From: Samuel Hernandez-Quinones
To: Robert Palla
Date: 08/02/2006 8:21:55 AM
Subject: Fwd: VY License Renewal SAMA RAI Response

Bob,

Attached is the SAMA RAI response PDF file.

Sam

>>> "Hamer, Mike" <mhamer@entergy.com> 08/01/2006 4:25 PM >>>
Attached is Part 1 of the VYNPS Severe Accident Mitigation Alternatives (SAMA) responses to the RAIs received on June 1, 2006. Per our telecom with the NRC Environmental Group on July 27, 2006, and as detailed in the cover letter to this correspondence, we will provide the responses to the remaining questions no later than September 30, 2006.

<<BVY 06-071 SAMA RAI Responses - Part 1.PDF>>
Please call me if you have any questions.

Mike Hamer
Licensing Specialist
Entergy Nuclear Vermont Yankee
(802) 258-4226
mhamer@entergy.com

CC: Richard Emch

Mail Envelope Properties (44D098E3.F20 : 24 : 35376)

Subject:	Fwd: VY License Renewal SAMA RAI Response
Creation Date	08/02/2006 8:21:55 AM
From:	Samuel Hernandez-Quinones
Created By:	SHQ@nrc.gov

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Entergy

Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

August 1, 2006

Docket No. 50-271
BVY 06-071
TAC No. MC 9670

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

Reference:
1. Letter, Entergy to USNRC, "Vermont Yankee Nuclear Power Station, License No. DPR-28, License Renewal Application," BVY 06-009, dated January 25, 2006.
2. Letter, USNRC to Entergy, "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the Vermont Yankee Nuclear Power Station, NVY 06-068, dated June 1, 2006.

Subject: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
License Renewal Application, Amendment 7

On January 25, 2006, Entergy Nuclear Operations, Inc. and Entergy Nuclear Vermont Yankee, LLC (Entergy) submitted the License Renewal Application (LRA) for the Vermont Yankee Nuclear Power Station (VYNPS) as indicated by Reference 1. In Reference 2, the NRC provided a Request for Additional Information (RAIs) pertaining to Severe Accident Mitigation Alternatives (SAMA).

During the development of the recent Mitigating Systems Performance Index (MSPI), changes to the Vermont Yankee Probabilistic Safety Assessment (PSA) model were identified. A sensitivity study is underway to determine the impact of these changes on the SAMA analysis. Based on this, Attachment 1 includes responses to RAIs that are not expected to be impacted by the model changes. Responses to the remaining RAIs and any changes to the License Renewal Application will be submitted to the Staff by September 30, 2006.

Should you have any questions concerning this letter, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 1, 2006.

Sincerely,

Ted A. Sullivan
Site Vice President
Vermont Yankee Nuclear Power Station

Attachment 1
cc: See next page

cc: Mr. James Dyer, Director
U.S. Nuclear Regulatory Commission
Office O5E7
Washington, DC 20555-00001

Mr. Samuel J. Collins, Regional Administrator
U.S. Nuclear Regulatory Commission, Region 1
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Jack Strosnider, Director
U.S. Nuclear Regulatory Commission
Office T8A23
Washington, DC 20555-00001

Mr. Jonathan Rowley, Senior Project Manager
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
MS-O-11F1
Rockville, MD 20853

Mr. Richard Emch
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
MS-O-11F1
Rockville, MD 20853

Mr. James J. Shea, Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop O8G9A
Washington, DC 20555

USNRC Resident Inspector
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157 *(for mail delivery)*
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Attachment 1

Vermont Yankee Nuclear Power Station

License Renewal Application

Amendment 7

REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE ANALYSIS OF SEVERE ACCIDENT MITIGATION ALTERNATIVES (SAMAs)

FOR THE VERMONT YANKEE NUCLEAR POWER STATION (VYNPS)

SUBMITTAL 1 OF 2

DOCKET NO. 50-271

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Tables

NRC RAI 1

The SAMA analysis is said to be based on the most recent version of the VYNPS Probabilistic Safety Analysis (PSA) (VY04R1). Provide the following information regarding these PSA models:

a. Table E.1-8 indicates that the core damage frequency (CDF) associated with station blackout sequences (Classes IBE and IBL) is 1.2E-06 per year. This is considerably more than the CDF due to loss of offsite power (LOOP) (7.2E-7 per year in Table E.1-2) and is comparable to the total CDF due to LOOP and loss of alternating current (ac) bus initiating events. Provide the station blackout (SBO) CDF frequency along with its derivation.

b. The VYNPS extended power uprate (EPU) application and response to EPU requests for additional information indicate that the VY02R6 model had a CDF of 7.77E-06 per year and that this increased to 8.1E-06 for EPU conditions. This is different from the current value of 5E-06. Provide a summary of the major Levels 1 and 2 PSA versions and their CDFs from the individual plant examination (IPE) to the present, including the version reviewed by the Boiling Water Reactors Owners Group (BWROG). Also, indicate the major changes to each version from the prior version and the major reasons for changes in the CDF.

c. Discuss the overall conclusion of the BWROG peer review relative to the use of the VYNPS PSA.

d. Internal flooding initiating events are the dominant contributors to CDF at VYNPS. Briefly describe the internal flooding analysis and its evolution, including internal and external peer reviews, the results of these reviews, and any subsequent model updates. It is noted that the BWROG A and B facts and observations did not include internal flooding. Clarify whether the internal flooding analysis was covered in the BWROG peer review.

Response to RAI 1a

Response is to be provided by September 30, 2006.

Response to RAI 1b

Response is to be provided by September 30, 2006.

Response to RAI 1c

It was the assessment of the peer review team that the VYNPS PSA can be effectively used to support applications involving risk significant determinations supported by deterministic analysis, once the significant Facts and Observations (F&Os) are addressed. All of the significant ('A' and 'B' priority) BWROG peer review comments have been resolved and appropriate modeling changes have been implemented. Therefore, the VY04R1 model provides the necessary and sufficient scope and level of detail to allow the calculation of CDF and large early release frequency (LERF) changes in support of the SAMA analysis.

Response to RAI 1d

The internal flood analysis was performed to satisfy an NRC request for information regarding the IPE. VYNPS chose to evaluate internal flooding events within the scope of the Individual Plant Examination for External Events (IPEEE) rather than within the IPE.

The Vermont Yankee IPEEE internal flood assessment methodology included the following approach:

- Review of plant information and documentation applicable to internal flooding.
- Plant walkdown surveys.
- Screening of buildings and areas that have no safety-related, IPE equipment, or significant flooding sources.
- Deterministic evaluation to assess whether flooding sources in an area can affect equipment in the area.
- Quantitative evaluation of the remaining areas using: (i) the flood initiating event frequencies based on specific pipe segments; and (ii) event trees constructed to model the flood conditions.

A review of the internal flooding analysis was conducted as part of the BWROG PRA PEER Review Certification conducted in 2002. This review included under PRA Strengths the following: 'Internal flooding and HVAC dependencies were systematically evaluated and documented'.

