Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, New York 13093

January 31, 2005
NMP1L 1921

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

SUBJECT: Nine Mile Point Units 1 and 2
Docket Nos. 50-220 and 50-410
Facility Operating License Nos. DPR-63 and NPF-69

License Renewal Application – Revised and/or Supplemental Responses to NRC Requests for Additional Information Regarding the Analysis of Severe Accident Mitigation Alternatives (TAC Nos. MC3274 and MC3275)

Gentlemen:

By letter dated May 26, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an application to renew the operating licenses for Nine Mile Point Units 1 and 2, including the Environmental Report – Operating License Renewal Stage (ER).

In letter NMP1L 1894 dated December 6, 2004, NMPNS provided responses to requests for additional information (RAI) contained in the NRC letter dated October 20, 2004, regarding the analysis of severe accident mitigation alternatives (SAMAs). This letter revises the NMPNS response to RAI 3a regarding the offsite consequence portion of the SAMA evaluation, as provided in Attachment 1. In addition, Attachment 2 provides supplemental responses for RAIs 1a, 1c, 1d, 2a, 2f, 3b, and 4 to address follow-up NRC information requests transmitted to NMPNS by email on December 29, 2004. This letter contains no new regulatory commitments.

If you have any questions about this submittal, please contact Peter Mazzaferro, NMPNS License Renewal Project Manager, at (315) 349-1019.

Very truly yours,

James A. Spina
Vice President Nine Mile Point

JAS/DEV/jm

A107

STATE OF NEW YORK :
: TO WIT:
COUNTY OF OSWEGO :

I, James A. Spina, being duly sworn, state that I am Vice President Nine Mile Point, and that I am duly authorized to execute and file this supplemental information on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this submittal are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 31st day of January, 2005.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires: 1/31/05
Date

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10/25/05

Attachments:
1. Revised Response to NRC Request for Additional Information (RAI) 3a Regarding the Offsite Consequence Portion of the Severe Accident Mitigation Alternative (SAMA) Evaluation
2. Responses to Follow-up NRC Requests for Additional Information (RAI) Regarding the Analysis of Severe Accident Mitigation Alternatives (SAMAs)
3. Level B Facts and Observations That Have Not Been Incorporated into the Nine Mile Point Unit 1 PRA Model To Date
4. Level B Facts and Observations That Have Not Been Incorporated into the Nine Mile Point Unit 2 PRA Model To Date

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR
Ms. L. C. Fields, Environmental Project Manager, NRR
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1

Nine Mile Point Nuclear Station

Revised Response to NRC Request for Additional Information (RAI) 3a

Regarding the Offsite Consequence Portion of the

Severe Accident Mitigation Alternative (SAMA) Evaluation

In letter NMP1L 1894 dated December 6, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) provided a preliminary response to request for additional information (RAI) 3a contained in the NRC letter dated October 20, 2004. That response stated that NMPNS would perform a formal re-analysis of the Nine Mile Point Unit 2 (NMP2) SAMA results based on a General Electric (GE)-produced core inventory, and that a bounding analysis approach would be used for Nine Mile Point Unit 1 (NMP1). NMPNS has completed these analyses. RAI 3a is repeated below, followed by the revised NMPNS response that is based on the re-analysis results.

RAI 3a

The MACCS2 analysis for both units uses a core inventory scaled by power level from a reference BWR core inventory at end-of-cycle calculated using ORIGEN. The ORIGEN calculations were based on a 3-year fuel cycle (12 month reload) with an average power density for the assembly groups ranging from 24 to 30 MW/MTU. Current BWR fuel management practices use longer fuel cycles (time between refueling) and result in significantly higher fuel burnups. The use of the reference BWR 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. Justify the adequacy of the SAMA screening and dispositioning given the fuel enrichment and burnup expected at NMP during the renewal period.

Revised Response 3a

NMPNS has performed a SAMA net benefit sensitivity calculation using a GE-produced plant-specific fission product inventory to account for uncertainties related to the reference boiling water reactor (BWR) core inventory used in the Environmental Report – Operating License Renewal Stage (ER). The end-of-cycle activity levels in the inventory were based on a bounding case of 1,400 effective full power days (EFPD) with an average 4.1 percent enrichment and are directly applicable to the power level and fuel loading used at NMP2. This case exceeds the actual expected average core exposure at end-of-cycle (approximately 1,277 EFPD). NMPNS considers the NMP2 core inventory to be bounding for NMP1 given the significant thermal power difference between the units.

In the sensitivity case, NMPNS calculated the offsite consequences and the corresponding SAMA net benefits using the same methodology and assumptions used in the original calculation, and compared the results to the NMP2 base case used in the ER. NMPNS notes that the activity levels for Sr-90, Cs-134 and Cs-137 are 60 to 73 percent higher than those in the reference BWR inventory, and all of the net benefit results are higher than the original SAMA analysis. However, the overall results are similar to the 3 percent discount rate sensitivity case performed in the original analysis (i.e., no additional SAMAs became cost beneficial). Given that the results of the 3 percent discount rate sensitivity case are fully considered for implementation, the current SAMA screening remains valid. Furthermore, the initial screening value of $5 million (see ER Section 4.16.3) bounds the revised maximum attainable benefit using the GE-produced plant-specific fission product inventory. Therefore, the initial screening based on excessive cost remains valid.

ATTACHMENT 2

Nine Mile Point Nuclear Station

Responses to Follow-up NRC Requests for Additional Information (RAI)

Regarding the Analysis of Severe Accident Mitigation Alternatives (SAMAs)

In letter NMP1L 1894 dated December 6, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) provided responses to requests for additional information (RAI) contained in the NRC letter dated October 20, 2004. The NRC subsequently transmitted follow-up information requests for RAIs 1a, 1c, 1d, 2a, 2f, 3b, and 4 by email dated December 29, 2004. This attachment provides the NMPNS responses to these follow-up information requests. Each NRC request is repeated, followed by the NMPNS supplemental response for Nine Mile Point Unit 1 (NMP1) and/or Nine Mile Point Unit 2 (NMP2), as applicable.

RAI 1a Follow-up Information Request

In the response to RAI 1a, it is stated that the BWROG reviews (performed in March 1998 and April 1997) were of the IPEs. Since the reviews were performed approximately 5 years after the IPEs were submitted to NRC, please confirm that the documents reviewed were in fact the IPEs, and not a subsequent revision. It is also stated that the most risk significant observations of the BWROG reviews were incorporated into the current PRA models. Were any Level A or B Facts and Observations (F&Os) from the reviews not implemented in the current PRA models? If so, please provide the original F&Os and discuss their impact on the SAMA analysis.

Supplemental Response 1a

The NMP1 and NMP2 Individual Plant Examinations (IPEs) reviewed by the Boiling Water Reactor Owners Group (BWROG) peer reviewers were the IPEs with some minor changes and enhancements.

