RCS does not depressurize to allow low-pressure injection, current plant systems allow very little time for depressurization before core damage would be expected. Table F.4-2 should have indicated that procedural change alone would not be practical or effective in reducing risk, and that SAMA Item 154 could not be implemented without the hardware changes proposed in SAMA Item 153.

   The cost estimate for SAMA Item 171 was taken from a project currently being implemented at CNP to improve the effectiveness of debris removal from raw water systems. This project was initiated after two recent events involving fish intrusion and silt intrusion caused operational challenges.

SAMA RAI 9:

Please provide the following information concerning the MELCOR Accident Consequences Code System analyses:

a. *The discussion of meteorology indicates that there are data voids in the 1997 data set used. A power law was used to extrapolate the 60-meter wind speeds from the 10-meter wind speeds. Provide a more detailed description of the power law application and a justification of its use (such as comparison of 10-meter and 60-meter wind speeds from known months against a power law extrapolation). Confirm that the 1997 data set is representative of the CNP site and justify its use.*

b. *The MACCS2 analysis uses a reference PWR core inventory at end-of-cycle calculated using ORIGIN. The ORIGIN calculations were based on a 3-year fuel cycle (12 month reload), 3.3% enrichment, and three region burnup of 11000, 22000, and 33000 Mwd/MTU. Current PWR fuel management practices use higher enrichments and significantly higher fuel burnup (>45000 Mwd/MTU discharge burnup). The use of a reference PWR core instead of a plant specific cycle could significantly underestimate the inventory of long-lived radionuclides important to population dose (such as Sr-90, Cs-134 and Cs-137), and thus impact the SAMA evaluation. For example, SAMAs 49, 124, 125, 139, 149, 153, and 185 offer a significant reduction in person-rem (per Table F.4-2), and might become cost-beneficial using a higher inventory. Please evaluate the impact on population dose and on the SAMA screening and dispositioning if the SAMA analysis were based on the fission product inventory for the highest burnup and fuel enrichment expected at CNP during the renewal period.*

c. *I&M estimated the population for year 2038 by extrapolating the growth rate between the 'actual' year 2000 and the 'estimated' year 2020 populations. I&M then applied this growth rate to the actual year 2000 population through the year 2038 assuming the growth rate would remain constant. If the actual population at year 2000 was higher than the estimated population at year 2000, the I&M extrapolation method would automatically predict a slowdown in population growth (and possibly a decrease) to year 2038 in the face of accelerated growth at year 2000. This is non-conservative and could significantly under predict the year 2038 population. Please evaluate the impact on the SAMA analysis if a more conservative approach for extrapolating population for year 2038 were used, such as using the estimated year 2000 population rather than the actual year 2000 population.*

d. *The I&M reported total population for year 2000 is consistent with the population tables reported in SECPOP2000 (NUREG/CR-6525, Rev. 1, Appendix F) for the 50 mile radius. However, the rosette population distribution differs significantly in the 30-40 and 40-50*

mile radius for the sector regions NE/ENE and SW/WSW (Table F.2-7) compared to both the SECPOP2000 and licensee's reported populations (page F-4 of the stated reference). The I&M evaluation references SECPOP90. Please provide a discussion of the differences noted and potential impact. If the impact is significant, provide justification for which distribution is appropriate.

CNP Response to SAMA RAI 9:

a. Although 60-meter meteorological data is discussed in Appendix F of the ER, the MACCS2 modeling for performing the CNP Level 3 PRA analysis only used the 10-meter meteorological data. The 10-meter meteorological data did not have any gaps, and was collected from the onsite CNP meteorological tower. Therefore, the meteorological data set is representative of the CNP site. As discussed in Section F.2.2.1 of the ER, confirmation that the meteorological data was representative was performed by checking the data against meteorological data from three previous years. The 10-meter data from 1997 was selected as the best available and most representative data.

b. I&M used the NRC-approved MACCS2 computer code model and instruction manual in performing the CNP Level 3 PRA analysis, and followed the precedent of at least 12 previous license renewal applications that used the reference PWR core inventory. The base core inventory in the MACCS2 computer code model is for a reference PWR producing 3,412 MWt. Since the reference PWR core inventory is included in the MACCS2 computer code model developed by the NRC and NRC contractors, I&M does not have information about the bases for the values used. As stated in Section F.2.2.1 of the ER, CNP is a two-unit PWR plant that produces a power level of 3,304 MWt for Unit 1 and 3,468 MWt for Unit 2. Therefore, the core inventory for CNP was obtained using the higher Unit 2 power level by adjusting the end-of-cycle values for a 3,412 MWt PWR by a linear scaling factor of 1.0164. Current CNP fuel enrichments (Region A with 3.4 percent and Region B with 3.8 percent, by weight) remain close to the values indicated for the reference PWR core inventory. However, average core burnup at CNP is higher than that described in SAMA RAI 9.b, averaging 30,764 MWD/MTU at the end of cycle using the current nuclear fuel management practices.

I&M evaluated the sensitivity of the dose risk to increases in core inventories at CNP by substituting the core inventories indicated in CNP UFSAR Table 14.2.1-3 in place of those based on the MACCS2 computer code sample input. This inventory is based on a maximum reactor thermal power of 3,588 MWt, 30,764 MWD/MTU average burnup at end of cycle, with two nuclear fuel enrichments of 3.4% and 3.8%, representing the highest burnup and enrichment expected at CNP under the current nuclear fuel management practices. Modeled inventories of some of the transuranic isotopes (Am, Cm, Pu) increased almost 4 times above the reference PWR core inventory provided in the MACCS2 computer code. Modeled inventories of Sr-90, Cs-134, and Cs-137 increased by 30%, 60%, and 40%, respectively. The resulting increase in baseline dose

risk was 21%, as further described in the response to SAMA RAI 7.a. Based on this sensitivity analysis, and based on the larger uncertainties in the CNP Level 3 PRA analysis already accounted for in the methodology used to calculate benefits of individual SAMA candidates, it is concluded that use of the reference PWR core inventory with the MACCS2 computer code model, adjusting the end-of-cycle values for a 3,412 MWt PWR by a linear scaling factor of 1.0164, yields sufficiently accurate and reasonable results for dose risk.

c. The Year 2020 county population projections were based on 1990 Census data projected to Year 2000 by the states for counties within the 50 mile radius of CNP. I&M determined the growth rate used for those projections, and then used them to project Year 2038 county populations starting with actual Year 2000 Census data. As correctly stated in SAMA RAI 9.c, if those growth rates based on 1990 Census data were too small, then the projection to Year 2038 will not conservatively estimate future population.

As recommended in SAMA RAI 9.c, a way to ensure the projections are conservative would be to compare the estimated Year 2000 population with the actual Year 2000 population. I&M has further evaluated the original state projections, and has found that the estimated population in Year 2000 for most counties is less than the actual Year 2000 population. However, the sum total of the estimated Year 2000 county populations was less than the sum total of the actual Year 2000 county populations by 0.3 percent.

I&M has calculated that if the actual Year 2000 county populations were decreased by 0.3 percent, the resulting increase in estimated Year 2038 county populations, using the same methodology as originally used, would be 0.29 percent. Therefore, the corresponding increase in population dose risk would be approximately 0.29 percent. Substituting a decrease of even 1 percent in the actual Year 2000 county populations would result in an increase in estimated Year 2038 county populations and in population dose risk of only 0.95 percent. Based on the results of this evaluation, it is concluded that the impact of the method originally employed has negligible effect on the results of the CNP Level 3 PRA analysis.

d. I&M did not use either SECPOP90 or SECPOP2000 to determine the rosette population. As described in Section F.2.2.1 of the ER under "Population Distribution," I&M used Geographic Information Systems methods with 2000 Census block-group data as inputs. The distribution to the 160 population sectors used methods described in the referenced section (area and density weighting) for determining how to weight block-groups that were dissected by the population sector boundaries. The projections to Year 2038 were performed on a sector-wise/county area-weighted basis.

Given that the SECPOP2000 total population agrees well with the total used in the ER, it is evident that a different method of parsing the population to the sectors was used by

SECPOP2000. Also, it appears that SECPOP2000 may use census tract data rather than block-group data, and may use a different map projection than that used by I&M.

To confirm the validity of the methods used, I&M performed a sensitivity calculation substituting the SECPOP2000 population distribution for CNP (as referenced in SAMA RAI 9.d) for the Year 2000 population distribution used in the baseline study. Population growth rates were not changed from the baseline study. The use of the SECPOP2000 population distribution would result in a decrease of 4% in the baseline dose risk. Based on the results of this sensitivity calculation, it is concluded that the impact of the method originally employed has negligible effect on the results of the CNP Level 3 PRA analysis.