No 'Recommended Areas for Improvement' were identified for internal flooding events.

Subsequent to the Peer Review, the following significant changes have been made to the internal flooding model:

1. <u>The 2002 Update of the PRA model included the following modifications.</u>

   - SW Discharge Pipe Break in Torus Room

   A large service water system (SW) break in the Reactor Building torus room (RBTRF2 initiator) was originally characterized as an "un-isolable". Because this initiator was modeled as "un-isolable", and would occur in the discharge line (which is common to both SW supply headers), the only assumed means of mitigating the flood was to stop the SW pumps, thus rendering the SW system unavailable and non-recoverable. Consequently, termination of all SW flow was assumed to fail all equipment which depended on SW. Applying this assumption left only the condensate transfer system available for inventory makeup. Decay heat removal could only be accomplished by the hard-piped torus vent.

   The VY procedure addressing "Loss of Service Water," was significantly revised to address large breaks in the service water system. The revised procedure contains mitigation strategies, and as a result, the RBTRF2 initiator is no longer characterized as an "un-isolable" break.

   In this model update, RBTRF2 was modeled as two initiating events: RTRFR2 - SW discharge pipe break on the reactor building side of SW-18, and RTRFT2 - SW discharge pipe break on the turbine building side of SW-18. For both initiators, operator actions are credited to realign the SW discharge (if necessary) for recovery of the SW function.

- HEP (Human Error Probability) for Flood Mitigation

The human error probability for the "initial operator action" (top event IOA) for flood mitigation was revised for many of the flooding events. The revised HEP is 1E-04, which is based on the joint-HEP, time dependent, lower bound curve from the THERP methodology for time window >30 minutes. Use of the lower bound value is appropriate for many of the flood scenarios where mitigation procedures, training, time window, and action(s) to be taken were collectively judged to significantly increase the likelihood of operator success.

2. The 2004 Update of the PRA model included the following modification.

   - Large Service Water Break on 280' Elevation of Reactor Building

   The original model assumed that a major break in the service water supply piping on Elevation 280' (north) had the potential to fail ECCS instrument panel 6B (S2), channels A and C. Thus, ECCS signals would be degraded, with the result that this function could only rely on channels B and D, ECCS Cabinet 5B (S1), located to the south. Internal inspection of the 5B/6B cabinets revealed that the lower lip of the rear panel door is approximately 9-inches above floor level. Cabinet 5B (S1) is located away from the postulated pipe break site and the flood level at its location should not be as great as that at the cabinet 6B (S2) location. Vermont Yankee also note that ECCS Panel 6B and Panel 5B are sealed cabinets (for EQ purposes), and are unlikely to experience intrusion due to water spray.

   The model was modified to eliminate a guaranteed failure of ECCS instrument panel 6B (S2) in the event of a large service water break on this elevation.

NRC RAI 2

Provide the following information relative to the Level 2 analysis:

a. Section E.1.2.2.5 implies that the binning of Level 1 results into plant damage states (PDSs) is the principal means of ensuring the proper Level 1 to Level 2 interface. Section 4.3 of the IPE states that binning is only used to summarize and report the results. Clarify the use of PDSs, including whether the containment event tree is directly linked to the Level 1 models (such that Level 1 failures are recognized by the Level 2 analysis).

b. Provide the fission product release characteristics for each release category, including fission product release fractions, release times and duration, warning time, release elevation, and energy of release.

c. Briefly describe the approach used to determine the source terms for each release category. Clarify whether new modular accident analysis program (MAAP) analyses were performed as part of the development of the current model and how the MAAP cases were selected to represent each release category (i.e., based on the frequency-dominant sequence in each category or on a conservative, bounding sequence).

d. Clarify whether the Level 2 model was included in the BWROG peer review. If so, describe the conclusion relative to this element. If not, describe the internal and external reviews of the Level 2 analysis that have been performed, the results of these reviews, and any subsequent model updates.

e. Approximately 75 percent of the CDF results in an "early" release. Explain this relatively high percentage and describe the containment failures/release modes that lead to these releases.

Response to RAI 2a

The Level 2 analysis uses the RISKMAN code capability of linking event trees. This method allows the Level 2 containment event tree (CET) to be coupled directly with the Level I event trees to allow a direct computation of containment dependencies for each sequence using rules and split fractions that recognize the status of systems from the Level I event trees as well as the nature of the initiating event.

The link between the Level 1 PSA accident sequences and the CET occurs in the definition of the Level I end states. The definitions of the end states are developed to transfer the maximum amount of information regarding the accident sequence characteristics to the CET assessment. A broad spectrum of accident sequences have been postulated that could lead to core damage and potentially challenge containment. The Vermont Yankee Level 1 PSA has calculates the frequency of those accident sequences that contribute to CDF for Vermont Yankee using system oriented (systematic) event trees. Each of these sequences may result in different challenges to containment. These challenges to containment have similarities in their functional failure characteristics.

While this method precludes the necessity of formally defining functionally related plant damage states, it is useful for assuring completeness of the CET derivation to address the functional basis of accident sequence types and for portraying Level I results in terms of specific plant damage states.

Response to RAI 2b

Response is to be provided by September 30, 2006.

Response to RAI 2c

Response is to be provided by September 30, 2006.

Response to RAI 2d

The BWROG peer review concluded that the Level 2 model was performed and documented very well. The model was graded as appropriate to support risk significance evaluations. Furthermore, as indicated in ER Section E.1.4, the level of detail in the Level 1/Level 2 interface, including the plant damage state and containment event tree end state definitions, was identified as a strength. The fact that the full spectrum of severe accident phenomena listed in the ASME PRA draft standard was considered in the Level 2 evaluation was also identified as a strength.

Response to RAI 2e

Early releases are dominated by sequences in which there is a total loss of core cooling in the Level 1 analysis of core damage. In the Level 2 analysis of these sequences, recovery of core cooling fails for early release sequences. The total loss of core cooling leads directly to failure of drywell shell integrity due to core melt-through of the reactor vessel, with insufficient water on the drywell floor in the pedestal region to prevent migration of the core debris to the steel containment shell.