There are no Level A Facts and Observations that have not been incorporated into the probabilistic risk assessment (PRA) models for either NMP1 or NMP2. The Level B Facts and Observations that have not been incorporated into the PRA models to date are provided in Attachments 3 and 4 for NMP1 and NMP2, respectively. As indicated in the "Plant Response or Resolution" section of the individual Fact and Observation sheets, NMPNS concludes that none of these outstanding Facts and Observations would significantly impact the plant's core damage frequency (CDF) and, therefore, the SAMA results would also not be impacted.

RAI 1c Follow-up Information Request

For NMP1, the IPEEE indicates that two components don't meet the 0.3 g review level earthquake (RLE). These are battery boards 11 and 12 (a HCLPF of 0.27g) and containment spray raw water pumps (a HCLPF of 0.29g). The failure of these two components would not be expected to be correlated and hence each would contribute to a separate core damage sequence. In the present model, these two items are presumably included in the fragility COMP2, which is based on a HCLPF of 0.3g, and which is assumed to lead directly to core damage. The unconditional failure frequency for COMP2 is 1.0E-06 per year using the EPRI hazard curve and 4.5E-06 per year using the NUREG hazard curve. Because the HCLPF for each of these components is less than 0.3g, their failure could therefore lead to core damage sequences with a frequency somewhat greater than the above values for COMP2. Please provide a further evaluation of potential SAMAs to address sequences due to seismic failure of these components, or justification as to why such SAMAs would not be cost-beneficial.

Supplemental Response 1c

NMPNS considers the 0.27g and 0.29g high confidence low probability of failure (HCLPF) values for Battery Boards 11 and 12 and the containment spray raw water pumps, respectively, to be close enough to the 0.3g screening value such that additional analyses and potential modifications would not be cost-beneficial. The following provides additional information with regard to this assumption:

- Benefit (General) - Based on previous SAMA evaluations, the benefit would be on the order of $10K for a 1E-7 change in CDF and $100K for a 1E-6 change in CDF.

- Benefit (General) - To accurately establish a change in CDF, a more detailed seismic fragility and risk evaluation of the plant would be required (eliminating raw water pumps by increasing their capacity does not remove the COMP2 contribution to risk). The capacity of the containment spray raw water pumps and battery boards would also be required (a more detailed analysis would likely indicate a higher capacity).

- Benefit (Containment Spray Raw Water Pumps) - The difference in risk is not distinguishable between the 0.29g HCLPF and the 0.3g screening value. The change in CDF is likely closer to 1E-7 than 1E-6 particularly since failure of these pumps primarily impacts the containment heat removal function (containment venting and time to recovery).

- Benefit (Battery Boards) - The difference in risk between the 0.27g HCLPF and the 0.3g screening value is likely closer to 1E-6 because DC power is important to the automatic operation of equipment (e.g., breaker operation) particularly since the seismic event is likely to have also caused loss of offsite power. There would be an opportunity for operator action to start pumps, etc., particularly with emergency condenser operation.

- Cost (General) – As described above, a more detailed fragility and risk evaluation, including uncertainty analysis is required to support a realistic cost-benefit analysis. This alone would easily exceed $100K.

- Cost (Containment Spray Raw Water Pumps) – The cost to perform a realistic evaluation of the pump is probably less than $100K. If the HCLPF is still assumed to be less than 0.3g, pump modifications would have to consider that the pump casing length exceeds the 20-foot limit allowed in the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment (the A-46 evaluation found the pumps adequate), which would require pump replacement of the four pumps. This would easily exceed $100K.

- Cost (Battery Boards) – The cost to perform a realistic evaluation of the battery board anchorage is probably less than $100K. If the HCLPF is still assumed to be less than 0.3g, battery board modifications would have to consider that cinch anchors with low strength lead inserts are used for anchorage (the A-46 evaluation found anchorage to be adequate); thus, the existing anchors would have to be drilled out and/or additional Hilti anchors added. This would likely require removal of equipment and/or partial disassembly of the battery board. Interferences with embedded conduit and rebar, as well as interferences with equipment internal to the panel, would also need to be considered. Given these factors and uncertainties, the cost to modify the battery board anchorage is judged to easily exceed $100K.

Based on the above considerations, NMPNS concludes that potential modifications to upgrade the seismic capability of the Containment Spray Raw Water Pumps and Battery Boards 11 and 12 would not be cost-beneficial.

RAI 1d Follow-up Information Request

The information provided in response to RAI 1d indicates some substantial changes in the relative contribution from significant release categories, as compared to the IPE results. For example for Unit 1, the total frequency of high magnitude release categories has decreased from 42% to 22% of the total CDF, the frequency of high magnitude releases for Class IA accidents has increased from 6% to 31% of the class total, and the frequency of high magnitude releases for Class ID accidents has increased from 6% to 10%. For Unit 2, the "no release" category frequency has decreased from 58% of the total CDF to 26%, the total frequency of high magnitude releases has decreased from 53% to 25% of the total CDF, the frequency of high releases for Class IA accidents has decreased from 51% to 15% of the class total, the frequency of high releases for Class IB accidents has increased from 1% to 15%, and the frequency of high releases for Class ID accidents has decreased from 99% to 68%. Given that, as stated, no major changes were made in the Level 2 analysis, one might expect the distributions, particularly within a class, to be largely unaffected. Please explain the reasons for the above differences.

Supplemental Response 1d

The changes in release categories are due to significant modeling changes that have occurred since the initial IPE. PRA model changes include the addition of initiating events, numerous system and event tree model changes, and data updates. These changes have been based on NRC safety evaluation reports, plant-specific BWROG peer reviews, and self-assessments. The latest PRA models for each plant also have seismic and fire PRA models integrated, whereas the IPE models were based on internal events only. While dominant sequences in each release category have been reviewed to ensure correct binning, it would take a substantial effort to reconsider all the individual model changes since the IPE and explain their contribution to Level 2 release changes. A qualitative review is provided below to highlight the dominant changes in the high-magnitude releases for each unit.

NMP1

NMP1 Class IA had the greatest change in high magnitude releases since the IPE (an increase of more than a factor of 5). The Class IA total CDF increased by more than a factor of 10. Besides modeling changes that occurred over the years, the addition of seismic initiators added since the IPE accounts for a major portion of the change, as they dominate the early high (EHGH) release. Note that numerous fire initiators added since the IPE may also contribute, but they account mostly for the increase in Class ID and the intermediate high (IHGH) and late high (LHGH) releases.

NMP2

NMP2 Class IB, which contains non-recovered station blackout (SBO) sequences, had the greatest change in high-magnitude releases since the IPE (an increase of more than a factor of 10). The Class IB total CDF increased by a factor of about 6. The following summarizes model changes identified as part of this review:

- The loss of offsite power (LOSP) initiating event frequency increased from 0.04 to 0.08 and emergency diesel generator control room cooling was added as a dependency, which explains a large portion of the Class IB increase.
- The SBO model was completely revised including the timing of sequences, but it is not clear that this would have a significant impact on release changes.
- A number of initiating events in the IPE that resulted in station blackout were binned to Class IA and ID. In the updated PRA, these sequences are binned to Class IB. Dominant scenarios that now show up in the EHGH and IHGH releases are from the control building flood initiator. Other initiators include partial loss of offsite power and loss of emergency switchgear. There is less recovery in these scenarios than LOSP, and they would tend to have earlier releases.
- Additional initiating events have been added to the model and contribute to Class IB, including loss of normal switchgear and several fire initiating events. These would tend to be less recoverable and have earlier timing.