SAMA RAI 10:

In light of the issues raised in NUREG/CR-6427 concerning the likelihood of early containment failure in SBO events, please provide the following:

a. *A reevaluation of the benefits associated with SAMA 39 (Create/enhance hydrogen igniters with independent power supply) and SAMA 40 (Create a passive hydrogen ignition system) assuming a containment response consistent with the findings in NUREG/CR-6427 (i.e., using the conditional containment failure probabilities for DCH and non-DCH events provided in Tables 4.21 and 4.24 of NUREG/CR-6427, respectively). Indicate whether the PRA model was modified to reflect these conditional failure probabilities, and, if so, how the PRA model was modified to reflect these conditional failure probabilities.*

b. *A breakout of the SBO CDF frequency at CNP in terms of the contribution from fast-SBO and from slow-SBO,*

c. *The estimated time to the onset of core damage for the frequency-dominant fast-SBO and slow-SBO sequences,*

d. *An assessment of the benefits (person-rem per year and dollars) of a pre-staged versus a portable backup power source for the hydrogen igniter system given the conditional containment failure probabilities in NUREG/CR-6427, and the estimated effectiveness of each implementation option in fast-SBO and slow-SBO events.*

e. *A description of the basis for the estimated cost of $147,000 for both SAMA 39 and SAMA 40. Clarify whether this value reflects the per unit cost or the site cost.*

CNP Response to SAMA RAI 10:

a. One input used for the benefit portion of the SAMA analyses was the October 2003 Level 2 PRA, which uses containment failure probability values taken from NUREG/CR-6595. As a sensitivity analysis, the containment failure probability values given in NUREG/CR-6427 were incorporated into the Level 2 PRA. Then, using the revised Level 2 PRA, cost-benefit analyses were performed for selected SAMA candidates.

Table 4.21 of NUREG/CR-6427 provides recommended values for the probability of containment failure due to overpressure following a DCH event. The recommended value for SBO events is 0.820. The recommended value for non-SBO events is 0.000. As stated on page 66 of NUREG/CR-6427, the value is higher for SBO events because hydrogen igniters are not available to control hydrogen concentrations prior to RPV breach, and because containment sprays are not available. As described on page 67 of NUREG/CR-6427, containment failure probabilities are lower for non-SBO events because containment sprays are available to control containment pressure.

Table 4.24 of NUREG/CR-6427 provides recommended values for the probability of containment failure due to overpressure following a non-DCH event. The recommended value for an SBO event is 0.935, and this value is associated with hydrogen burn events. The recommended value for non-SBO events is 0.084, and this value is associated with containment failures caused by steam spikes. As described on page 70 of NUREG/CR-6427, high levels of hydrogen are available for combustion only on SBO events because igniters would likely be available for events in which AC electrical power is available.

The CNP Level 2 PRA is similar in structure to the simplified logic tree used in NUREG/CR-6427 to evaluate early containment failure. However, because of modeling differences, the failure probability values identified above could not be directly incorporated into the CNP Level 2 PRA. Therefore, containment failure caused by issues related to DCH is evaluated in the revised CNP Level 2 PRA by asking three questions:

- Are hydrogen igniters operating?
- Is the RCS pressure at RPV failure high or low?
- Is core damage arrested in-vessel?

For each core damage sequence, operation of hydrogen igniters is evaluated from the conditions specified in the Level 1 PRA event trees.

RCS pressure at RPV failure is evaluated for each core damage sequence from the conditions specified in the Level 1 PRA event trees. RCS pressure is evaluated high

unless specific actions to depressurize the RCS that are evaluated in the Level 1 PRA event trees are successful, or if the initiating event is a large break LOCA.

Arresting core damage in-vessel can only occur for LOCA events (other than SGTR events) where RCS injection was successful. If the initiating event is not an LOCA or if RCS injection is not successful, then arresting core damage in-vessel is evaluated as not possible.

As summarized above, NUREG/CR-6427 evaluated containment failure due to DCH events for SBO events and non-SBO events, with the biggest factor that distinguishes the two being the functionality of hydrogen igniters. For SBO events, NUREG/CR-6427 recommends using a value of either 0.82 or 0.935 for the probability of containment failure. Since each of these values is significantly higher than the values used in the CNP Level 2 PRA, the sensitivity analysis conservatively used a value of 1.0 for the probability of containment failure given failure of the hydrogen igniters. For non-SBO events, NUREG/CR-6427 recommends using a value for containment failure of either zero or 0.084 depending on the particular phenomena that are occurring. The sensitivity analysis conservatively used a probability of 0.09 for containment failure when hydrogen igniters are operating.

After incorporating the revised containment failure probability values into the CNP Level 2 PRA, the frequency of each of the eight STCs shown in Table F.2-8 of the ER was determined. The conditional offsite dose and conditional offsite property damage that would result for each of the STCs was then calculated. The revised values are shown in Table 6 of Attachment 2.

Using these revised consequence values, the unmitigated risk monetary value was recalculated using the methodology presented in Section F.2.3 of the ER. The results of this calculation are shown in Table 7, and result in a maximum attainable benefit of $3,417,660.

Two SAMA evaluations specifically addressed hydrogen control measures. The first evaluation considered SAMA Items 39 and 40, and determined the impact of removing the dependence of hydrogen igniters on the currently available AC electrical power sources. The second evaluation considered SAMA Item 41, and determined the impact of simplifying actions in the procedures associated with energizing hydrogen igniters. The benefits of these cases were re-evaluated to determine the impact that would be expected if the containment failure probability values from NUREG/CR-6427 were used. The results of these evaluations are shown in Table 7 of Attachment 2.

From the cases evaluated, it is seen that use of the containment failure probability values from NUREG/CR-6427 would increase the benefits that are calculated for the cases evaluated. However, for each of the cases, use of the revised values would not change

the conclusions of the cost-benefit analysis. In other words, SAMA Items 39 and 40 were determined to be potentially cost-beneficial, and SAMA Item 41 was determined to not be cost-beneficial, using either the NUREG/CR-6595 or NUREG/CR-6427 values.

b. A definition of a "fast-SBO" is given in Volume 1, page B-1 of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants." Consistent with that definition, the CNP PRA model defines a "fast-SBO" as an SBO event with failure of all AFW and failure to recover power before core damage. For this analysis, all other SBO events are designated as "slow-SBOs." As shown in Table F.2-2 of the ER, SBO events contribute approximately 37 percent to total CDF and essentially all of these events are considered as "slow-SBO" events. "Fast-SBO" events contribute substantially less than 1 percent to total CDF. While the split between "fast-SBO" and "slow-SBO" events may appear to be unusual when compared to other plants, the result is expected because of several plant-specific unique design features described below.

At CNP, each unit is equipped with two motor-driven and one turbine-driven AFW pump, and any single pump is capable of providing sufficient flow to remove decay heat and to support a controlled RCS cooldown. In addition, each motor-driven AFW pump can be aligned to supply flow to the opposite unit, and the actions to implement this alignment are contained in the emergency operating procedures. Control power for each turbine-driven AFW pump is provided from a dedicated and independent DC electrical power source supplied from separate and independent battery systems. Each unit turbine-driven AFW pump battery system is provided with two battery chargers, with one battery charger powered from the associated unit Train A AC electrical power sources and the other battery charger powered from the associated unit Train B AC electrical power sources. In addition, each unit is provided with a 500,000 gallon condensate storage tank, with each tank required by the Technical Specifications to contain sufficient water inventory to provide 9 hours of decay heat removal. Each unit supplies generated AC electrical power to a separate offsite electrical power distribution system. CNP Unit 1 supplies power to a 345 kV offsite electrical distribution system, while CNP Unit 2 supplies power to a 765 kV offsite electrical distribution system.

The overall effect of these plant-specific unique design features on the CNP PRA model is to effectively provide five separate and independent trains of AFW to each unit. For an SBO on a unit, the two motor-driven AFW pumps would be failed. However, the two motor-driven AFW pumps on the opposite unit are potentially available to supply flow if the turbine-driven AFW pump fails. For an LOSP event affecting a single unit, AC electrical power would be available to the opposite unit. Therefore, if the turbine-driven AFW pump were to fail on the unit affected by the LOSP event, the opposite unit motor-driven AFW pumps would be used. For a dual-unit LOSP event, one or more motor-driven AFW pumps may be available to either or both units depending on the availability of onsite emergency AC power in one of the units. Because the DC electrical power system for each turbine-driven AFW pump is supplied with a battery charger from

each AC electrical power train, the availability of one AC electrical power supply on the non-blackout unit ensures power to operate the turbine-driven AFW pump indefinitely, and the motor-driven AFW pump from the same train is available to support the other unit.