NRC RAI 3

With regard to the treatment and inclusion of external events in the SAMA analysis:

a. The environmental report (ER) uses the staff's conclusions from a prior SAMA evaluation to justify that the VYNPS fire CDF is conservative by a factor of three. Provide a description of the conservatism in the dominant VYNPS fire CDF sequences (e.g., related to fire initiating event frequencies, severity factors, or recovery actions that were not credited) that would support this factor of three.

b. The seismic CDF at VYNPS is not mentioned in the ER or included within in the multiplier used to account for additional SAMA benefits in external events. Provide the estimated seismic CDF at VYNPS, and an assessment of the impact on the external event multiplier, and on the SAMA analysis results if the seismic CDF is included.

c. Entergy's baseline evaluation of SAMA benefits considers only the risk reduction associated with internal events, and neglects the additional risk reduction that a SAMA could have in external events. Entergy does consider the potential for additional risk reduction in external events, but this is done in the context of an upper bound assessment in which the internal event benefits are increased by a factor of ten to account for the combined effect of external events and analysis uncertainties. The impact of external events should be reflected in the baseline evaluation, rather than combining the impact of external events with the uncertainty assessment. In this regard, provide a revised baseline evaluation (using a 7 percent discount rate) that accounts for risk reduction in both internal and external events, and an alternate case using a 3 percent discount rate. (Note that the CDF for external events after Entergy's adjustment in the ER is 3.7 times higher than the internal events CDF. This would justify a multiplier of 4.7 or 5, rather than a multiplier of 4 as stated in the ER.)

d. Provide an assessment of the impact on the baseline evaluation results (i.e., the revised baseline evaluation, which accounts for external events) if risk reduction estimates are increased to account for uncertainties in the analysis.

__Response to RAI 3a__

The IPEEE fire analysis incorporates numerous conservative assumptions that are applied uniformly throughout the analysis. These conservatisms include the following.

- Radiant damage to cable trays is calculated ignoring whether or not the cable tray may have a solid steel bottom (which will limit radiant exposure to the actual cable).

- For target-specific modeling, it was assumed (conservatively) that both 383 & non-383 cable will spread 10 ft/hr (2 inch/minute) horizontally when there is no exposure fire to preheat the trays. Upward fire spread in vertical trays was assumed instantaneous unless limited by a fire wrap, coating or penetration seal.

- Thermal damage to cable inside steel conduit is calculated ignoring the heat transfer considerations (and protection) provided by the conduit.

- Targets located in the fire plume are examined for thermal damage by ignoring the horizontal distance (0.2 times the fire to ceiling distance allowed) from the plume centerline to the target. Only the vertical distance is considered.

- Target damage values (target response parameters) for various power cables are assumed to be similar to values for smaller diameter cable of similar insulation and jacket composition.

- Except for several specific cases, fires were assumed to burn at the peak heat release rate for the entire burn period, with no allowance for slow growth. This is especially conservative for cabinet and bus fires and non-liquid fires.

- Liquid combustible spills are assumed to spill the entire contents. If a floor drain is present, no credit is given for removal of the liquid.

Additional conservatisms were included in the scenarios which are significant contributors to fire induced CDF. The significant scenarios occur in the control room or cable vault, requiring control room evacuation and use of the alternate shutdown (ASD) panel; in the east or west switchgear room; and on reactor building 252' elevation. Specific conservatisms included the following.

- The fire frequencies and their severity were conservative. Since this analysis, the trend has been toward lower frequency and less severe fires.

- For scenarios requiring ASD control for cable vault and control room fires, it was assumed that a SRV failure to close resulted in guaranteed core damage, based on the assumption that this would occur while the operating crew was transitioning from the control room to the ASD. As a result, LPCI would not be initiated in time to restore the inventory lost.

- The original model only credited the Vernon Tie if emergency power was needed. Emergency diesel A can also be started during cable vault and control room fires, as part of the ASD strategy.

- Within the switchgear rooms, only automatic actuation of the CO2 systems was considered. If automatic actuation fails, indication will be available to the operators, who can, if needed, actuate the CO2 systems from outside the switchgear rooms.

- For cable vault fires that are dominate contributors, no credit was taken for automatic suppression, based on conservative fire modeling assumptions. In fact, automatic suppression would likely be possible for most of these scenarios.

- Switchgear fires with the potential to damage cables associated with offsite power automatic fast transfer control were considered to cause a loss of offsite power. If fast transfer has failed, offsite power could still be established by operators closing the required supply breakers.

- No credit was given for manual action to locally restore equipment initially lost in switchgear room and reactor building fires. In addition, no credit was given for restoration of offsite power during cable vault and control room fires.

- All equipment (with minor exceptions) that could be impacted within the area is assumed to fail in the reactor building. No credit was taken for manual suppression in reactor building fires.

Due to these many conservatisms, it is appropriate to use the Staff's conclusion from a prior SAMA evaluation to justify that the VYNPS fire CDF is conservative by a factor of three.

Response to RAI 3b

As stated in ER section E.1.3.1, a seismic margin assessment (SMA) was performed for the seismic portion of the IPEEE. Since the SMA approach is a deterministic evaluation that does not calculate risk on a probabilistic basis, a CDF was not calculated. The limiting values for the high confidence of low probability of failure (HCLPF) were 0.25g peak ground acceleration from failure of the condensate storage tank and the main fuel oil storage tank (TK-40-1A) with a HCLPF of 0.29g. These values, although below the 0.3g review level earthquake, represent significant margin to the design basis 0.14g earthquake. See response to questions 5a and 5b for further discussion of the condensate storage tank and diesel fuel oil storage tank. A number of other plant improvements were identified in NUREG-1742, which were implemented. As seismic events are not dominant contributors to external event risk and all outliers have been addressed, further cost-beneficial seismic improvements are not expected and seismic events are considered negligible in estimation of the external events multiplier.

Response to RAI 3c

Response is to be provided by September 30, 2006.

Response to RAI 3d

Response is to be provided by September 30, 2006.

NRC RAI 4

Provide the following information concerning the MACCS analyses:

a. Annual meteorology data from the year 2002 were used in the MACCS2 analyses. Provide a brief statement regarding the acceptability of use of this year's data rather than a different year's data.

b. For the emergency response assumptions, indicate what percentage of the population was assumed to evacuate.

c. The MACCS2 analysis for VYNPS is based on a core inventory from a mid-1980 analysis, scaled by the power level for VYNPS. Current boiling water reactor BWR fuel management practices use longer fuel cycles (time between refueling) and result in significantly higher fuel burn-ups. The use of the older BWR core inventory, 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. Justify the adequacy of the SAMA cost benefit evaluation, given the fuel enrichment and burn-up expected at VYNPS.

<u>Response to RAI 4a</u>

The 2002 meteorological data set was the most current and complete at the time of data collection for this study. The on-site primary meteorological system, which was the major data source, constitutes more than 99% of the 8760 hourly values required by MACCS2. The remaining data were obtained from the backup system, the 140-foot tower on the VYNPS site.

<u>Response to RAI 4b</u>

For the emergency response assumptions, the entire population (or 100% of the population) within the 10-mile emergency planning zone was assumed to evacuate.

<u>Response to RAI 4c</u>

Best-estimate inventory of long-lived radionuclides such as Sr-90, Cs-134, and Cs-137 were derived from an ORIGEN calculation assuming 4.65% enrichment and average burn-up according to the expected fuel management practice. It was found that the best-estimate inventory differed from the power-scaled reference inventory by less than 25%.

The revised baseline benefits to be reported in response to RAI 3c will include the impact of the 25% increase in the inventory values for Sr-90, Cs-134, and Cs-137 for each analysis case.