RAI 2a Follow-up Information Request

In response to RAI 2a, tables containing basic event importance information were provided for NMP Units 1 and 2. The right column of the tables identifies the SAMAs that address the event. Several SAMAs are identified in these tables that are not discussed or defined anywhere in the ER or RAI responses. For the following SAMAs, please provide a brief description and evaluation information (similar to that provided in Section F.3) justifying the disposition of the SAMA (i.e., why it did not make it to Phase 2):

Unit 1: SAMAs 29, 30, 64, 106, 110, 148, 154, and 180
Unit 2: SAMAs 57, 58, 150, 153, and 161.

Supplemental Response 2a

Tables 2a-1 through 2a-4 in the initial RAI response (letter NMP1L 1894 dated December 6, 2004) list the importance results, and cross reference SAMAs that addressed the various PRA basic events. Table 2a-5 (shown below) provides a brief description and evaluation of SAMAs identified in the response that were screened from detailed evaluation. Information for six other SAMAs that are referenced in the comments field of Table 2a-5 (SAMAs 214, 215, 216, 217, 218, and 221) is also provided for reference. Further discussion is provided below for those SAMAs identified in the follow-up RAI and listed in Table 2a-5 that were eliminated from detailed evaluation based on a screening criteria of "C" (i.e., proposed SAMA already implemented).

Modifications have been implemented to address basic events that have significance greater than 1 percent in the PRA model. In some cases, the improvement(s) did not render the contributing item completely unimportant. In such cases, further improvement is deemed not cost-beneficial, as discussed below.

<u>NMP1 – SAMA 30:</u> This SAMA involves the development of an enhanced drywell spray system, and is potentially associated with a number of basic events. These basic events include containment spray train maintenance unavailability and operator action to operate the system in intermittent spray mode.

Containment spray system maintenance is conservative in the SAMA evaluation because the average maintenance model was used, which allows all maintenance contributions to contribute simultaneously. For normal plant operation, an on-line maintenance risk evaluation process is used to balance maintenance contributions and minimize the potential for maintenance unavailability to contribute simultaneously with other maintenance activities or plant configurations. The workweek schedules are reviewed and recommendations for schedule adjustment are made, as appropriate. No other potentially cost-beneficial options were identified.

Operator action is another contributing basic event. Operator reliability is driven by many factors, including the time window available, training, stress, procedures, and instrumentation. Ranking and insights for use in operator training have been provided.

The applicable emergency operating procedures and instrumentation have also been reviewed. No other potentially cost-beneficial options were identified.

In addition, both units' containment spray design includes alternate injection sources: containment spray raw (lake) water for NMP1 and fire water for NMP2. Based on these considerations, NMPNS has not identified any potentially cost-beneficial options for further improvement.

NMP1 – SAMA 148: This SAMA notes the benefit of operator training on severe accidents including simulator training. As part of the rollout of the severe accident management procedures, operators received training including simulator review. The improvement in this area was to make operators aware of the options available, and modeled in the PRA, for severe accident mitigation. This training does not make the operators perfectly reliable and issues such as stress play a more significant role in severe accident scenarios.

The specific action of interest is the recognition that reactor pressure vessel (RPV) injection is beneficial even during core melt progression. This was specifically addressed in the recent emergency and severe accident procedures in that external sources are no longer completely inhibited during containment challenge scenarios. Current direction requires only the termination of external injection sources not needed for core cooling. The PRA has credited operator training to the extent reasonable, but operator reliability remains a key aspect of the PRA model. Based on these considerations, NMPNS has not identified any potentially cost-beneficial options for further improvement.

NMP1 – SAMA 154: This SAMA deals with providing fire pump makeup. NMP1 design currently incorporates this configuration to allow fire water to be injected to the RPV, as well as to the isolation condenser shell. The NMP1 configuration is one electric fire pump and one diesel-driven fire pump. This is the configuration modeled in the PRA. In LOSP scenarios, only the single diesel-driven pump is available. Additionally, the NMP1 fire system was provided with a crosstie to NMP2. This crosstie can be easily operated but is not credited in the PRA. The NMP2 system is similar to NMP1 and would give both plants redundancy of diesel-driven pumps for LOSP scenarios. Crediting this capability would reduce the related basic event impact to the point where no additional cost-beneficial options are viable.

NMP1 – SAMA 180: This SAMA deals with reducing operator unavailability. As discussed above, the operator training program has been provided with operator action rankings and insights. This information has been used to improve the operator training process. Operator action is expected to remain a significant contributor to the plant risk profile, and no potentially cost-beneficial options for further improvement have been identified.

NMP2 – SAMA 161: This SAMA deals with redundancy in the NMP2 vacuum breaker system. NMP2 design currently incorporates redundant vacuum breakers in each of the drywell-to-suppression chamber airspace downcomers. One basic event that contributes

a 1.2E-2 Fussel-Vesely importance to the large-early release frequency (LERF) is included in the PRA ranking. This basic event is a common-cause failure of all the vacuum breakers. Further improvement beyond current design is very costly since modifications would be required inside containment. To put this in perspective, CDF cost avoidance can be considered. A 1.2 percent change in CDF would represent a benefit of approximately $150K. Since the 1.2 percent is based on LERF and not CDF, the benefit is even lower. Considering these factors, no potentially cost-beneficial options for further improvement can be identified.

Table 2a-5 Screening Details for Selected SAMAs

ID	SAMA Title	Description of Potential Enhancement	Screening Criteria		Comments
			NMP1	NMP2	
29	Install an independent method of suppression pool cooling	SAMA would decrease the probability of loss of containment heat removal.	E	E	Cost is judged in excess of $10,000,000.
30	Develop an enhanced drywell spray system	SAMA would provide a redundant source of water to the containment to control containment pressure, when used in conjunction with containment heat removal.	C	C	Both plants have backup: NMP1 (Raw Water), NMP2 (Fire Water). See discussion for NMP1.
57	Provide additional DC battery capacity	SAMA would ensure longer batter capability during an SBO, reducing the frequency of long-term SBO sequences.	E	E	Cost is judged in excess of $1,000,000. Benefit is <10% of CDF. SAMA 215 identified as more cost beneficial option. SAMA 215 is included in the detailed analysis provided in the ER, and is listed below for reference.
58	Use fuel cells instead of lead-acid batteries	SAMA would extend DC power availability in an SBO.	E	E	Cost is judged in excess of $2,000,000.
64	Create AC power cross-tie capability with other unit	SAMA would improve AC power reliability.	E	E	Cost is judged in excess of $5,000,000. See NMP2 SAMAs 214, 215, 216, 218, and 221, which are included in the detailed analysis provided in the ER, and are listed below for reference.
106	Increase the reliability of safety relief valves. (Adding signals to add electrical signal to open automatically)	SAMA would reduce the probability of a certain type of medium break loss of coolant accident (LOCA). Hatch evaluated medium LOCA initiated by a main steam isolation valve (MSIV) closure transient with a failure of safety relief valves (SRVs) to open. Reducing the likelihood of the failure for the SRVs to open subsequently reduces the occurrence of this medium LOCA.	E	E	Cost is judged in excess of $1,000,000. Benefit is <1% of CDF.