The features discussed above, including the availability of electric power and other support systems on the opposite non-blackout unit, are explicitly modeled in the CNP PRA.

c. For the dominant "slow-SBO" sequence, the onset of core damage is estimated to occur approximately 3.4 hours after the initiation of the event. Since "fast-SBO" sequences are very minor contributors to core damage in the CNP PRA, no analysis of the time to core damage was performed.

d. As described in the responses to SAMA RAI 10.a through SAMA RAI 10.c, nearly all the SBO sequences at CNP are "slow-SBO" sequences. Therefore, nearly all the benefits that would be obtained by providing a backup electrical power supply to the igniters would be obtained by reducing the source terms from "slow-SBO" sequences. Since all the benefits come from "slow-SBO" sequences, there would be no difference in the calculated benefits between using a pre-staged versus a portable backup power source for the hydrogen igniters.

e. The cost estimates for SAMA Items 39 and 40 are from an internal NRC memorandum from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, to John T. Larkins, Executive Director, Advisory Committee on Reactor Safeguards, "RES Proposed Recommendation for Resolving Generic Safety Issue 189: 'Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident'," dated May 13, 2002 (Accession Number ML021340114). In that memorandum, total costs for a two-unit site were estimated as $294,000 or $147,000 per unit. Since a passive ignition system achieves the same goal as a backup power supply to the hydrogen igniters, no separate cost estimates were performed for that SAMA candidate. Benefits for these SAMA candidates were calculated assuming that hydrogen control was perfectly reliable. Therefore, differences between the two options would not change the results of the cost-benefits analysis. In addition, since the cost-benefit analysis using these cost estimates determined that these SAMA candidates were potentially cost-beneficial, performing a more-detailed cost estimate is not expected to change the outcome of the analysis.

SAMA RAI 11:

Figure 6.1 of NUREG/CR-6427 displays the containment fragility curves for CNP. Confirm that this curve is the same as the curves used in the current CNP Level 2 PRA for Units 1 and 2 (October 2003). If not, please explain the differences and their impact on results.

CNP Response to SAMA RAI 11:

The ultimate pressure capacity of the CNP containment used in the PRA is taken from the fragility curve used in the CNP IPE, which is the curve shown in Figure 6.1 of NUREG/CR-6427. The high-confidence, low-probability of failure (HCLPF) value of 36 pounds per square inch – guage (psig) from that curve was used in the Level 2 PRA to determine timing and release characteristics. The probability of containment failure due to phenomenological events, including, but not limited to, induced SGTR, DCH, and steam explosions, is taken from values used or referenced in NUREG/CR-6595. A sensitivity analysis, discussed in the response to RAI 10, used the probability values in NUREG/CR-6427 for phenomenological events causing containment failure. As discussed in the response to RAI 10, use of the NUREG/CR-6427 values would not be expected to change the conclusions of the SAMA analysis.

SAMA RAI 12:

Provide the requested information on the following issues:

a. *I&M review of "reliability issues" appears to have led to identification of 10 candidate SAMAs (SAMAs 185 through 194). Please provide the importance measures for risk increase for these reliability issues from the latest PRA model. Clarify whether the failure rates used in the PRA were modified from generic values to account for this reliability (failure) experience. If not, identify how the CDF and the relative contributions and importance measures might change using these failure data instead of generic failure rates. Also, please explain how both the cost and benefit was estimated for these SAMAs as the descriptions do not specifically identify how the reliability would be improved.*

b. *Several SAMAs involving implementation of procedures were identified in Table F.4-1, at least in part, from the PRA. Some of these were screened out as already being implemented. Please address the following:*

   i. *If these SAMAs were implemented (and the implementation included in the PRA), but the related human error is still important based on the PRA, then another alternative needs to be identified. If this is the case, please identify a new alternative and re-evaluate that alternative.*

ii. *If these SAMAs were implemented after the PRA was completed, please identify when/how the implementation was accomplished. Also, include a discussion of how the implementation changes the CDF based on the importance measures for each related human error.*

c. *SAMAs 5, 9, 10, 12, and 13 all have relatively large benefits and are considered potentially cost beneficial. Each of these has some relationship to the loss of component cooling water (CCW); yet, adding a CCW pump (SAMA 17) has a relatively low benefit and is not cost beneficial. Please explain why SAMA 17 has a relative low benefit, given it can impact other SAMAs with higher benefits. Also, please identify any other improvements considered that would improve the CCW reliability and achieve some of the benefits identified in SAMAs 5, 9, 10, 12, and 13.*

d. *SAMAs 5 and 9 concern adequate charging pump seal cooling, and SAMA 160 concerns emergency core cooling system (ECCS) pump seal cooling. SAMA 160 identified that ECCS self-cooling may be cost-beneficial. However, this does not appear to be considered as an option for the charging pumps. Please address whether self-cooling for the charging pump seals would be cost-beneficial.*

e. *The IPE identified that common mode failures for the safety injection (SI) pumps and compressed air system were significant contributors to CDF. The IPE also identified that a major reason for the significance of common mode failures was due to the modeling approach (i.e., not realistic). SAMAs 141, 142, and 143 appear to identify that the compressed air system did have legitimate common mode issues. However, the SI pumps do not appear to have had a significant common mode issue based on the treatment of SAMA 134 (screened out). Please verify that the significance of the IPE identified SI common mode failures were dominated by the modeling approach (or were corrected elsewhere).*

CNP Response to SAMA RAI 12:

a. The reliability issues that were included as potential SAMA candidates in SAMA Items 185 through 194 were identified by a CNP equipment reliability programmatic review. CNP plant management initiated the programmatic review with the goal of improving equipment reliability by identifying, prioritizing, and resolving equipment performance issues essential to consistent operation of the plant. The programmatic review identified potential issues through reviews of root cause evaluations, condition report evaluations, system health reports, job orders, past plant trips, forced outages, load reductions, and Technical Specification Action Statement entries. The reviews focused on items that would cause a plant trip, a power reduction, or a Technical Specification required shutdown because of operability concerns. Then a scoring process was used to prioritize and to rank the identified equipment performance

issues. The top 10 CNP equipment reliability issues at the time of the performance of the SAMA analysis were conservatively included as potential SAMA candidates. It should be noted that none of the ten items was identified directly from PRA results. However, correlation of the items to PRA impacts is discussed below.

SAMA Item 185 identified the plant initiative to improve the reliability of EDGs. As part of that initiative, diesel control systems were to be improved. Importance measures for EDGs are provided in the response to SAMA RAI 1.c. To model the potential improvement of implementing the initiative, it was assumed that the initiative would result in reducing by a factor of two the probability of EDG failure. Failure data for the EDGs used in the PRA is based on CNP-specific failure data. Cost estimates to implement this item were taken from actual costs associated with projects to implement the CNP equipment reliability improvement initiative.

SAMA Items 186 and 189 address plant initiatives to reduce plant challenges caused by lake debris entering plant raw water systems. The benefits of implementing these items were modeled by assuming that plugging of components cooled by raw water systems would not be possible. Furthermore, it was assumed in the benefits modeling that no plugging of the main condensers would be possible, and that the initiating event frequency for loss of main feedwater events (which include loss of condenser vacuum events) would go to zero. Initiating event frequency data for loss of main feedwater events and strainer plugging data are based on CNP-specific data. Importance measures for strainers are provided in the response to SAMA RAI 1.c. Cost estimates to implement these items were taken from actual costs associated with projects to implement the CNP equipment reliability improvement initiative.

SAMA Items 187, 188, and 190 address plant initiatives to reduce plant challenges caused by component failures resulting in unit trips. The benefits of implementing these items were modeled by assuming that implementation would eliminate loss of main feedwater initiating events entirely. Furthermore, the initiating event frequency of all other reactor trips were assumed to be reduced by one-third. This reduction was based on a review of the CNP-specific events included in the initiating events frequency analysis. Initiating event frequency data for loss of main feedwater events and other reactor trip events are based on CNP-specific data. Cost estimates to implement these items were taken from actual costs associated with projects to implement the CNP equipment reliability improvement initiative.

The CNP equipment reliability programmatic review identified in SAMA Item 191 would establish a program of preventive maintenance and inspection for expansion joints, boots and bellows. Failure of such items would be reflected in the PRA models through a reduction in the internal flooding initiating event frequency. However, as is shown in the IPE analysis, internal flooding is not considered a significant contributor to overall CDF with a contribution of 2E-07 per year. Because internal flooding events are not

significant contributors to CDF, any initiatives to reduce the contribution of such events would result in minimal benefit.

SAMA Item 192 identified the plant initiative to improve the reliability of the turbine-driven AFW pumps. As part of that initiative, the turbine-driven AFW pump trip and throttle valves were to be improved. Importance measures for turbine-driven AFW pumps are provided in the response to SAMA RAI 1.c, which show that the pumps have a low importance to overall CDF. Based on these importance measure results, it was concluded that initiatives to improve the reliability of the turbine-driven AFW pumps would result in a negligible benefit. Failure data for the turbine-driven AFW pumps used in the PRA is based on CNP-specific failure data.