NRC RAI 5

Provide the following with regard to the SAMA identification and screening processes:

a. Section E.1.3.1 indicates that no simple cost-effective enhancements have been identified that will significantly improve the high confidence in low probability of failure (HCLPF) for the condensate storage tank (CST) of 0.25. Provide a cost benefit analysis for the seismic improvement of the CST similar to that for the other SAMAs.

b. The individual plant examination of external events (IPEEE) found that the diesel fuel oil storage tank had a HCLPF of 0.29. The ER states that all improvements identified in NUREG-1742 (which include the diesel fuel oil storage tank) have been implemented. Describe the actions taken for the diesel fuel oil storage tank.

c. The VYNPS IPEEE lists a number of seismic improvement opportunities that are not specifically included in NUREG-1742 (specifically, seismic items 3 (ii) and 7 of IPEEE Section 7.2.2). Confirm that these have been implemented.

d. Describe any further efforts made to determine if any SAMA candidates exist to address seismic risk beyond those already identified in the IPEEE.

e. The listing of "risk significant terms," provided in Table E.1-3, includes numerous different internal flooding initiators, and the SAMAs considered to address these initiators. For most of these initiators, various Phase I SAMAs are identified as having been implemented, and Phase II SAMA 47 was evaluated to further reduce the internal flooding contribution.

   I. For each of the previously implemented changes, clarify whether the change is credited in the current PSA. If not, provide an assessment of the impact of the change on the internal flood CDF. If the change has already been credited, it would not appear to have been completely effective (as evidenced by the high residual risk of the initiating event) and additional SAMAs specific to the flooding event listed in the table could be cost-beneficial.

   II. Phase II SAMA 47 does not appear to address any of the specific internal flooding events listed in the table. Clarify which specific flooding scenario is addressed by SAMA 47.

f. Provide the current status of the 14 opportunities for improvement identified in the IPEEE for internal flooding, indicating if they have been implemented and if credit is taken for them in the current PSA. For those not implemented, indicate their importance and why they should not be considered as SAMA candidates.

g. The fire CDF, even after the factor of three reductions, is almost four times the internal events CDF. While the ER states that the improvements that address fire risk at VYNPS recommended in NUREG-1742 have all been implemented, the fire CDF is still substantial. SAMA candidates based on internal risk contributors will not necessarily address the fire risk. For each fire area or dominant fire sequence, explain what measures were taken to further reduce risk, and explain why the fire CDFs can not be further reduced in a cost-effective manner.

h. In Table E.1-3, the entry for "Transient with [power conversion system] available - initiating event" (risk reduction worth (RRW) of 1.0287) cites SAMA 046 to improve main steam isolation valve (MSIV) design. Explain how this impacts the initiator which must have the MSIV open.

i. As an alternative to Phase II SAMA 2, consider operating procedure revisions to provide additional space cooling via the use of portable equipment or blocking doors open.

j. Phase II SAMA 59 considers installing instruments for opening safety/relief valves (SRVs) for medium loss of coolant accidents (LOCAs). Explain why the benefits of this SAMA in small LOCAs and transients are not included in the benefit assessment.

k. Table E.1-3 indicates that failure of torus venting components has a RRW of 1.0948. Describe the failures considered in this assessment. Provide an assessment of the costs and benefits associated with: 1) adding redundant components, and 2) converting the vent system to a passive design.

l. The Table E.1-3 entry for "Operator Action: Operator fails to start a [turbine building closed cooling water] (TBCCW) pump" indicates that no Phase II SAMAs were recommended. Provide an assessment of the costs and benefits of starting a TBCCW pump automatically.

<u>Response to RAI 5a</u>

Response is to be provided by September 30, 2006.

<u>Response to RAI 5b</u>

There are two separate fuel oil storage tanks reference in NUREG-1742: the first is the main fuel oil storage tank, TK-40-1A (75,000 gallons) referred in the Section 3.2.4 of the IPEEE and the second is the diesel fire fuel pump fuel oil storage tank (day tank) TK-43-1A (350 gallons) referred in Section 7.2.2 of the IPEEE submittal.

No enhancements or modifications were considered for TK-40-1A (main fuel oil storage tank). A potential vulnerability with TK-43-1A (diesel fire pump fuel oil storage tank) support to resist seismic loads was identified in Section 7.2.2 of the IPEEE (and documented in Tables 2.7 and 2.12 of NUREG-1742). This seismic vulnerability was addressed with an enhancement re-routing the tubing to put a flexible loop into it that eliminated the "hard" point vulnerability. This enhancement has been implemented.

<u>Response to RAI 5c</u>

IPEEE Section 7.2.2, item 3(ii) refers to a masonry wall that is part of the diesel fire pump enclosure. This wall is enveloped in configuration/materials of construction by the walls screened and evaluated under the IPEEE. All masonry wall evaluations concluded that the limiting HCLPF was > 0.3g (refer to IPEEE Report Section 3.2.4).

For IPEEE Section 7.2.2, item 7, it was concluded that control room ventilation could be removed from the safe shutdown equipment list.

Response to RAI 5d

Additional SAMA candidates were evaluated to address seismic risk beyond those already identified in the IPEEE during Phase I SAMA candidates screening. These SAMAs are presented Table RAI.5-1.

Response to RAI 5e

I. Each of the changes related to the internal flooding initiators in Table E.1-3 has been implemented and is credited in the current PSA.

II. Phase II SAMA 47, "Shield injection system electrical equipment from potential water spray," specifically addresses, "Internal Flooding Initiator, SW pipe break at El. 303' of the reactor building" (ER page E.1-9). One specific break in the SW system 18-inch diameter supply piping on reactor building elevation 303' has the potential to impact one ECCS 24V DC distribution panel due to spray.

Response to RAI 5f

The following is the current status of the 14 opportunities for improvement for internal flooding identified in Section 7.2.3 of the IPEEE:

1. RB252 Equipment Locker: The proposed improvement is to raise the equipment storage locker at the east end of the CRD stairway to minimize flow blockage to the CRD stairwell.

   Completed, credited in the current PSA.

2. RB252 Floor Sleeves: The proposed improvement is to lower the sleeve height at El. 252' (30" and 24" diameter sleeves) to improve water flow to torus room.

   Completed, credited in the current PSA.

3. ECCS Corner Room Equipment Hatches: The proposed improvement is to seal/modify hatch lift points to ensure that the hatches are water tight.

   Completed, credited in the current PSA.

4. ECCS Corner Room Flood Berms: The proposed improvement is to increase the berm height to prevent flooding of the ECCS corner room stairwell and pipe/electrical chase which penetrates the ceilings of the ECCS corner rooms (El. 252').

   Completed, credited in the current PSA.

5. El 303 Floor Chase Berms: The proposed improvement is to either increase the berm height at the existing floor chases along the north wall (or seal floor chase opening or the panel) or otherwise ensure that panel CP82-2 (located below on El. 280') is not adversely affected.