ID	SAMA Title	Description of Potential Enhancement	Screening Criteria		Comments
			NMP1	NMP2	
110	Increase the SRV reseat reliability	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after standby liquid control system injection.	E	E	Cost is judged in excess of $2,000,000. Benefit is <1% of CDF.
148	Simulator Training for Severe Accident	SAMA would lead to improved arrest of core melt progress and prevention of containment failure.	C	C	See discussion for NMP1.
150	Improved Depressurization	SAMA would improve depressurization system to allow more reliable access to low pressure systems.	D	D	See SAMAs 215 and 217, which are included in the detailed analysis provided in the ER, and are listed below for reference.
153	Additional Active High Pressure System	SAMA would improve reliability of high pressure decay heat removal by adding an additional system.	E	E	Cost is judged in excess of $10,000,000.
154	Improved Low Pressure System (Fire pump)	SAMA would provide fire protection system pump(s) for use in low pressure scenarios.	C	D	See discussion for NMP1. Also, see NMP2 SAMA 216 included in the detailed analysis provided in the ER, and listed below for reference.
161	Improved Vacuum Breakers (redundant valves in each line)	SAMA would reduce the probability of a stuck open vacuum breaker.	E	C	Cost is judged in excess of $2,000,000 with minor benefit. See discussion for NMP2.
180	Improved Operating Response	SAMA would improve the likelihood of success of operator actions taken in response to an abnormal condition.	C	C	See discussion for NMP1.
214	Enhance SBO Procedure	SAMA would enhance procedure to provide diverse entry conditions for SBO procedure use.	C	Retain	Identified as important at NMP2 during certain electrical configurations. Detailed analysis provided in the ER.
215	Provide an alternate DC charger	SAMA would provide a portable unit for temporary alignment to divisional batteries, with procedure/training.	Retain	Retain	Included in NMP2 revised emergency diesel generator (EDG) Tech Spec submittal program. Includes SAMA 217. Detailed analysis provided in the ER.

ID	SAMA Title	Description of Potential Enhancement	Screening Criteria		Comments
			NMP1	NMP2	
216	Hard-Pipe Diesel-Driven Fire Pump (DFP) to RPV	SAMA would provide adequate flow to RPV from DFP.	C	Retain	Included in NMP2 revised EDG Tech Spec submittal program. Detailed analysis provided in the ER.
217	Provide for operation of Nitrogen supply solenoids to SRVs and Containment Vent given loss of AC power	SAMA would provide extension cord or permanent rewire to allow SOVs to be powered from uninterruptible power supply buses, plus procedures and training.	N/A	D	Considered to be included in SAMA 215. Detailed analysis provided in the ER.
218	Improve High Pressure Core Spray (HPCS) EDG Crosstie	Given loss of offsite power and divisional batteries, SAMA would provide the capability for HPCS EDG to supply Div 1 or 2. Provide enhanced procedures, training, and/or hardware to allow expedited alignment.	N/A	Retain	Detailed analysis provided in the ER.
221	Reduce Unit Cooler Contribution to EDG Unavailability	SAMA addresses EDG control room unit cooler standby failure probability.	N/A	Retain	Detailed analysis provided in the ER.

Screening Criterion Key

N/A Indicates that the proposed SAMA is not applicable to the design.

A Indicates that the proposed SAMA is related to mitigation of an Intersystem LOCA (ISLOCA). Per NRC Information Notice 92-36, and its supplement, ISLOCA contributes little risk for boiling water reactors because of the lower primary pressures. Because of the low risk contribution due to ISLOCA, this SAMA has not been developed further.

B Indicates that the proposed SAMA is related to reactor coolant pump (RCP) seal leakage. A review of NUREG-1560 indicates that although RCP seal leakage is important for pressurized water reactors, recirculation pump leakage does not significantly contribute to CDF in BWRs. For the NMP1 design, recirculation pump seals are more important and may not be screened.

C Indicates that the proposed SAMA has already been installed.

D Indicates that similar item is addressed under other proposed SAMA.

E Indicates that SAMA did not pass initial screening to move into Phase II (cost obviously exceeds benefit, not feasible, or does not provide significant benefit).

Retain Means that this SAMA is retained for a more detailed evaluation of cost benefit.

RAI 2f Follow-up Information Request

The IPE SER for NMP Unit 1 identifies the following potential improvements:

1a. Shedding the non-safety battery load
1b. Portable battery charger (for same purpose as 1a)
2. Improved calibration of low vessel pressure ECCS permissive sensors
3. Capability to locally operate certain air-operated valves
4. Increased drywell head preload
5. Modify containment venting procedure
6. Improved operator training in areas where the IPE credited recovery

The response to RAI 2f does not address items 3, 4, and 5 above, and only partially addresses item 6 and several other items. Please provide the following additional information:

a. Item 1a: Load shedding procedures for Station Blackout is stated to be covered by SAMA 211. While this SAMA addresses the DC system it does not appear to include load shedding. Please explain.

b. Item 3: The IPE discussion mentions valves to vent containment as well as to provide torus cooling. It appears that SAMA U1-212 addresses containment vent valves. Please address improvements/SAMAs related to torus cooling.

c. Item 4 was not addressed in the response. It appears that SAMA 208 may address this issue. Please explain.

d. Item 5 was not addressed in the response. Please address.

e. Item 6: The IPE SER lists improved operator training for loss of screenhouse intake. This does not appear to have been addressed by a SAMA, nor has its disposition been addressed. Please explain.

f. Item 6: The IPE lists improved operator training for loss of service water. It appears that SAMA 213 may address this issue (see RAI response 2d), but this SAMA was screened out on the basis that it was not applicable to unit design, even though the potential for procedure/training improvement was discussed in the Unit 1 IPE. Please explain.

The IPE SER for NMP Unit 2 identifies the following potential improvements:

1. Isolate standby gas treatment filters with valves
2. EOP procedure for aligning containment vent to locally open the outside purge valve when instrument air or division I emergency AC is unavailable
3. Procedure to open doors from aux building into pump rooms upon loss of cooling to HPCS, RCIC, and LPCI pump rooms
4. Guidance on opening doors and isolation of flood source

5. *Low pressure injection test and maintenance procedures*
6. *SBO procedures*

The response to RAI 2f does not address items 2 and 4 above and only partially addresses several other items. Please provide the following additional information:

a. *The IPE apparently took credit for manual valves to allow bypassing the SGTS in order to vent the containment. This improvement involves providing these valves. As noted in the RAI response, the use of blank flanges and procedure changes were implemented in lieu of the hardware modifications. It would appear that the reliability of the current approach is less than that for the proposed modification. It is noted that SAMA 219 appears to address this issue but the cost of implementation is for a fully-automatic system, which could be much more than a manually-actuated system. Please address whether a manually-actuated vent system could be cost-beneficial alternative.*

b. *Item 2 was not addressed in the response. Please address.*

c. *Item 3: The auxiliary bay pump room cooling enhancement identified in the IPE was to provide a procedure for opening the room doors. This is not part of SAMA 23 and it is not clear if it is part of SAMA 213. Please explain.*

d. *Item 4 was not addressed in the response. It appears that SAMA 223 may address a portion of this issue (i.e., control building flooding). Please address the other scenarios discussed in the IPE SER.*