SAMA Item 193 identified the plant initiative to prevent the MSIVs from slowly drifting closed from the full open position during low power, shutdown, and startup conditions. When the valves start to drift closed, operator action is required to open the valves thereby posing a distraction to the operators. If the MSIVs were to drift closed, then a reactor trip could result. However, since the event would be initiated from low power or shutdown conditions, the transient would be less challenging than an event from full power thereby presenting less of a risk. The plant-specific data used in the initiating event frequency analysis for CNP show no events caused by the issue addressed by SAMA Item 193. In the CNP PRA, the only function explicitly modeled for MSIVs is that they close when required to isolate the steam generators. Since the intent of this initiative is to improve the reliability of the MSIVs to stay open, there would be no impact on the function modeled in the PRA. Therefore, implementation of this SAMA candidate is expected to result in negligible benefit, and no further evaluation of costs or benefits was performed.

SAMA Item 194 identified the plant initiative to upgrade main turbine controls. Based on actual costs for plant projects underway to replace the current main turbine control system, costs to implement the item would greatly exceed twice the maximum benefit calculated. Therefore, this SAMA candidate was screened from further consideration.

b. Six SAMA candidates involving procedural or training enhancements were identified from plant-specific sources and are listed in Table F.4-1 of the ER as having been effectively implemented. SAMA Items 2, 6, 175, and 176 were identified from the CNP IPE submittal of 1992 and the associated staff evaluation of 1996. SAMA Items 173 and 174 were identified from the results of the IPE update performed in 1995.

SAMA Item 2 was identified because, at the time of the IPE, the loss of ESW procedure did not provide a clear step to trip the RCPs. For the loss of ESW scenario, the operators needed to diagnose the need to use the loss of CCW procedure where it is directed to trip the reactor and RCPs on entry into the procedure. Because there was no direct procedural

step to enter the loss of CCW procedure, a value of 0.27 was used for the probability of operator action to trip the RCPs following a loss of ESW.

In May 2000, the loss of ESW procedures were changed to add a step directing the operators to transfer immediately to the loss of CCW procedure if the ESW system is intact but flow cannot be restored. This procedure revision was included in the 2001 PRA update, and the associated human error has a Fussell-Vesely importance of less than 0.5 percent.

SAMA Item 6 was identified from the results of the IPE. In the current loss of CCW procedure, the third step directs that RCP seal injection be initiated using the chemical and volume control system (CVCS) cross-tie from the opposite unit. The first two steps direct that the reactor be tripped, and then direct that the RCPs be tripped. Importance analysis results of the 2001 PRA update indicate that the basic event representing operator action failures involving inter-unit cross-tie activities for the CVCS has a Fussell-Vesely importance of slightly less than 4 percent. This event, "CCW-CVCS--MHHE," is evaluated with SAMA Item 172. In this evaluation, all aspects of the HRA associated with the actions, including total time available, total crew workload, available cues, and other higher priority actions, were considered when identifying alternatives to minimize the importance of the operator action. Although this basic event could still be labeled "significant" with respect to the 2001 PRA results, SAMA Item 6 can still be considered as effectively implemented since the concern raised by the general observation is addressed by existing plant procedures.

SAMA Item 173 was identified during the 1995 update of the IPE, specifically during the update of the HRA. At the time of the update, no credit was taken for the use of the foldout pages when calculating cognitive errors for failure to initiate primary bleed and feed. During the 2001 PRA update, no changes were made to the previously calculated HRA. Therefore, the current PRA model still does not credit implementation of the change. However, cognitive human errors that cause failure of primary bleed and feed are considered insignificant contributors to the 2001 PRA model since they have a Fussell-Vesely importance of less than 0.5 percent.

SAMA Item 174 was identified during the 1995 update of the IPE, specifically during the update of the HRA. During that analysis, a general observation was made that some of the human actions analyzed in the IPE were similar to actions credited in the 10 CFR 50 Appendix R fire procedures. In the Appendix R procedures, specific operators are assigned responsibility for specific actions. Specific responsibilities are not delineated in the emergency operating procedures that were analyzed in the IPE update, and it was this difference that precipitated the comment. Other procedures, specifically Operations Head Instructions and Plant Manager Instructions, define crew responsibilities for response to plant events. However, these procedures were not considered in the updated IPE analyses.

Importance analysis results of the 2001 PRA update show three basic events that represent operator action failures involving inter-unit cross-tie activities and that have a Fussell-Vesely importance of greater than 0.5 percent. Each of the three events, "AFW-CROSSTIEHE," "CCW-CVCS--MHHE," and "CCW--XTIE-MHHE," is evaluated with SAMA Item 172. In these evaluations, all aspects of the HRA associated with the actions, including total time available, total crew workload, available cues, and other higher priority actions, were considered when identifying alternatives to minimize the importance of the operator actions. Although three basic events could still be labeled "significant" with respect to the 2001 PRA results, SAMA Item 174 can still be considered as effectively implemented since the concern raised by the general observation is addressed by existing plant procedures.

SAMA Item 175 was identified from the 1992 IPE analyses where it was noted that core decay heat removal via the steam generators provides a substantial benefit with respect to long-term containment pressure response. This SAMA candidate was identified not because of a specific concern with respect to any failure event, but to highlight the complex interactions occurring between decay heat removal systems for ice condenser containment plants. The awareness of these interactions could help when prioritizing recovery actions that may occur post accident. However, these recovery actions are not included in the PRA models.

Since completion of the IPE, significant improvements have been made to the simulator containment models. As a result, simulator response better reflects the expected containment response. Therefore, this SAMA candidate is considered as effectively implemented. Implementation of the SAMA candidate has no impact on the PRA results because the SAMA candidate was not intended to address a specific failure or vulnerability.

SAMA Item 176 was identified from the 1992 IPE submittal, and suggested that operator training emphasize injecting the maximum amount of water from the RWST to the containment prior to switching the ECCS to recirculation. The intent of the SAMA candidate was to ensure that a significant quantity of water is present in the reactor cavity in order to scrub fission products following RPV failure.

As part of the plant restart efforts from 1997 to 2000, significant changes were made to the plant, procedures, training programs, and analyses in order to ensure that sufficient containment recirculation sump inventory is available for ECCS recirculation. As part of the 2001 PRA update, the HRA associated with the switchover to recirculation was re-analyzed using new procedures.

Importance analysis results of the 2001 PRA update show three basic events that represent operator action failures involving the switchover to recirculation, and that have

a Fussell-Vesely importance of greater than 0.5 percent. These three events, "CSR-HIGHDEP-HE," "HPRC-LPR-EXEME," and "HPRA-LPR-CSRME," are evaluated with SAMA Item 172. In addition, SAMA Item 139 evaluated the benefits of automating the recirculation transfer process. In these evaluations, all aspects of the HRA associated with the actions, including total time available, total crew workload, available cues, and dependence with other higher priority actions, were considered. Although three basic events could still be labeled "significant" with respect to the 2001 PRA results, such results are expected, and SAMA Item 176 can still be considered as effectively implemented.

c. The importance measures provided in response to SAMA RAI 1.c show the importance of loss of CCW initiating events with the basic event "IE-CCW," which has a Fussell-Vesely importance of about 5 percent. In a letter from J. E. Pollock (I&M) to the U. S. NRC Document Control Desk, "License Amendment Request for One-Time Extension of Essential Service Water System Allowed Outage Time – Additional Information," dated August 23, 2002, I&M requested a one-time extension of the 72-hour allowed outage time for the ESW system. This submittal described the risk profile of the August 2001 CNP PRA model, and noted that loss of cooling water events are dominated by loss of ESW events, which cause a loss of cooling to the CCW heat exchangers. Loss of ESW events are evaluated separately from loss of CCW events. The importance of loss of ESW events is shown by basic events "IE-ESW4" and "IE-ESW2," which have a Fussell-Vesely importance of about 13 percent and 5 percent, respectively.

The addition of a third CCW pump, as proposed by SAMA Item 17, would address only one specific failure of the many failures that could occur within the CCW system, and would not preclude loss of CCW events. In addition, a third CCW pump would provide no benefit for loss of ESW events. The alternatives proposed in SAMA Items 5, 9, 10, 12, and 13 would provide benefits for loss of ESW as well as loss of CCW events. Furthermore, the alternatives that are proposed in SAMA Items 5, 9, 10, 12, and 13 would provide a means to recover any failure that resulted in failure of either the ESW or CCW systems. The lower benefit for SAMA Item 17 is expected and consistent with the importance measure results.