   Completed, credited in the current PSA. RHR alternate shutdown panel CP82-2 is located on elevation 280' below the floor chase along the north wall on elevation 303'. Engineering evaluation determined that this physical arrangement will not result in spray from a flood on elevation 303' due to overflow of the 4" berms. Panel CP 82-2 is supported on a 4 inch pad and the lowest required component is approximately 2 feet above the floor. No physical changes were necessary.

6. Upper RCIC Water Relief: The proposed improvement is to provide a relief path at El. 232' so water accumulation in the upper RCIC area (due to random fire pipe failure) will relieve to the lower RCIC area before floor failure occurs.

   Completed, credited in the current PSA. Analysis concluded that existing flood relief will occur prior to floor collapse. No physical changes were necessary.

7. RB Unisolable SW Break: The proposed improvement is to evaluate procedural enhancements, hardware changes and possible restoration/recovery actions for mitigating an "unisolable" SW break in the reactor building including any adverse affects on the torus.

   Completed, credited in the current PSA. Procedure ON 3148, "Loss of Service Water", was significantly revised to address large breaks in the service system, including large discharge line breaks in the reactor building. The upgraded procedure distinguishes whether the SW discharge pipe break is located on the reactor building side of SW manual valve SW-18 or the turbine building side of SW-V18. The revised procedure contains a mitigation strategy for each break location, which uses either the SW discharge block or the deep basin (west cooling tower) for the recovered SW discharge path.

8. FOB/Switchgear Room Doors: The proposed improvement is to seal switchgear room doors to reduce the potential for internal flooding interaction with the front office building (FOB). The doors include: (i) single door - west switchgear room entrance from control building, and (ii) double doors - west switchgear room entrance from turbine building.

   Completed, credited in the current PSA.

9. FOB to Switchgear Room Vestibule Door: The proposed improvement is to ensure that the outer door to the west switchgear room vestibule does not latch and will open toward the FOB.

   Completed, credited in the current PSA.

10. FOB to Turbine Building Door: The proposed improvement is to ensure that the FOB double door to the turbine building will open toward the turbine building to relieve water from the FOB to the turbine building.

    Completed, credited in the current PSA.

11. FOB Flooding Procedures: The proposed improvement is to evaluate procedural enhancements for mitigating internal flooding in the FOB turbine building heating ventilation and air conditioning room.

    The intent of this improvement was to provide additional mitigative guidance until previously discussed modifications were completed. With FOB modifications 8, 9 and 10 installed as described above, proposed improvement is not needed.

12. Diesel Generator Room Independence: The proposed improvement is to evaluate procedural enhancements, hardware changes and possible

restoration/recovery actions for mitigating the effects of a SW line break in a diesel generator room.

The total CDF for both diesel room SW flooding events is low (approximately IE-07/yr) with no credit given for operator action to mitigate the event (i.e., operator opens the diesel room doors). Based on the low CDF, hardware or procedural changes are not warranted.

13. Torus Integrity: The proposed improvement is to evaluate the potential for containment failure during a major flood in the reactor building basement (torus room).

    The potential for containment failure during a major flood in the reactor building basement was evaluated and determined to be non-credible. Therefore, hardware or procedure changes are not warranted.

14. Alternate Cooling Alignment: The proposed improvement is to evaluate procedural and hardware enhancements for aligning alternate cooling mode during a major flood in the reactor building basement (torus room).

    Sensitivity studies show that significant water level on the torus room floor during postulated SW break scenarios leading to the inability to-align alternate cooling, is not a significant contributor to plant risk. Therefore, alternate cooling procedural and hardware changes are not warranted.

Response to RAI 5g

As described in the response to RAI 3a, one reason the fire risk is so high is the significant conservatisms inherent in the analysis. The significant scenarios occur in the control room or cable vault, requiring control room evacuation and use of the ASD panel; in the east or west switchgear room; and on reactor building 252' elevation. Turbine building fire scenarios follow these scenarios in significance.

Dominant switchgear room fires, reactor building fires and turbine building fires scenarios can be grouped into three distinct core damage classes.

- Loss of all high pressure injection and failure to depressurize. Core damage occurs with the reactor at high pressure (Class IA).
- Loss of all injection with core damage occurring at low reactor pressure (Class ID).
- Loss of all containment heat removal. Core damage is caused by containment failure (Class IIA).

These fire scenario core damage classes are also significant contributors to the internal events core damage frequency. Therefore, SAMA candidates to respond to internal risk contributors are also applicable to these fire scenarios. Several Phase I and Phase II SAMAs related to improvements to high pressure injection capabilities, reactor vessel depressurization capabilities, low pressure injection capabilities and loss of containment decay heat removal capabilities that were evaluated would reduce the CDF contribution from fires in these areas.

Fire-related Phase I SAMAs were also considered. These Phase I SAMAs are presented in Table RAI.5-2.

Therefore, for switchgear room, reactor building, and turbine building fires, no additional cost-effective hardware or procedural changes were identified to reduce CDF in these areas.

Control room fires and cable vault fires resulting in evacuation of the control room and subsequent control from the ASD panels are also mitigated by SAMAs responding to internal risk contributors and the fire-related SAMAs in Table RAI.5-2. Both areas are equipped with a detection system that alarms in the control room and the cable vault has an automatic suppression system. Therefore, no cost-effective hardware changes were identified to reduce CDF in these areas. Following the VYNPS Fire Hazards Analysis provisions and procedures provides assurance that risk in these areas is minimized. Therefore, no cost-effective procedural changes were identified to reduce CDF in these areas.

Response to RAI 5h

The goal of SAMA 046 is to improve MSIV valve and actuator design for long term reliability. The scope of the proposed design includes both improved MSIV seating capability and actuator operation. The improved seating capability would decrease the likelihood of containment bypass scenarios. The improved actuator design would decrease the probability of inadvertent MSIV closure (and subsequent reactor scram) when periodic testing is conducted with the reactor at power.

Response to RAI 5i

Response is to be provided by September 30, 2006.

Response to RAI 5j

Phase II SAMA 59 provides a means to reduce the consequences of a medium LOCA by increasing SRVs reliability to open automatically. Since this SAMA is considered only for Medium LOCAs, the benefits for this SAMA are applied only to the occurrence of Medium LOCAs.

The potential impact on SRV reliability for small LOCAs and transients is evaluated in SAMA 60. This SAMA would improve SRV design to increase the likelihood that accident sequences could be mitigated using low pressure heat removal. This SAMA was evaluated by eliminating the probability of SRV failure to open for vessel depressurization for applicable accident sequences.

Response to RAI 5k

Response is to be provided by September 30, 2006.

Response to RAI 5l

Response is to be provided by September 30, 2006.