Supplemental Response 2f

<u>NMP1</u>

a. Item 1a: SAMA 211 discusses a modification to allow the use of Battery Boards 12 and/or 14 (non-safety related battery board) to recover offsite power. Implicit in this SAMA is that a procedure would have to be developed to shed loads from the batteries if it is expected that these battery boards will be useful later in an accident scenario. Therefore, SAMA 211 does address the shedding of loads from the non-safety related battery. Note that implementation of SAMA 215 (portable battery charger) makes implementation of SAMA 211 unnecessary.

b. Item 3: The capability to locally operate certain air operated valves to align torus cooling on loss of instrument air was determined to not be cost beneficial prior to the performance of the SAMA analyses for License Renewal. NMPNS has re-confirmed this conclusion.

c. Item 4: After completion of the IPE, the potential improvement related to increased drywell head preload was entered into the site corrective action program for evaluation. The result of this evaluation determined that the current torque is adequate. Therefore, this issue was closed prior to conducting the SAMA analyses for License Renewal.

d. Item 5: During the original IPE, an insight was developed that involved a lower overpressure capability for the torus vent line bellows. Capacity at 40 psig with elevated temperature (i.e., 400°F) drives a failure probability in the range of 10 percent. Since the Emergency Operating Procedures (EOPs) specify a containment vent pressure of 43 psig, the IPE suggested consideration of a lower primary containment vent pressure.

   As part of the accident management program at NMP1, a list of insights for the development of the plant-specific severe accident management procedures was prepared. The primary containment vent pressure insight was included in this list, which was provided to the NMP1 Severe Accident Management Program Team.

   The severe accident management team decided not to implement any changes relative to this issue. Some considerations involved included:

   - The EOPs direct operators to vent before reaching the primary containment pressure limit (43 psig). Therefore, it is likely that current procedures would lead to a vent pressure that is adequately low.
   - Lowering vent pressure limits the time wherein core spray could be potentially available. Once venting occurs, core spray pump net positive suction head (NPSH) becomes a more significant threat to pump operation. Therefore, venting too early is not desirable.
   - Venting as late as possible provides the greatest benefit in terms of evacuation effectiveness.
   - Vent line bellows failure, while significant, represents a scrubbed release. While controlled venting of primary containment would be preferred, the vent line bellows failure mode is not the most severe containment failure mode.

   In considering the competing risks associated with this insight, NMPNS concluded that the decision to not alter the EOP vent pressure procedure was prudent based on all information available. Therefore, further action on this insight was closed. Furthermore, it was screened from the SAMA process for the same reason.

e. Item 6: The intent of the original response to RAI 2f was to address all important operator actions relative to core damage prevention. The PRA Team has provided the Training department with a list of operator actions that are modeled in the PRAs. Training has identified these actions as PRA risk significant in each applicable system training module, and they are highlighted during the training on each particular system. Therefore, these items are considered complete and were not further addressed during the SAMA evaluation.

f. Item 6: See the response to Item e above.

NMP2

a. Item 1: SAMA 219 was originally generated to evaluate making improvements to the current operator actions required to vent containment by bypassing the standby gas treatment filter trains. However, no improvements in the manual actions could be identified that would increase the operator reliability and result in a corresponding reduction in CDF. Therefore, NMPNS concluded that only a fully automated system, which is clearly not cost beneficial, would provide some reduction in CDF.

b. Item 2: Procedures for aligning containment venting to add guidance on locally opening the outside purge valve when instrument air or the Division 1 emergency alternating current power source is unavailable are addressed in SAMA 215. The noted procedures would be part of modifications that have been proposed to justify a Technical Specification change to increase the emergency diesel generator allowed outage time. Therefore, there is not a separate SAMA for this issue.

c. Item 3: The "Auxiliary Bay pump room cooling" referred to in the NRC IPE safety evaluation report (SER) for NMP2 applies to those rooms that contain the residual heat removal (RHR), high pressure core spray (HPCS), and reactor core isolation cooling (RCIC) pumps. Therefore, this issue is covered by both SAMA 23 and 213.

d. Item 4: SAMA 223 (ER Appendix F, page F-52) includes internal flooding scenarios in the Control Building from the fire water system, and in the emergency diesel rooms from the service water system, which is consistent with the discussion in the IPE SER.

RAI 3b Follow-up Information Request

In response to RAI 3b, the release fractions for each release category are provided. It is stated that these were taken from NUREG/CR-4551, Vol. 4, Rev. 1, Part 1 which is the NUREG-1150 Level 2 analysis for Peach Bottom. A review of the release fractions used for NMP SAMA indicates that only one of the high magnitude release categories (NMP2 late-high category) has a total release fraction greater than 10% for iodine. The others range from 4.8% to 9.5% and do not actually meet the definition of a high magnitude release given in response to RAI 1e (i.e., a CsI release of > 10%.) In contrast, a review of NUREG/CR-4551 indicates that many of the source terms evaluated for Peach Bottom exceed 10%, but those source terms were not adopted for NMP. It is also noted that the I and Cs release fractions provided for several of the release categories for both units are identical or differ by a factor of 2 or 10 (suggesting that the release fractions were arbitrarily adjusted.) Please describe in more detail how the release fractions were obtained and justify the apparently low release fractions for the high magnitude release categories.

Supplemental Response 3b

The basis for the PRA radiological magnitude release bins is potential offsite health impact fractions of CsI releases. Although a major effort was involved in defining the accident sequence binning with respect to magnitude and timing during the IPE development, a detailed

source term evaluation and Level 3 model was not included in the PRA. For the SAMA evaluation, a more detailed consideration of release magnitude, timing, etc. was considered. As described in the initial response to RAI 3b (NMPNS letter NMP1L 1894 dated December 6, 2004), NUREG/CR-4551, Volume 4, Revision 1, Part 1 (Section 3.3, including Tables 3.3-1 through 9) for the Peach Bottom BWR plant was used to estimate radiological releases, energy of release, and durations. Additionally, plant-specific MAAP calculations for NMP2 and plant-specific accident sequences were used to establish the source terms and timing. With regard to radiological releases, representative scenarios in NUREG/CR-4551 that match up well with the Nine Mile Point sequences were difficult to find; thus, factors of 2 and 10 were used to estimate these releases. The following summarizes the basis for defining the release bins.