SAMA Items 5, 9, 10, 12, and 13 all address improvements related to minimizing the occurrence or consequence of RCP seal failure. The alternatives described in these SAMA candidates consider a diverse means to minimize plant risk caused by RCP seal failure events, including those seal failures caused by cooling water system failures.

For SAMA Items 5 and 9, implementation would remove the dependence of the centrifugal charging pumps on cooling water (CCW and ESW) so that a loss of cooling water does not lead to a RCP seal LOCA. Alternatives could include a permanent change that provides centrifugal charging pump cooling from a system other than CCW, or installation of connections and hardware to allow backup cooling to be provided using

temporary connections. In addition, increasing the capacity of the centrifugal charging pumps lube oil sump would be required, so that pump failure on high oil temperatures would be precluded for the time period that it takes to provide the back-up cooling supply.

The option to permanently change centrifugal charging pump cooling to a different system was not evaluated in detail. Such a change would require extensive design and accident analyses to verify system and electrical loading requirements. In addition, flow from the centrifugal charging pump coolers would require installation of additional radiation monitoring equipment because of the potential that a leak in the centrifugal charging pump coolers could allow radioactive isotopes to be released to the environment. Implementation of such an option is likely to be significantly more expensive than using temporary connections to another system. Therefore, implementation of these SAMA candidates evaluated use of temporary connections only.

Because only a short time period would be available from initial loss of cooling water until failure of the operating centrifugal charging pump, a means to provide backup cooling to the centrifugal charging pumps must require a minimum amount of time to implement. Therefore, implementation of this SAMA would require that supply and return piping be provided to the vicinity of the centrifugal charging pumps, so that an excessive amount of time to run hoses for the temporary cooling supply can be avoided.

Implementation of SAMA Item 10 would remove the dependence of RCP thermal barrier cooling on CCW so that loss of cooling water (CCW or ESW) does not lead to a RCP seal LOCA. Alternatives could include a permanent change that provides RCP thermal barrier cooling from a system other than CCW or installation of connections and hardware to allow backup cooling to be provided using temporary connections. Implementation of SAMA Item 10 would be similar to the modifications described for SAMA Item 5.

Implementation of SAMA Item 12 would require that a pump, which can operate independently of an external cooling source, be provided. In addition, motive power for the pump must be independent of AC sources. Options to power the pump could include a direct-drive diesel-powered pump, a diesel-powered electrical generator to supply an AC motor-driven pump, or a DC motor-driven pump supplied by a battery of sufficient capacity to allow operation of the pump for a significant time period.

Implementation of SAMA Item 13 is similar to that of SAMA Item 12 except that an AC-independent power source is not required.

d. As described in Section 6.2 of the CNP UFSAR, the ECCS includes the centrifugal charging pumps. Therefore, the centrifugal charging pumps as well as the SI pumps were considered in the evaluation of benefits for SAMA Item 160. For SAMA Item 5, benefits

were evaluated by assuming that all dependence of the centrifugal charging pumps on CCW was removed. For SAMA Item 160, all dependence of the centrifugal charging pumps and SI pumps on CCW was eliminated to evaluate the benefits. The only improvement considered in the evaluation of benefits for SAMA Item 160 that was not considered in the evaluation of SAMA Item 5 is the removal of dependence of the SI pumps on CCW. As can be seen from Table F.4-2 in the ER, the benefits for both SAMA Items 5 and 160 are similar, with the benefits for SAMA Item 160 slightly higher because an additional dependence has been removed. However, it should be noted that in addition to providing self-cooled pump seals, a means to cool the lubricating oil for the pumps would be required. Therefore, a modification that only replaced ECCS pump seals with a self-cooling design would provide little benefit without additional modifications to provide backup cooling for pump lubricating oil.

e. SAMA Items 141, 142, and 143 should not be construed to mean that the compressed air system had legitimate common mode failure issues. As shown in Table 3.4-4 of the IPE submittal (Reference 1), each plant air compressor had a Fussell-Vesely importance of over 10 percent. Common cause failures of the compressed air system are shown on the same table to have an importance of about 3 percent. The Staff Evaluation Report on the IPE was issued in a letter dated September 6, 1996 (Reference 2). From the importance measures provided in response to SAMA RAI 1.c, the compressors have an importance of less than 1 percent in the 2001 PRA update. The air compressors were identified as significant to the 2001 PRA model because they had an importance of greater than 0.5 percent. SAMA Item 142 was identified as a result of the IPE findings, which resulted in a plant modification that installed new air compressors. SAMA Item 143 was identified from generic industry sources, and was considered as effectively implemented because backup nitrogen bottles are permanently installed to support operation of the steam generator power operated relief valves.

As described on page 7-1 of the IPE submittal (Reference 1), common cause failures dominate the IPE models because all common cause failures within a system, whether for pumps, valves, or any other component, are represented in the model with a single basic event, including common cause failures of the SI pumps. In the 2001 update, common cause failure combinations are explicitly represented in the model with individual basic events for each failure combination. The improved representation of common cause failures using multiple basic events has resulted in a significant reduction in the importance of common cause failures for the SI pumps. Therefore, SAMA Item 134 was screened out.

References

1. Letter from E. E. Fitzpatrick, I&M, to NRC Document Control Desk, "Individual Plant Examination Submittal; Response to Generic Letter 88-20," AEP:NRC:1082E, dated May 1, 1992.
2. Letter from J. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Review of the D. C. Cook Individual Plant Examination Submittal – Internal Events (TAC Nos. M74398 and M74399)," dated September 6, 1996.

Reference Tables for Response to Request for Additional Information

This attachment provides the tables referred to in Attachment 1, including the following:

- Table 1, Core Damage Frequency (CDF) Importance Analysis
- Table 2, Large Early Release Frequency (LERF) Importance Analysis
- Table 3, Source Terms Associated with Each Release Category
- Table 4, Severe Accident Mitigation Alternative (SAMA) Items Potentially Beneficial to Both Units
- Table 5, Cost-Benefit Estimates for Severe Accident Mitigation Alternatives (SAMA) Candidates Screened in Phase 2
- Table 6, Summary of Off-Site Consequence Results for Each Release Mode, Sensitivity Analysis of Revised Containment Failure Probability Values
- Table 7, Sensitivity Evaluation of Revised Containment Failure Probabilities on Selected Severe Accident Mitigation Alternatives (SAMA) Items

Table 1, Core Damage Frequency (CDF) Importance Analysis					
Rank	Event Name	Point Estimate	F-V Importance	Risk Achievement	Risk Reduction
2	EH1	5.800E-001	2.858E-001	1.21	1.400
3	IE-LSP	2.800E-002	2.321E-001	9.06	1.302
6	IE-SLO	2.500E-003	1.705E-001	69.02	1.206
7	IE-DLSP-1	8.600E-003	1.425E-001	17.42	1.166
8	IE-TRA	1.300E+000	1.325E-001	0.97	1.153
9	IE-ESW4	3.650E+002	1.287E-001	0.87	1.148
11	SBDG-----1ABFR	1.078E-001	1.172E-001	1.97	1.133
12	SADG-----1CDFR	1.078E-001	1.162E-001	1.96	1.132
15	BFLSTRNPL-CCF1-4	6.059E-006	1.122E-001	18524.07	1.126
16	HI1-FAILURE-HE	1.000E+000	9.571E-002	1.00	1.106
17	OLI---13B-EHHE	5.000E-001	9.434E-002	1.09	1.104
18	SBO-SINGLE-XH1	3.620E-002	8.801E-002	3.34	1.097
20	SBO---DUAL-XH1	9.420E-002	6.991E-002	1.67	1.075
21	TRAIN-A-OUTAGE	4.100E-002	6.279E-002	2.47	1.067
22	TRAIN-B-OUTAGE	4.100E-002	5.436E-002	2.27	1.057
23	LERF2	2.000E-001	5.409E-002	1.22	1.057
25	IE-ESW2	3.650E+002	4.968E-002	0.95	1.052
26	SN1	1.990E-002	4.823E-002	3.38	1.051
27	IE-CCW	3.650E+002	4.631E-002	0.95	1.049
29	RRI---CCW-EHHE	1.301E-002	4.402E-002	4.34	1.046
30	BPM-RUNFR-CCF12	2.320E-005	4.032E-002	1738.91	1.042
31	CCW-CVCS--MHHE	2.001E-002	3.834E-002	2.88	1.040
32	BPM-RUNFR-CCF34	2.320E-005	3.476E-002	1499.17	1.036
33	BPM-RUNFR-CCF13	2.320E-005	3.451E-002	1488.45	1.036
34	DGFR------CCF1-4	1.540E-003	3.382E-002	22.93	1.035
35	CSR-HIGHDEP-HE	5.000E-001	3.262E-002	1.03	1.034
36	SBDG-----1ABFS	7.659E-003	3.182E-002	5.12	1.033
37	SADG-----1CDFS	7.659E-003	3.159E-002	5.09	1.033
38	SN6	4.501E-002	3.016E-002	1.64	1.031
39	RCC---EXE-EHHE	7.908E-002	3.002E-002	1.35	1.031
41	FBCV--SI151WFC	9.999E-004	2.786E-002	28.84	1.029
42	FACV--SI151EFC	9.999E-004	2.786E-002	28.83	1.029
43	LERF1	5.000E-002	2.733E-002	1.52	1.028
44	UNIT-1-E-CCW	5.000E-001	2.554E-002	1.03	1.026
45	UNIT-1-W-CCW	5.000E-001	2.543E-002	1.03	1.026
46	OA2----E3-MHHE-L	5.003E-002	2.278E-002	1.43	1.023
47	OA1--E3CD-MHHE-M	5.907E-003	2.263E-002	4.81	1.023
48	RRI---AFW-EHHE	6.504E-003	2.143E-002	4.27	1.022
49	BBPM---1PP7WTM	3.300E-003	2.134E-002	7.45	1.022
50	SDGFANABCDEFGHFO	2.430E-004	2.124E-002	88.39	1.022
51	BAMV-ESWWESTHE	3.000E-003	2.096E-002	7.97	1.021
52	BAMV-ESWEASTHE	3.000E-003	2.088E-002	7.94	1.021