Table RAI.5-1 Improvements Related to Reduce Seismic Risk

Phase I SAMA ID number	SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
205	Increase seismic ruggedness of plant components	SAMA would increase the availability of necessary plant equipment during and after seismic events.	#3 - Already installed	VYNPS components whose seismic ruggedness could be improved were identified in the IPEEE and SQUG programs. These items have been addressed in response to those efforts and satisfy the intent of this SAMA.
206	Provide additional restraints for CO_2 tanks	SAMA would increase availability of fire protection given a seismic event.	#3 - Already installed	VYNPS does rely on CO_2 fire suppression systems to minimize fire risk in the switchgear room, cable vault, and diesel driven fire pump room. The CO_2 bottles located in these room, the cable vault, and switchgear room have been designed to prevent them overturning in the event that a design basis SSE (safe shutdown earthquake) were to occur. All piping and components that comprise the initial and extended discharge system header located in the east and west switchgear rooms are seismically mounted and supported. This precludes the possibility of the header failing during a seismic event and affecting the safety class switchgear located below the header. The CO_2 system, while designated as non-safety-related, performs a function important to personnel safety during postulated fire scenarios. For this reason it was designed for seismic loading considerations, thereby accounting for potential IPEEE related concerns. The low pressure tank, tank access platform, and piping outside of the switchgear rooms has been seismically installed and supported to satisfy seismic design parameters pertaining to VYNPS Class II structures (as a minimum).
207	Increase seismic capacity of the plant to a high confidence of a low probability of failure (HCLPF) of twice the safe shutdown earthquake.	Reduce the plant risk contribution from seismic event	#3 - Already installed	The IPEEE seismic margin analysis determined that the plant HCLPF based on seismic faults only (random failures and human failures excluded) was 0.30g, which is more than twice the design basis safe shutdown earthquake (0.14g) except that CST HCLPF value is 0.25g.
208	Ensure that MCCs are adequately secured per seismic or other requirements	Increased reliability of MCCs during and after a seismic event	#3 - Already installed	Through completion of the A-46 program, VYNPS verified the seismic adequacy of anchorage for all MCCs on the safe shutdown equipment list.

Table RAI.5-1 Improvements Related to Reduce Seismic Risk

Phase I SAMA ID number	SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
209	Ensure that control cabinets are adequately secured per seismic or other requirements	Reduce the plant risk contribution from seismic induced internal fire event	#3 - Already installed	Through completion of the A-46 program, VYNPS verified the seismic adequacy of anchorage for all control cabinets on the safe shutdown equipment list.
210	Ensure that compressed gas, gas, propane, or tanks containing other flammable/ combustible fluids are adequately secured per seismic or other requirements	Reduce the plant risk contribution from seismic induced internal fire event	#3 - Already Installed	VYNPS procedures require that compressed gas cylinders are stored in secure manner that will prevent overturning during a seismic event.
212	12.a. Increased Seismic Margins	This SAMA would reduce the risk of core damage and release during seismic events	#2 - Similar item is addressed under other proposed SAMA 207	See disposition on SAMA 207. VYNPS completed A46 project and IPEEE report.

Table RAI.5-2 Fire Related Phase I SAMAs

Phase I SAMA ID number	SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
214	Enhance control of transient combustibles	SAMA would minimize risk associated with important fire areas.	#3 - Already installed	Procedures to control the transportation of combustible material are in place at VY. Based on IPEEE insights.
215	Enhance fire brigade awareness	SAMA would minimize risk associated with important fire areas.	#3 - Already installed	The fire brigade is trained and maintained per the referenced documents.
216	Upgrade fire compartment barriers	SAMA would minimize risk associated with important fire areas.	#3 - Already installed	VY fire compartment barriers are maintained to reduce fire propagation. Based on IPEEE insights.
217	Enhance procedures to allow specific operator actions	SAMA would minimize risk associated with important fire areas.	#3 - Already installed	VY safe shutdown procedures are available for use to accomplish safe shutdown in response to fire. The purpose of this procedure is to outline those actions necessary to safely shutdown the plant in the event that the Control Room must be evacuated, or there is a fire in the cable vault or other plant area affecting the operation of equipment needed for a safe shutdown.
218	1.f. Remote Shutdown Station	This SAMA would allow alternate system control in the event that the control room becomes uninhabitable.	#3 - Already installed	VY Procedure OP-3126, Rev.16, Shutdown Using Alternate Shutdown Methods outlines the remote shutdown activities necessary to safely shutdown the plant in the event that the control room becomes uninhabitable.
219	Isolate combustible sources for seismic or other events	Limit combustible source to that enclosed in line	#3 - Already installed	Hydrogen storage is located in secure configurations, with piping systems designed to preclude release of combustible gases in plant areas. Vent lines are provided with flame suppressors to preclude ignition. Battery systems are located in areas that are well ventilated to preclude accumulation reaching flammable limit.
220	Restrain or locate flammables cabinets to reduce the likelihood of overturning caused by seismic or other events.	Eliminate probability of cabinets overturning, spilling flammable liquid contents.	#3 - Already installed	VY flammables cabinets contain small quantities of flammables, usually in the original containers that seal tightly, so overturning a cabinet would not result in releasing a significant amount of flammable material.

Table RAI.5-2 Fire Related Phase I SAMAs

Phase I SAMA ID number	SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
221	Ensure that the quantity of combustible materials in critical process areas is monitored	Minimize combustibles and chance of prolonged fire in safety-related areas	#3 - Already installed	VY has a procedure governing the fire-safe use and storage of combustible materials within the process buildings.
222	Monitor and control pre-staging of outage materials	Reduced fire risk	#3 - Already installed	VY Procedure AP-0042 establishes the requirements for the control of site specific combustible material storage, ignition sources and impairments of fire systems to prevent or minimize the effects of a fire at Vermont Yankee. This procedure also provides a control mechanism for tracking system impairments and instituting compensatory measures to minimize the effects that those impairments may have on safety.
223	Limit switches and torque switches would not be bypassed during a fire induced hot short for Control Room and Cable Vault fire events	This SAMA would address the reconfiguration of the MOVs control circuits and protect the motor operator via the limit and torque switches due to fire induced hot short.	#3 - Already installed	VY has reconfigured the control circuits of the Appendix R motor operated valve. With this modification, a hot short cause an MOV to inadvertently transfer position, however, the motor operator will remain protected via the limit and torque switches. Thus, the MOV itself is not damaged and remains available for later manipulation at the alternate shutdown panel.
224	Install and use additional transfer/isolation switches	SAMA would reduce the number of spurious actuation during a fire.	#3 - Already installed	This fire related risk mitigation measure has been considered as part of the VY Appendix R program and IPEEE Internal Fire Analysis.
282	North wall lower NE corner room	This SAMA would reduce the internal fire events contribution to plant risk	#3 - Already installed	The top 6" of the north wall in the lower NE ECCS Corner Room (just under floor El. 232'-6") was included in the plant fire barrier inspection program.
283	Vertical cable tray fire stops	This SAMA would reduce the internal fire events contribution to plant risk	#3 - Already installed	The inspection and maintenance program of vertical cable tray fire stops at each floor in the Reactor Building; to limit fire spread from one elevation to another was enhanced.
284	Periodic Fire Prevention Inspections	This SAMA would reduce the internal fire events contribution to plant risk	#3 - Already installed	The periodic fire prevention inspections of the Reactor Building and Control Building have been changed to monthly basis.