- NMP2 releases were developed first since more recent MAAP calculations are available.
- NMP2 EHGH release - Based on the original NMP2 IPE, HGH magnitude releases are characterized as >10 percent CsI releases. However, more recent MAAP 4.0 estimates suggest that this simplistic approach is conservative; the magnitude could be in the range of 3 percent to 11 percent. A reasonable comparison was found in NUREG/CR-4551 for the dominant NMP2 EHGH sequence. EHGH is based on NUREG/CR-4551, Table 3.3-4, Bin 12 releases. This is judged conservative, since the CsI releases are approximately 10 percent versus the MAAP prediction of 6 percent.
- NMP2 IHGH release is based on reducing the EHGH releases by a factor of 2, except that noble gases (NG) remain the same (100 percent). This is conservative since the CsI releases are approximately 5 percent versus the MAAP prediction of 3 percent. The fact that containment fails several hours after vessel breach for NMP2 scenarios also supports some reduction. Additionally, NUREG/CR-4551 did not provide a reasonable scenario to use.
- NMP2 LHGH release is based on increasing the EHGH releases by a factor of 2, except that noble gases (NG) remain the same (100 percent). This is conservative since the CsI releases are approximately 19 percent versus the MAAP prediction of 11 percent. The fact that the reactor vessel fails before or during containment failure for NMP2 supports some increase. Additionally, NUREG/CR-4551 did not provide a reasonable scenario to use.
- NMP2 MED release - Based on the original NMP2 IPE, MED releases are characterized as 1 to 10 percent CsI releases. More recent MAAP 4.0 estimates suggest that the magnitude could be in the range of 0.7 percent (EMED) to 5 percent (LMED). All MED release magnitudes are based on an IHGH release, which has a CsI release on the order of 5 percent.
- NMP2 LO release - Based on the original NMP2 IPE, LO releases are characterized as <1 percent CsI releases. All LO release magnitudes are based on 20 percent of MED, except that noble gases (NG) remain the same (100 percent). The resulting CsI release is on the order of 1 percent.
- NMP1 EHGH release is based on NMP2 EHGH releases (~10 percent CsI), which are judged to be appropriate for NMP1.

- For NMP1 IHGH release, the containment failure scenarios are similar to EHGH, although later in time. Therefore, EHGH releases are used (~10 percent CsI).
- NMP1 LHGH release is based on NMP2 IHGH releases (~5 percent CsI), which are judged to be appropriate for NMP1 given that containment failure occurs after core damage.
- NMP1 MED release is based on LHGH releases (~5 percent CsI), which are judged appropriate given that containment failure occurs after core damage.
- NMP1 LO release is considered to be 20 percent of MED (1 percent CsI).
- NOREL is based on the following:
    - Noble gases (NG) and I are based on NUREG/CR-4551, Table 3.3-5, Bin 12. Release energy is also based on this case.
    - Cs, Te, Sr, and Ru are based on NUREG/CR-4551, Table 3.3-1, Bin 1.
    - La, Ce, and Ba are based on NUREG/CR-4551, Table 3.3-1, Bin 2.

  Although this release category has an insignificant impact on risk, the highest release fractions are taken from different tables to ensure conservatism. The NOREL case is identified in NUREG/CR-4551, Table 2.4-2, Characteristic 5 (I = no containment failure or venting).

RAI 4 Follow-up Information Request

In addressing SAMA U1-220, it is stated that conservatism exists in the model. As we understand it, implementing the SAMA by upgrades to the 4160V to 480V transformers would eliminate the need for load management actions by the operators. The model change eliminates load management failure from consideration, which appears consistent with implementation of the hardware modification. Please justify the stated conservatism in the benefit calculation. Furthermore, the response states that when a detailed cost estimate for the modification is performed, the actual cost will be higher than the value used in the SAMA analysis. Please provide a more realistic estimate of the implementation cost.

Supplemental Response 4

NMPNS believes there is potential conservatism in the modeled benefit for SAMA U1-220 due to setting operator action to perfect. Although it may be reasonable to remove this operator action with new transformers, the baseline risk could be overstated for this operator action because a single human failure basic event is used in the model for both redundant power boards. This assumes complete dependency between human failures that could lead to overloading both redundant emergency alternating current power sources. In reality, operator load management activities could be different for each emergency power division, and the timing of operator actions is likely to be different for each division. A more detailed analysis would likely reduce this total dependency assumption, which would reduce the baseline risk and potential benefit.

Further review of SAMA U1-220 reveals that the transformers identified for replacement are attached to their corresponding Power Boards 16B and 17B. Even if it is possible to replace the transformers without replacing the entire power boards, NMPNS believes that this fact alone would complicate the modification sufficiently to increase the engineering and installation costs. Therefore, NMPNS expects the conservatism in the model coupled with the additional cost considerations to exceed a factor of 3.

ATTACHMENT 3

Nine Mile Point Nuclear Station

Level B Facts and Observations That Have Not Been Incorporated into the Nine Mile Point Unit 1 PRA Model To Date

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element IE Subelement 4 Turbine Building Ventilation (TBVent) IPE screening indicates loss of TBVentilation may cause MSIV closure on high steam tunnel temperature (see p. 3.2.1-4 and Procedure NI-OP-26). This is not further developed. It is treated in the model as MSIV closure which would allow reopening MSIVs. However, the high temperature trip of the MSIVs may result in a perception that a steam line break may have occurred. This could limit the recovery or reopening of the MSIVs Without TBVentilaiton this would appear to be unlikely.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Resolve the apparent disconnect of subsuming this event in an event that does not address the dependent closure caused by the IE.
PLANT RESPONSE OR RESOLUTION
On loss of ventilation a number of alarms come in (N1-ARP-L1). The alarm response procedures direct restoration of the system and indicate that prolonged unavailability of turbine building ventilation could result in MSIV closure. It is judged unlikely that this would be perceived as steam line break based on how the scenario is expected to develop. In particular, for vent induced MSIV closure no indication of high flow, loss of RPV inventory, or high steam line radiation would be indicated. There is no explicit direction to bypass the closure signal except in case of ATWS. If the MSIVs do close, there is no direction that allows isolation bypass and re-opening of MSIVs. However, operators could open doors and restore condenser especially if needed. Specifically, if temperature can be reduced, APRs F1-1-7 and F3-1-2 allow operators to reset the RPS channels. Also, given that this would be a relatively slow moving event, they could open the door and prevent MSIV isolation before closure. While the Cert comment indicates that the event is not recoverable, there clearly is a potential for recovery and it is judged adequately represented by the current model. MSIV initiator is based on actual plant specific events; none have occurred due to loss of ventilation. Use of the main condenser or its recovery for MSIV closure in top events OD and OM have a RAW of 1.0.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element IE Subelement 9 BOC Special initiating events were discussed in the initiating event notebooks. However the following initiating events are believed to be incorrectly screened from the quantification process: • Breaks outside containment (BOC) 1. Main steam line 2. Isolation condenser steam line or multiple tube ruptures 3. Feedwater lines 4. RWCU Because of the potential for Level 2 impacts (e.g., LERF), there is not a good reason presented to eliminate from quantification. These sequences emphasize the need for isolation and its consequential importance. (See IE 8) The break outside containment (BOC) evaluation discusses that there are no breaks of high energy lines that may occur in the reactor building. The discussion for BOC appears to be in error because it neglects the possibilities of: 1) an EC line break in the reactor building 2) Massive EC tube sheet failure can also be a potential BOC contributor 3) RWCU pipe, pressure regulator, or Hx failures could also lead to a BOC The BOC can influence the LERF determination.
LEVEL OF SIGNIFICANCE
B