Table 1, Core Damage Frequency (CDF) Importance Analysis					
Rank	Event Name	Point Estimate	F-V Importance	Risk Achievement	Risk Reduction
53	EPORVMANOPENHE	1.001E-001	2.070E-002	1.19	1.021
54	CCW-----COG-HE	1.501E-003	2.045E-002	14.61	1.021
55	BAPM---1PP7ETM	3.300E-003	2.030E-002	7.13	1.021
56	CCW--XTIE-MHHE	1.001E-001	1.878E-002	1.17	1.019
57	CCW-REPAIR--HE	6.000E-001	1.878E-002	1.01	1.019
58	BPM-RUNFR-CCF23	2.320E-005	1.821E-002	785.95	1.019
59	BPM-RUNFR-CCF14	2.320E-005	1.817E-002	784.15	1.019
60	PP10PR----CCF12	7.439E-007	1.794E-002	24112.88	1.018
61	AFW-OPENDOORHE	5.406E-002	1.785E-002	1.31	1.018
62	AAFN--HVDGS2FS	4.021E-003	1.639E-002	5.06	1.017
63	ABFN--HVDGS1FS	4.021E-003	1.639E-002	5.06	1.017
64	ABFN--HVDGX1FS	4.021E-003	1.630E-002	5.04	1.017
65	AAFN--HVDGX2FS	4.021E-003	1.630E-002	5.04	1.017
66	HPRC-LPR-EXEME	2.903E-004	1.624E-002	56.92	1.017
67	AFW-CROSSTIEHE	1.001E-001	1.551E-002	1.14	1.016
68	Q-CC-SI-SIGNAL	2.000E-004	1.548E-002	78.38	1.016
70	DGFR------CCF12	7.698E-004	1.516E-002	20.69	1.015
71	BAPM----PP7EPR	6.081E-004	1.511E-002	25.84	1.015
72	BBPM----PP7WPR	6.081E-004	1.501E-002	25.67	1.015
73	SBDG-----1ABTM	3.574E-003	1.359E-002	4.79	1.014
74	SADG-----1CDTM	3.574E-003	1.351E-002	4.77	1.014
75	RPL	8.400E-001	1.293E-002	1.00	1.013
76	IE-SLB-5	8.500E-004	1.275E-002	15.99	1.013
77	AABS-MS-T11DHE	1.000E+000	1.210E-002	1.00	1.012
78	AACB--T11D12FO	3.010E-003	1.210E-002	5.01	1.012
79	AACB---T11D8FC	3.010E-003	1.210E-002	5.01	1.012
80	P-ELECTRIC--CM	4.299E-006	1.207E-002	2809.72	1.012
81	ABCB---T11A9FO	3.010E-003	1.202E-002	4.98	1.012
82	ABCB--T11A11FC	3.010E-003	1.202E-002	4.98	1.012
83	ABBS-MS-T11AHE	1.000E+000	1.202E-002	1.00	1.012
84	BPM-RUNFR-CCF24	2.320E-005	1.159E-002	500.70	1.012
85	IRH108FO--CCF12	2.000E-004	1.147E-002	58.35	1.012
87	CBPM---PP10WTM	3.487E-003	1.105E-002	4.16	1.011
88	CAPM---PP10ETM	3.487E-003	1.096E-002	4.13	1.011
89	DGFR------CCF123	5.160E-004	1.050E-002	21.35	1.011
90	SBO-SINGLE-XH2	4.451E-002	1.049E-002	1.23	1.011
92	SBO---DUAL-XH2	1.130E-001	1.032E-002	1.08	1.010
93	IPP35PS---CCF12	1.780E-004	1.020E-002	58.30	1.010
94	DGFR------CCF124	5.160E-004	1.015E-002	20.67	1.010
95	IE-SGTR-2	1.499E-003	1.009E-002	7.72	1.010
96	IE-SGTR-3	1.499E-003	1.009E-002	7.72	1.010
97	IE-SGTR-1	1.499E-003	1.002E-002	7.67	1.010
98	IE-SGTR-4	1.499E-003	1.002E-002	7.67	1.010

Table 2, Large Early Release Frequency (LERF) Importance Analysis					
Rank	Event Name	Point Estimate	F-V Importance	Risk Achievement	Risk Reduction
1	LERF2	2.000E-001	4.693E-001	2.88	1.884
2	IE-LSP	2.800E-002	3.031E-001	11.52	1.435
4	LERF1	5.000E-002	2.371E-001	5.51	1.311
5	IE-DLSP-1	8.600E-003	2.191E-001	26.26	1.281
7	SBDG-----1ABFR	1.078E-001	1.641E-001	2.36	1.196
8	SADG-----1CDFR	1.078E-001	1.636E-001	2.35	1.196
11	HI1-FAILURE-HE	1.000E+000	1.574E-001	1.00	1.187
12	SBO-SINGLE-XH1	3.620E-002	1.519E-001	5.05	1.179
13	SBO---DUAL-XH1	9.420E-002	1.204E-001	2.16	1.137
14	EH1	5.800E-001	1.152E-001	1.08	1.130
15	EPORVMANOPENHE	1.001E-001	1.122E-001	2.01	1.126
17	SN1	1.990E-002	8.274E-002	5.07	1.090
18	IE-SLO	2.500E-003	7.660E-002	31.56	1.083
19	IE-TRA	1.300E+000	7.091E-002	0.98	1.076
20	LERF5	1.300E-001	6.086E-002	1.41	1.065
22	BFLSTRNPL-CCF1-4	6.059E-006	5.817E-002	9601.00	1.062
24	X-CM---OME42PS	7.999E-002	5.217E-002	1.60	1.055
25	X-CM--1OME41PR	4.687E-002	5.196E-002	2.06	1.055
26	SN6	4.501E-002	5.176E-002	2.10	1.055
27	DGFR------CCF1-4	1.540E-003	4.967E-002	33.21	1.052
28	IE-ESW4	3.650E+002	4.860E-002	0.95	1.051
29	IE-ISL	6.200E-007	4.563E-002	73597.34	1.048
30	SBDG-----1ABFS	7.659E-003	4.457E-002	6.77	1.047
31	OIB-DYNAM-EHHE	4.000E-001	4.438E-002	1.07	1.046
32	SADG-----1CDFS	7.659E-003	4.438E-002	6.75	1.046
33	OLI---13B-EHHE	5.000E-001	4.183E-002	1.04	1.044
34	RCC---EXE-EHHE	7.908E-002	4.131E-002	1.48	1.043
35	IE-SGTR-3	1.499E-003	4.045E-002	27.94	1.042
36	IE-SGTR-2	1.499E-003	4.045E-002	27.94	1.042
37	IE-SGTR-1	1.499E-003	4.007E-002	27.69	1.042
38	IE-SGTR-4	1.499E-003	4.007E-002	27.69	1.042
39	TRAIN-B-OUTAGE	4.100E-002	3.539E-002	1.83	1.037
40	SDGFANABCDEFGHFO	2.430E-004	3.121E-002	129.40	1.032
41	X-CM---OME42PR	4.687E-002	3.030E-002	1.62	1.031
42	X-CM--2OME41PS	7.999E-002	2.745E-002	1.32	1.028
43	LERF6	2.800E-001	2.699E-002	1.07	1.028
44	TRAIN-A-OUTAGE	4.100E-002	2.549E-002	1.60	1.026
45	X-CM--1OME41PS	7.999E-002	2.486E-002	1.29	1.025
46	CSR-HIGHDEP-HE	5.000E-001	2.290E-002	1.02	1.023
47	AAFN--HVDGS2FS	4.021E-003	2.284E-002	6.66	1.023
48	ABFN--HVDGS1FS	4.021E-003	2.284E-002	6.66	1.023
49	ABFN--HVDGX1FS	4.021E-003	2.275E-002	6.64	1.023