NRC RAI 6

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

a. For a number of the Phase II SAMAs listed in Table E.2-1, the information provided does not sufficiently describe the associated modifications and what is included in the cost estimate. Provide a more detailed description of the modifications for Phase II SAMAs 6, 9, 10, 13, 23, 24, 33, 41, 52, 56, and 63.

b. Several of the cost estimates provided were drawn from previous SAMA analyses for a dual-unit site (e.g., Peach Bottom). As such, many of those cost estimates reflect the cost for implementation in two units. Since VYNPS is a single-unit site, some of the cost estimates should be one-half of what has been cited (i.e., Phase II SAMAs 29, 35, 40, 49, 50, 51, 52, 53, and 54) while others are specific to a plant's design, such as the number of valves or batteries that need to be replaced or added (i.e., Phase II SAMAs 46, 55, and 60). For these cases, provide appropriate (specific to VYNPS) cost estimates. (Note that Phase II SAMAs 49, 50, 51, 53, and 54 are close to being potentially cost-beneficial when a 3 percent real discount rate is used.)

c. Phase II SAMA 27 uses the same analysis case (Strengthen Containment) as Phase II SAMAs 13, 18, and 19 to evaluate the benefit. Yet, Table E.2-1 lists SAMA 27 as having a CDF reduction of 0.0 percent, while all other SAMAs for this analysis case list a CDF reduction of 7.36 percent. Explain this discrepancy.

d. For Phase II SAMA 28 and 29 (and others) a 3 percent reduction in CDF was estimated by changing the time available to recover off-site power before high pressure coolant injection/reactor core isolation coolant (RCIC) are lost from 4 hours to 24 hours. According to Table E.1-8, late SBO sequences (Class IBL) contribute about 17 percent of the total CDF. Explain why only a 3 percent reduction in CDF was estimated for this SAMA.

e. For Phase II SAMA 42, a 1.3 percent reduction in offsite dose was estimated by reassigning the interfacing systems loss of coolant accident (ISLOCA) sequences to the same end states as medium LOCAs. For Phase II SAMA 43, a 1.2 percent reduction in offsite dose was obtained by eliminating the CDF contribution due to ISLOCA. One would expect the dose reduction for SAMA 43 to be greater than that for SAMA 42. Also, the CDF contribution from ISLOCA is given in Table E.1-2 as 0.32 percent, while the CDF reduction from SAMA 43 is given as 0.83 percent. Explain these apparent discrepancies.

f. Phase II SAMA 57 is stated to include items which reduce the contribution of anticipated transient without scram. Indicate which items are included.

g. Phase II SAMA 59 involves providing instrument signals to open SRVs for medium LOCA. Discuss whether the signals already exist in the automatic depressurization system.

h. Phase II SAMA 63, Control Containment Venting within a Narrow Band of Pressure, is intended to eliminate failures associated with successful venting. The benefit of this SAMA was determined by reducing the operator failure to vent by a factor of three. It is not clear that reducing the failure to vent probability is related to the actual benefit from this SAMA. Also, the cost of $250,000 appears high for what appears to be a procedure and training issue. Justify the benefit and cost for this SAMA.

i. Phase II SAMA 64, Provide Cross Tie from the residual heat removal service water (RHRSW) System to residual heat removal Loop B, has an estimated CDF reduction of 0.2 percent. The description given in Table E.1-3 for term diesel fire pump and John Deere Diesel for Alternate Injection, though, indicates that this term involves a cross tie for fire protection to RHRSW and has a RRW of 1.0584. Describe this SAMA more completely and indicate why the reduction in CDF is so small relative to the RRW.

j. In Table E.2-1, the percent change in CDF and population dose is reported for each analysis case. However, the change in the offsite economic cost risk (OECR) is not reported. Provide the change in the OECR for each analysis case.

<u>Response to RAI 6a</u>

SAMAs 6 (Install a containment vent large enough to remove ATWS decay heat) and 56 (Install an ATWS sized vent) provide a means to remove decay heat during an ATWS event. The proposed design modification for these SAMAs involves installation of a larger vent pipe than the existing 8-inch containment vent pipe. The proposed design would require a vent pipe of sufficient size to remove decay heat following an ATWS with MSIV closure and successful recirculation pump and feedwater pump.

SAMAs 9 (Provide modification for flooding the drywell head) and 23 (Provide a method of drywell head flooding) would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail. The proposed design modification requires extensive structure modification to accommodate a drywell head flooding system. In order to flood the drywell head seal at elevation 321-foot, a new penetration would have to be installed in the drywell head at the 321-foot elevation. The new vent penetration would have to be tied into the existing vent line and would have to permit removal of the drywell head at each refueling outage.

SAMAs 10 (Enhance fire protection system and standby gas treatment system hardware and procedure) and 24 (Use alternate method of reactor building spray) would improve fission product scrubbing in severe accidents. The proposed design modification would upgrade the current standby gas treatment and fire protection systems to sufficient capacity to handle postulated loads from severe accidents due to a bypass or breach of the containment. Loads produced as a result of reactor pressure vessel or containment blow-down would require large filtering capacities.

SAMA 13 (Strengthen primary and secondary containment) would reduce the probability of containment over-pressurization failure. This SAMA is intended for a new plant; hence, it is not practical to back fit this modification into a plant which is already built and operating. Since VYNPS has a MARK I containment, early release risk is dominated by events that result in early failure of the drywell shell due to direct contact with debris and events that bypass the containment. Strengthening of primary and secondary containment would have a small impact on the overall risk of these accidents. The cost estimated for ABWR was $12 million and a retrofit for an existing containment would cost more.

SAMAs 33 (Provide 16-hour SBO injection) and 41 (Extended SBO provisions) would improve the capability to cope with longer station blackout scenarios. The proposed design modification for this SAMA involves adding a battery to improve the coping capability during SBO scenarios.

SAMA 52 (Improved high pressure systems) would improve prevention of core melt sequences by improving reliability of high pressure capability to remove decay heat. The proposed design modification considers replacing one CRD pump with a flow capacity equal to the RCIC system (400 gpm).

SAMA 63 (Control containment venting within a narrow band of pressure) would establish a narrow pressure control band to prevent rapid containment depressurization when venting is implemented thus avoiding adverse impact on the low pressure ECCS injection systems taking suction from the torus. Hence, the modification for SAMA 63 requires a detailed engineering analysis examining the impact of opening the torus vent path and an examination of the NPSH requirements for LPCI and core spray systems. It would also require an engineering study of the feasibility of re-closing containment vent motor-operated valve V16-19-86 against high containment pressure and other hardware modifications. Procedure changes, simulator changes, and training would also be required.

Response to RAI 6b

Response is to be provided by September 30, 2006.

Response to RAI 6c

The discrepancy for Phase II SAMA 27 is due to an incorrect entry for CDF reduction. The Phase II SAMA 27 CDF reduction is the same as that for Phase II SAMAs 13, 18 and 19; 7.36 percent instead of 0.0 percent.