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

POSSIBLE RESOLUTION
Discuss the EC line break, EC tube sheet failure and RWCU failures. Include these quantitatively to address issues related to containment isolation valve operability in any applications and importance rankings. Include consideration of Level 2 and the low truncation used in the applications PSA in deciding on the retention of special initiators.
PLANT RESPONSE OR RESOLUTION
IPE Table 3.1.1-9 previously explained why these events are not risk significant. The BOC evaluation has been revised in PRA Section 5.3.3 to clarify previous inaccuracies and better explain why these events are low frequency. CDF and LERF contributions are judged to be <1E-8, which is < 1% contributor to CDF and LERF. EC isolation is modeled in Top Event EI and discussion in Section 5.3.3 has been enhanced. Future updates could consider explicit modeling of these but this is not viewed as a significant priority for the current update since the contribution is small. The addition of HELB/BOC initiators will not change results but rather provide a separate accounting of the individual contributions. Their exclusion does not impact day-to-day use of the PRA as long as the team is aware of the model's treatment.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element IE Subelement 9 ISLOCA The failure of the RWCU system and isolation valves can result in the discharge of reactor coolant through the relief valves or to the Reactor Building Equipment Drain Tank and to the torus at approximately 120 psig. This may cause severe Reactor Building environmental conditions or Torus loads.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Investigate and document the outcome of the failure of the RWCU system isolation. Consider the effect of the failure on the pressure control valve in the open position. Include the potential for environmental effects in the reactor building.
PLANT RESPONSE OR RESOLUTION
There are 2 MOVs and a fail-closed AOV that auto close on the suction path. There is an automatic MOV and 3 check valves to isolate discharge path (and pump trip is automatic). A scenario that includes over pressure of RWCU and failure of three valves is very low (<1E-7) and the impact of relief valve discharge is not judged significant because of its size. Note that the 1E-7/yr value represents an initiator frequency and not core damage or early release. CDF and LERF are several orders of magnitude lower. This was evaluated in some detail in IPE section 3.1.1. Future updates could consider explicit modeling of this but this is not viewed as a significant priority for the current update since the contribution is small. The addition of HELB/BOC initiators will not change results but rather provide a separate accounting of the individual contributions. Their exclusion does not impact day-to-day use of the PRA as long as the team is aware of the model's treatment.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element AS Subelement 9 It is not clear that the credit taken for operators keeping MSIVs open in an ATWS is appropriate. This action may be required within a relatively short time and the directions in the EOPs do not accelerate the diagnosis and action while many actions are competing for the operators' attention, and time is critical.
LEVEL OF SIGNIFICANCE
B -- Potential effect on SLC ranking as well as other functions.
POSSIBLE RESOLUTION
Reconsider the current HRA modeling approach.
PLANT RESPONSE OR RESOLUTION
This operator action is only credited when main condenser and feedwater are initially available (e.g., turbine trip); the most benign ATWS case. A 0.5 probability of failure is used because it was judged uncertain whether operators would do it in time based on operator interviews, timing considerations, and simulator experience. Increasing this to 1.0 would be insignificant and conservative.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element AS Subelement 10 SC = F (SORV) The impact of an SORV on ATWS accident sequences does not appear to be included in the rules. For SORV cases, the following detrimental impacts could occur. • condensate pumps could begin injecting as RPV pressure is lowered if they were not in PTL. • The main condenser may not be a viable method of heat removal under these conditions, i.e., ON may be higher failure probability. • CS pumps may inadvertently inject as pressure drops if CS pumps are not in PTL as assumed and cause RPV overfill and Boron washout.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Add the impact of SORV on the ATWS accident sequence nodes as noted above.
PLANT RESPONSE OR RESOLUTION
The main condenser and emergency condensers are not allowed to succeed if SORV (CNF, IV and MO are pass through, ECF). UL2 is used for operator action to control RPV level/power. In addition, there are operator actions to inject liquid poison (EP, LP), to terminate and prevent LPCS (CH), and others. Usually, operator failures in the ATWS model lead to core damage. Also, during ATWS conditions RPV will not significantly reduce pressure immediately (e.g., have impressions that reviewer think reactor will depressurize like a normal transient). N1-EOP-3 directs ATWS response. The first action is to install core spray jumpers to prevent auto-open of core spray valves (N1-EOP-1 Attachment 4). Regarding condensate, EOP-3 directs preventing FW injection via EOP-1 attachment 24. This procedure has operators close FW control valves and put FW pumps in PTL. In summary, while timing may be slightly affected, the impacts are already accounted for.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element AS Subelement 13 Model for Depressurization If the depressurization HEP is assessed as time dependent, it will become more important to separate out sequence dependencies such that the correct split fraction can be obtained. Examples include: • No EC, Seal LOCA, SORV • No EC, SORV • EC, SORV • Others
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Reconsider model structure and rules to ensure that the appropriate HEP for OD is used.
PLANT RESPONSE OR RESOLUTION
The model is already structured to allow time dependent HEPs. OD is in the model after all the conditions identified above such that OD can be conditional on just about anything, as required by individual applications. This observation appears to be in error.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element AS Subelement 17
SBO/LOSP

The response to an SBO in the model has the following characteristics:

- NSL = F (excess seal leakage) or RC = F (SORV) the ECs are not asked. This would appear to mean that ECs are assumed failed. This seems very pessimistic in modeling
- RC = F and no ECs asked, no time for AC recovery is provided beyond 1 hour
- NSL = F and no ECs asked, 2 hours is provided for offsite AC recovery.

These modeling assumptions do not appear to be explicitly discussed and there may be significant variations in the success for each depending on:

- the EC operation or not
- the size of the seal leak 25, 45, 115 or 300 gpm

These are not accounted for in the modeling

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Ensure that the SBO model accurately models events with seal LOCAs and provides a description of their treatment.

PLANT RESPONSE OR RESOLUTION

The ECs are not a success and provide only some time delay if NSL=F or RC=F. The timing treatment is conservative and additional modeling will not change results significantly because timing will still be in the 1-2 hour range (i.e., 1 hr 40 minutes does not lead to significantly different results than if 2 hours and 20 minutes is used for recovery of a particular scenario). Also, every permutation adds complexity to the model and increases run times – this must be balanced with the information provided.

Success criteria and SBO model documentation explains this, but does not explicitly describe timing conservatism in detail. The expansion of the documentation is not considered a priority for the current update as the information can be determined currently, albeit with some effort.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION

Element TH Subelement 4

The treatment of seal LOCA may be conservative. There is a reasonable likelihood that emergency condensers could be used to cool the plant for an extended period of time, potentially for 24 hours, given that there is no seal LOCA. Normal leakage is assumed to be 25 gpm. More basis is required for the assumed 25 gpm.

Generically, a more consistent and balanced treatment of the conservative/realistic plant responses to off-normal event is required for higher certification grades.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Investigate the operation of the emergency condensers within the industry to determine projected response and expected leakage. At a minimum consider a sensitivity case, included with results which addresses to potential for continued emergency condenser success over extended periods. Document the results of the investigation and/or sensitivity case.