Table 2, Large Early Release Frequency (LERF) Importance Analysis					
Rank	Event Name	Point Estimate	F-V Importance	Risk Achievement	Risk Reduction
50	AAFN--HVDGX2FS	4.021E-003	2.275E-002	6.64	1.023
51	AFW-CROSSTIEHE	1.001E-001	2.234E-002	1.20	1.023
52	DGFR------CCF12	7.698E-004	2.158E-002	29.01	1.022
53	AFW-OPENDOORHE	5.406E-002	2.066E-002	1.36	1.021
54	XCOM-CSR3	4.997E-001	1.931E-002	1.02	1.020
55	OA1--E3CD-MHHE-M	5.907E-003	1.890E-002	4.18	1.019
56	SBDG-----1ABTM	3.574E-003	1.888E-002	6.27	1.019
57	SADG-----1CDTM	3.574E-003	1.880E-002	6.24	1.019
58	X-CM-1OME41-TM	6.000E-002	1.859E-002	1.29	1.019
59	RRI---CCW-EHHE	1.301E-002	1.748E-002	2.33	1.018
60	SBO-SINGLE-XH2	4.451E-002	1.682E-002	1.36	1.017
61	AACB--T11D12FO	3.010E-003	1.681E-002	6.57	1.017
62	AABS-MS-T11DHE	1.000E+000	1.681E-002	1.00	1.017
63	AACB---T11D8FC	3.010E-003	1.681E-002	6.57	1.017
64	ABCB--T11A11FC	3.010E-003	1.679E-002	6.56	1.017
65	ABCB---T11A9FO	3.010E-003	1.679E-002	6.56	1.017
66	ABBS-MS-T11AHE	1.000E+000	1.679E-002	1.00	1.017
67	BPM-RUNFR-CCF12	2.320E-005	1.605E-002	692.84	1.016
68	SBO---DUAL-XH2	1.130E-001	1.580E-002	1.12	1.016
69	Q-CC-SI-SIGNAL	2.000E-004	1.499E-002	75.91	1.015
70	DGFR------CCF123	5.160E-004	1.495E-002	29.96	1.015
71	IE-ESW2	3.650E+002	1.482E-002	0.99	1.015
72	DNPT-----PP4PS	1.865E-002	1.452E-002	1.76	1.015
73	DGFR------CCF124	5.160E-004	1.437E-002	28.83	1.015
74	IE-CCW	3.650E+002	1.390E-002	0.99	1.014
75	BPM-RUNFR-CCF34	2.320E-005	1.360E-002	587.20	1.014
76	BPM-RUNFR-CCF13	2.320E-005	1.311E-002	566.23	1.013
77	IE-SLB-5	8.500E-004	1.308E-002	16.37	1.013
78	CCW-CVCS--MHHE	2.001E-002	1.302E-002	1.64	1.013
79	X-CM--OME42-TM	2.000E-002	1.293E-002	1.63	1.013
80	P-ELECTRIC--CM	4.299E-006	1.293E-002	3007.97	1.013
81	FBCV--SI151WFC	9.999E-004	1.250E-002	13.49	1.013
82	FACV--SI151EFC	9.999E-004	1.250E-002	13.49	1.013
83	SGTR-E3-COG-HE	3.308E-003	1.212E-002	4.65	1.012
85	IBMV--IMO128FO	1.082E-003	1.104E-002	11.19	1.011
86	IBMV--ICM129FO	1.082E-003	1.104E-002	11.19	1.011

Table 3, Source Terms Associated with Each Release Category

Source Term Category (STC)	General Emergency Declared (Hours after Scram)	Plume Start Time (Hours)	Plume Duration (Hours)	Release Fractions								
				Xe/Kr	I	Sr	Ru(Mo)	Cs	Ba	La	Ce	Te
STC-1	17.1	20.0	2.0	0.88	0.0082	1.4E-4	5.0E-3	0.0045	0.00184	4.0E-5	2.4E-4	0.020
		22.0	2.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total				0.88	0.0082	1.4E-4	5.0E-3	0.0045	0.00184	4.0E-5	2.4E-4	0.020
STC-2	1.01	1.5	0.5	1.0	0.70	0.00	0.0002	0.73	0.008	0.000	0.00	0.13
		2.0	2.0	0.0	0.15	0.05	0.0082	0.02	0.022	0.008	0.10	0.33
		4.0	20.0	0.0	0.15	0.00	0.0000	0.22	0.002	0.000	0.00	0.02
Total				1.0	1.00	0.05	0.0084	0.97	0.032	0.008	0.10	0.48
STC-3	0.81	1.0	1.0	0.030	1.5E-4	3E-7	1.1E-5	1.4E-4	3.3E-6	4.1E-8	9.5E-8	3.2E-5
		2.0	6.0	0.060	5E-6	0.0	0.0	5E-6	0.0	0.0	0.0	0.0
		8.0	16.0	0.025	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total				0.115	1.55E-4	3E-7	1.1E-5	1.45E-4	3.3E-6	4.1E-8	9.5E-8	3.2E-5
STC-4	2.83	4.0	20.0	0.55	0.0025	0.0002	0.0036	0.0015	0.0018	0.00011	0.0006	0.007
		24.0	4.0	0.45	0.0000	0.0000	0.0000	0.0000	0.0000	0.00000	0.0000	0.000
		42.0	3.0	0.00	0.0225	0.0010	0.0012	0.0120	0.0012	0.00003	0.0012	0.033
Total				1.00	0.0250	0.0012	0.0048	0.0135	0.0030	0.00014	0.0018	0.040
STC-5	2.14	3.0	3.0	0.2	0.015	0.0000	0.0020	0.010	0.0008	0.20E-4	0.0001	0.005
		6.0	12.0	0.8	0.000	0.0001	0.0045	0.002	0.0005	0.62E-4	0.0002	0.004
		33.0	3.0	0.0	0.055	0.0027	0.0011	0.044	0.0021	1.00E-4	0.0047	0.051
Total				1.0	0.070	0.0028	0.0076	0.056	0.0034	1.82E-4	0.0050	0.060
STC-6	2.82	7.0	0.5	0.55	0.024	0.0020	0.0024	0.012	0.0015	2.00E-4	0.004	0.070
		7.5	12.5	0.37	0.041	0.0002	0.0001	0.008	0.0008	3.0E-5	0.001	0.030
		20.0	24.0	0.08	0.005	0.0000	0.0000	0.050	0.0003	5E-6	0.000	0.008
Total				1.00	0.070	0.0022	0.0025	0.070	0.0026	2.35E-4	0.005	0.108
STC-7	2.82	15.0	3.0	0.50	0.024	3.3E-5	9.0E-5	0.003	0.00010	2.3E-6	5.4E-5	0.018
		18.0	6.0	0.35	0.036	5.0E-6	0.0	0.005	0.00007	1.0E-7	3.0E-6	0.010
		24.0	24.0	0.09	0.004	0.0	0.0	0.040	0.00014	0.0	0.0	0.013
Total				0.94	0.064	3.8E-5	9.0E-5	0.048	0.00031	2.4E-6	5.7E-5	0.041
STC-8	2.83											
Total		0	0	0	0	0	0	0	0	0	0	0

Table 4, Severe Accident Mitigation Alternative (SAMA) Items in which One Change Could Potentially Provide Benefit to Both Units	
SAMA Item Number	**Potential Improvement**
25	Stage backup fans in switchgear rooms.
26	Provide redundant train of ventilation to 480V board room.
27	Implement procedures for temporary heating, ventilation, and air conditioning (HVAC).
28	Provide backup ventilation for the emergency diesel generator (EDG) rooms that could be used if the room normal HVAC supply fails.
39	Create/enhance hydrogen igniters with independent power supply. (GSI-189)
41	The action to turn on hydrogen igniters fails frequently due to the time needed to remotely turn off the ice condenser air handling units, as committed to during the original installation of the hydrogen igniter system. This commitment will be investigated and removed if justifiable.
68	Provide alternate battery charging capability.
72	Create a cross-unit tie for EDG fuel oil.
73	Develop procedures to repair or change out failed 4KV breakers.
84	Develop procedures for use of pressurizer vent valves during steam generator tube rupture sequences.
96	Increase frequency of valve leak testing.
100	Revise emergency operating procedures to improve interfacing systems loss of coolant accident (ISLOCA) identification.
101	Revise ISLOCA procedure to specifically address the ISLOCA sequence with the frequency that was dominant in Rev. 1 of the Probabilistic Risk Assessment.
115	Provide portable generators to be hooked in to the turbine-driven auxiliary feedwater pump, after battery depletion.
127	Create the ability to manually align emergency core cooling system recirculation.
141	Replace old air compressors with more reliable ones.
154	Make procedural changes only for the reactor coolant system depressurization option.
172	Enhance training for important operator actions.
184	Provide a means to ensure RCP seal cooling so that RCP seal loss of coolant accidents are precluded for station blackout events.