Response to RAI 6d

Late SBO sequences (IBL) result from total loss of emergency AC power with initial success of either RCIC or HPCI, but with eventual failure when batteries deplete. Loss of the offsite power grid, with potential grid recovery, is one of several initiators that contribute to IBL sequences.

The dominant contributors to the IBL end-state are not the result of a loss of the offsite power grid. The largest contributors are transients with subsequent loss of both 4 kV emergency busses. Other contributions to IBL are transients with resulting failure combinations of AC busses and DC power. For these scenarios, the offsite power grid is not impacted so modifying the time to recover offsite power does not reduce their contribution to CDF.

Response to RAI 6e

A small error in the RISKMAN model rules was identified in the analysis case for SAMA 42. This resulted in a small loss of tabulated bin totals. When corrected, SAMA 42 and SAMA 43 have essentially identical results. SAMA 42 was created by eliminating ISLOCA events and the ISLOCA initiating event frequency to the MLOCA initiating event frequency. The resulting value was essentially the same as the value obtained by totally eliminating the ISLOCA events. The reason for this is as follows:

- MLOCA has an initiating event frequency of 3.5E-5/yr and a CDF of 2.79E-9/yr, resulting in a conditional core damage probability (CCDP) of 7.97E-5.

- The total ISLOCA initiating event frequency is 2.3E-7/yr. Applying the MLOCA CCDP results in a CDF of 1.83E-11/yr, this is a numerically insignificant addition to total CDF.

SAMA 43 CDF percent reduction is larger than the ISLOCA CDF contribution listed in Table E.1-2 of the ER because the CDF contributions to ISLOCA and loss of coolant accident outside containment (LOCAOC) reported on Table E.1-2 were reversed. The correct contribution to CDF from ISLOCA is 0.73% (conversely, LOCAOC contribution is 0.32%). The SAMA 43 value of 0.83% is slightly larger due to the rounding off of less significant digits.

Response to RAI 6f

To conservatively assess the benefit of SAMA 057 (Improve ATWS coping capability), the CDF contribution from all ATWS initiating events were eliminated from quantification.

Response to RAI 6g

Phase II SAMA 59 provides a means to reduce the consequences of a medium LOCA by increasing SRV reliability to open automatically. This SAMA provides adequate RCS pressure control to prevent an over pressurization condition in the RCS and therefore preclude the occurrence of a LOCA.

The proposed design modification was based on the design implemented at the James A. Fitzpatrick Nuclear Power Plant called, "SRV Electric Lift System". This plant modification involved opening the SRVs electrically by energizing existing solenoid valves on the pilot stage assembly located on each SRV when the appropriate RCS pressure setpoint is exceeded (the pressures ranges are 1080 psig to 1100 psig). The electric lift initiation is designed to assist the existing mechanical relief in performing its intended function. The SRV electric lift system functions only as an electrical back up to the mechanical setpoint and does not prevent the mechanical portion of the SRV from operating as designed.

Therefore, the proposed design modification does not impact any existing signals in the automatic depressurization system.

Response to RAI 6h

SAMA 63 (Control containment venting within a narrow band of pressure), would establish a narrow pressure control band to prevent rapid containment depressurization when venting is implemented thus avoiding adverse impact on the low pressure ECCS injection systems (core spray and LPCI) taking suction from the torus. Since the model assumes failure of the low-pressure injection systems following containment venting, it does not contain basic events for failure of these systems following successful venting. The operator action to control containment venting within a narrow pressure band would be subjected to the same human error conditions and would reduce the CDF contribution from the same sequences as the failure to vent action. Thus, the benefit for SAMA 63 was conservatively estimated by reducing the failure to vent basic event.

As stated in response to RAI 6a, the proposed modification for SAMA 63 requires a detailed engineering analysis to examine the impact of opening the torus vent path and an examination of the NPSH requirements for LPCI and core spray systems for this condition. It also requires an engineering study of the feasibility of re-closing the direct torus vent shut off valve V16-19-86 against high containment pressures as well as potential hardware modifications. Procedure changes, simulator changes, and training would also be required. Therefore, the cost estimate of $250,000 is appropriate.

Response to RAI 6i

VYNPS has the ability to inject fire water into the vessel via an interconnection to RHR Loop A. Phase II SAMA 64 examined adding an interconnection between RHRSW and RHR Loop B to provide an alternative injection path. The impact of this modification was conservatively assessed by assuming guaranteed success to open for the isolation valves between RHRSW and RHR Loop A. Failure of alternate injection via the fire water system is dominated by failure of operator action and failure of the John Deere diesel and the diesel-driven fire pump to start or run. Since providing an alternate injection path does not remove the CDF contribution from these dominant failures, the CDF reduction is small relative to the RRW for this term.

Response to RAI 6j

Response is to be provided by September 30, 2006.

NRC RAI 7

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower cost alternatives to those Phase II SAMAs considered in the ER would be viable and potentially cost beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at VYNPS:

a. Use portable generator to extend the coping time in loss of ac power events (to power battery chargers).

b. Enhance direct current (dc) power availability (provide cables from diesel generator or another source to directly power battery chargers).

c. Provide alternate dc feeds (using a portable generator) to panels supplied only by dc bus.

d. Modify procedures and training to allow operators to cross tie emergency ac buses under emergency conditions which require operation of critical equipment.

e. Develop guidance/procedures for local, manual control of RCIC following loss of dc power.

Response to RAI 7a

Response is to be provided by September 30, 2006.

Response to RAI 7b

This SAMA has already been considered and implemented.

In 1989, VYNPS added a cable between AC-DP-D1A and Bus 9. Panel AC-DP-D1A receives its power from the John Deere diesel generator (JDDG) transfer switch. With the transfer switch in the normal position, power is supplied from Bus 11; in the emergency position, power is supplied from the JDDG.

The cable between AC-DP-D1A and Bus 9 facilitates using the JDDG to energize loads on Bus 9 and, using the existing ties between Bus 9 and Bus 8, loads on Bus 8. Loads on MCCs 8B and 9B include the main station battery chargers.

Response to RAI 7c

Response is to be provided by September 30, 2006.

Response to RAI 7d

This SAMA has already been considered (Phase I SAMA 120) and implemented.

Phase I SAMA 120 considered providing increased reliability of the AC power system to reduce core damage and release frequencies. The operators already have procedural guidance to implement the cross-tie of 480VAC buses 8 and 9, and the Vernon tie line can be aligned to either 4160VAC bus 3 or 4. In addition, operators are aware of the ability to cross-tie buses 3 and 4 utilizing Vernon tie breakers 3V and 4V.

Response to RAI 7e

This SAMA has already been considered and implemented via the VYNPS Severe Accident Management Program. Procedure PP 7109, Appendix G, Attachment 5 – "Operation of RCIC with No DC Power", contains instructions for operation of RCIC without DC power available.