Table 5, Cost-Benefit Estimates for Severe Accident Mitigation Alternatives (SAMA) Candidates Screened in Phase 2

SAMA Candidates From Table F.4-2 of the Environmental Report (ER)	Total Averted Costs (APE + AOC + AOE +AOSC) From Table F.4-2 of the ER	Significant Costs Not Considered? (Yes/No)	Cost of Implementing SAMA Candidates	Net Present Value of Implementing SAMA Candidates	Potentially Cost-Beneficial? (Yes/No)
24,33,117	$11,437	Yes	$70,000	-$58,563	No
17	$87,880	Yes	$500,000	-$412,120	No
34,35,53	$0	Yes	$90,000	-$90,000	No
41	$9,923	Yes	$40,000	-$30,077	No
49	$765,463	Yes	$2,180,000	-$1,414,537	No
72	Negligible	Yes	Not Evaluated	Negligible	No
73	$20,423	No	$70,000	-$49,577	No
79,80	$42,811	No	$140,000	-$97,189	No
84,85	$19,022	Yes	$90,000	-$70,978	No
94,103,166,170	Negative	Yes	$50,000	-$50,000	No
95,96,168,169	$95,885	Yes	$530,000	-$434,115	No
100	$1,054	Yes	$20,000	-$18,946	No
108	$100,022	No	$2,532,000	-$2,431,978	No
68,115	$59,865	Yes	$294,000	-$234,135	No
117,123	$17,370	Yes	$70,000	-$52,630	No
124,125,134	$299,185	No	$2,000,000	-$1,700,815	No
126	Negligible	Yes	Not Evaluated	Negligible	No
127	$39,169	Yes	$100,000	-$60,831	No
139	$220,769	Yes	$795,000	-$574,231	No
141	$28,591	Yes	$110,000	-$81,409	No
144,145	$15,130	Yes	$70,000	-$54,870	No
149,153,154	$315,931	Yes	$1,090,000	-$774,069	No
157	$86,844	Yes	$700,000	-$613,156	No
162	Not Evaluated	Yes	Not Evaluated	Not Evaluated	No
163	Not Evaluated	Yes	Not Evaluated	Not Evaluated	No
167	$11,802	Yes	$50,000	-$38,198	No
171,186,189	$221,837	No	$2,535,000	-$2,313,163	No
172.1	Negligible	Yes	Not Evaluated	Negligible	No

Table 5, Cost-Benefit Estimates for Severe Accident Mitigation Alternatives (SAMA) Candidates Screened in Phase 2

SAMA Candidates From Table F.4-2 of the Environmental Report (ER)	Total Averted Costs (APE + AOC + AOE +AOSC) From Table F.4-2 of the ER	Significant Costs Not Considered? (Yes/No)	Cost of Implementing SAMA Candidates	Net Present Value of Implementing SAMA Candidates	Potentially Cost-Beneficial? (Yes/No)
172.19,172.20	Negligible	Yes	Not Evaluated	Negligible	No
172.2	Negligible	Yes	Not Evaluated	Negligible	No
172.3,172.9,172.21	$65,703	Yes	$220,000	-$154,297	No
172.4,172.13,172.14,172.15,172.24	$90,021	Yes	$220,000	-$129,979	No
172.5,172.17,172.23	$55,192	Yes	$200,000	-$144,808	No
172.6	$919	Yes	$10,000	-$9,081	No
172.7,172.8	$20,804	Yes	$50,000	-$29,196	No
172.10	Negligible	Yes	Not Evaluated	Negligible	No
172.11,172.12	Negligible	Yes	Not Evaluated	Negligible	No
172.16	Negligible	Yes	Not Evaluated	Negligible	No
172.18	Negligible	Yes	Not Evaluated	Negligible	No
172.22	$4,378	Yes	$40,000	-$35,622	No
172.25,172.26	$24,086	Yes	$60,000	-$35,914	No
177,179	Negligible	Yes	Not Evaluated	Negligible	No
185	$500,300	No	$3,176,000	-$2,675,700	No
187,188,190	$100,022	No	$341,000	-$240,978	No
191	Negligible	Yes	Not Evaluated	Negligible	No
192	Negligible	Yes	Not Evaluated	Negligible	No
193	Negligible	Yes	Not Evaluated	Negligible	No

Table 6, Summary of Off-Site Consequence Results for Each Release Mode, Sensitivity Analysis of Revised Containment Failure Probability Values

Source Term Category (STC)	Description	STC Frequency (per year)	Conditional Person-Sv Offsite	Conditional Person-REM Offsite	Expected Person-REM/yr Offsite	Expected Property Costs ($)
1	Containment bypassed with noble gases plus up to 1% of the volatiles released	1.781E-06	3.71E+03	3.71E+05	6.608E-01	5.29E+02
2	Containment bypassed with noble gases and more than 10% of the volatiles released	1.294E-06	9.67E+04	9.67E+06	1.251E+01	1.72E+04
3	Containment failure prior to vessel failure with noble gases and less than 1/10% of the volatiles released (containment isolation impaired)	5.795E-09	1.94E+02	1.94E+04	1.124E-04	2.56E-02
4	Containment failure prior to vessel failure with noble gases and up to 10% of the volatiles released (containment isolation impaired)	6.858E-09	8.39E+03	8.39E+05	5.754E-03	6.20E+00
5	Containment failure prior to vessel failure with the noble gases and more than 10% of the volatile fission products released (containment isolation impaired)	1.150E-09	1.74E+04	1.74E+06	2.001E-03	3.13E+00
6	Early containment failure with noble gases and up to 10% of the volatiles released (containment failure within six hours of vessel failure; containment not bypassed; isolation successful)	1.741E-05	2.16E+04	2.16E+06	3.761E+01	6.49E+04
7	Late containment failure with noble gases and up to 10% of the volatiles released (containment failure greater than six hours after vessel failure; containment not bypassed; isolation successful)	5.891E-06	1.58E+04	1.58E+06	9.308E+00	1.43E+04
8	No containment failure (leakage only, successful maintenance of containment integrity; containment not bypassed; isolation successful)	2.347E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	Total	**4.986E-05**			**60.09 REM**	**$96,944**

Table 7, Sensitivity Evaluation of Revised Containment Failure Probabilities on Selected Severe Accident Mitigation Alternatives (SAMA) Items*

Description	Base Case Using NUREG/CR-6427	SAMA Items 39 and 40	SAMA Item 41
Core Damage Frequency After Enhancements		4.986E-05	4.967E-05
Total Expected Person-REM per year Offsite ($F_A D_{PA}$)	60.09	42.58	59.66
Total Expected Offsite Property Damage $ per year Offsite ($F_A P_{DA}$)	$96,944	$64,549	$96,217
Averted Public Exposure (APE)	$1,293,484	$377,070	$9,258
Averted Offsite Property Damage Costs (AOC)	$1,043,399	$348,670	$7,821
Averted Immediate Occupational Exposure Costs (W_{IO})	$3,542	$0	$13
Averted Long-Term Occupational Exposure Costs (W_{LTO})	$15,437	$0	$59
Total Averted Occupational Exposure Costs (AOE)	$18,979	$0	$72
Averted Cleanup and Decontamination Costs (U_{CD})	$578,896	$0	$2,206
Averted Replacement Power Costs (U_{RP})	$482,902	$0	$1,840
Averted Onsite Costs (AOSC)	$1,061,798	$0	$4,046
Total Averted Costs (APE + AOC + AOE +AOSC)	**$3,417,660**	**$725,740**	**$21,198**

* See Appendix F, Table F.4-2 and Table F.6-1, of the Applicant's Environmental Report - Operating License Renewal Stage, for description of the selected SAMA candidates and original SAMA analysis and sensitivity results.

Reference Figures for Response to Request for Additional Information

This attachment provides the figures referred to in Attachment 1.

Figure 1, Containment Event Tree (CET) for October 2003 Probabilistic Risk Assessment Update

Figure 1, Containment Event Tree (CET) for October 2003 Probabilistic Risk Assessment Update