PLANT RESPONSE OR RESOLUTION

See NSL discussion in IPE Section 3.1.2.3. In the transient and SBO models, if there is no significant leakage (NSL=S), level is assumed to remain above fuel without makeup. With excessive seal leakage (NSL=F) the model does not assume it occurs at time zero and allows a couple hours for recovery actions. Whether it is 2 hours or 8 hours is only potentially important in the SBO model. A sensitivity case was run for SBO (LOSP). CDF base case = 1.8658E-6. Allowing recovery out to 8 hrs instead of 2 hours was run as a sensitivity for NSL=F. CDF=1.6801E-6. The reduction is about 1.9E-7, which is small and based on an optimistic assumption.

We disagree that analysis is inconsistent and unbalanced and needs improvement for "higher certification grades." The NRC safety evaluation recognized the seal LOCA and leakage as the "most comprehensive for any BWR IPE..." This is clearly not Level of Significance B.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element TH **Subelement 14** No independent review of MAAP runs were available.
LEVEL OF SIGNIFICANCE
B -- Review of the model input parameters and limitations is important, because it directly impacts the results of the individual runs.
POSSIBLE RESOLUTION
Expand the thermal hydraulic review process.
PLANT RESPONSE OR RESOLUTION
Agree, Kenton-Gabor/ERIN/NMP personnel conducted several reviews, but it appears not to be well documented in one place (e.g. numerous memos back and forth between ERIN, K&G, NMP, plus review of success criteria, etc.). The PRA Update includes sign-off reviews to resolve this. However, MAAP has not been a priority for the first PRA update and this will require effort for future updates.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element SY **Subelement 5** EC The IC is designed to be vented of non-condensibles from the primary side. Without this vent, the accumulation of non-condensibles can severely restrict the decay heat removal capability of the IC.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
The failure mode of the EC that results from non-condensibles accumulation in the IC tubes should be included in the fault tree. This may entail both pre-initiator errors and hardware failures and instrumentation failures or miscalibrations.
PLANT RESPONSE OR RESOLUTION
This was evaluated during the IPE development. Accumulation of non-condensables was found to be unlikely as the ECs are declared inoperable if the vent valves are closed for any reason. Therefore, ECs would be limited to the non-combustibles generated over a short period of time. Also, the vent lines are designed to close on a containment isolation signal and are unavailable for many scenarios anyway.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element SY **Subelement 5** AS-Built Check Containment Spray The Containment Spray System modeled appears to include a number of AIR valves that have since added hand wheels to allow local manual manipulation. SVs 3 SVs have been eliminated from the plant. Procedure There may be substantial procedural changes that are not incorporated into the model. These should be included in the update.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Update the model to be current with the as-built and as-operated plant.
PLANT RESPONSE OR RESOLUTION
Containment spray valves added were related to water-seal and do not benefit torus cooling operation. Modification to the four containment spray containment isolation valves would be required and is considered an unjustifiable expense. PRA model update has incorporated the retirement of the SVs. Procedure update review was accomplished in the PRA update and is considered an ongoing effort by the PRA team as part of their normal work.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element SY **Subelement 12** AC Power/DC Power Is there a basis to allow the charger to carry all required DC electrical loads with the battery failed or severely degraded. Specifically, will the DC loads following a LOCA signal be sufficiently high to overload the charger capacity if the battery is not available. Note that a LOCA signal may be generated by events that cause high drywell pressure and/or low RPV level. In other BWRs, it is found that the charger can be overloaded if the battery is unavailable to act as a "buffer" during load sequencing. The NMP-1 design feature that addresses this should be referenced. There are also unverified hand calculations that indicate the following: • charger capacity is 400-500 amp • DBA/LOCA REQUIRES 800 AMP—therefore need the battery in the circuit
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Document in the system and fault tree model of the DC power system the technical basis that allows the DC loads to be carried by the charger.
PLANT RESPONSE OR RESOLUTION
Section 4.2.1 documents that battery demand is modeled and required to start the diesel when normal 115Kv AC power is unavailable. For the case where normal AC power is available, the battery charger can successfully supply DC power board loads without the assistance of the battery. An extremely conservative battery load (e.g., several loads assumed to occur at the same instant in time along with coincident LOOP) can be found in the calculations which is on the order of 700-800 Amps. However, these loads occur over time (e.g., LOCA signals sequenced) and the steady state battery load on the order of 300 Amps is more realistic. This is below the Charger capacity of 400-500 Amps. In the extremely unlikely case that the Charger trips, it is recoverable. We believe modeling is appropriate but better documentation should be developed as part of future updates.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element SY **Subelement 26** SW Service Water is monitored for radiation before discharge. Is there a procedure or SOP to prevent SW if radiation is detected? Has this been included in model?
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Include false and real radiation leakage indication impacts on the use of SW.
PLANT RESPONSE OR RESOLUTION
Service water is secured only if there is a high signal confirmed by a grab sample. Thus spurious signals are unlikely to lead to system unavailability. This is considered a minor contributor to unavailability of SW, ESW, EDG, and CSRW. Actual high radiation would likely only occur following CD whereupon system unavailability is considered to have a minor impact on remaining mitigative options. This issue is unlikely to significantly affect LERF as any SW radiation would likely be close in time to the PC release. However, this treatment may be slightly optimistic for longer term releases. Since the issue is minor and these releases are not a significant portion of overall risk and are not used for risk metrics, this issue is not considered a significant priority for further investigation.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 12 Operator Action to Depressurize The Human Error Probabilities (HEP) for depressurization appear to be too high and show no time dependence as might be expected from models such as the EPRI ORE data model or the Time Reliability Correlation. ZOD05 (1 hr) = 3E-3 ZOD06 (2 hr) = 3E-3 Estimates from other PSAs reviewed by the BWROG Certification effort are in the range of 2 to 5E-4 at 1 hour.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Reassess the time dependence of the depressurization HEP.
PLANT RESPONSE OR RESOLUTION
The HRA evaluated this HEP and concluded that diagnosis dominated. Blowdown is not highly time dependent as blowdown is keyed to specific plant parameters. Operators may be assisted by trending a parameter(s) over time in a case where they can anticipate blowdown over time. However, anticipatory blowdown is not allowed and regardless of time operators must wait until specific plant parameters meet pre-defined values. In this regard, diagnosis and attendant failure modes such as information overload and distraction play a role. At this time the HRA treatment is judged adequate and potentially conservative. This could be revisited with additional simulator evaluation and/or review of other plant's analyses at some later date.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element HR Subelement 12

Inconsistent HEPs

ZOU03 Long term EC makeup with OMU failure
(> 4 Hour from event start)
= 3.5E-4

This appears much too low when comparing against the results of ZOD06 and ZOD03 for depressurization at 3E-3.

Zomu1
Operator locally in RB operates EC makeup valves under SBO conditions within 30 min. at 6.4E-3. This compared with emergency RPV depressurization at 3E-3 appears substantially inconsistent. A local manual action under severely degraded conditions using a non-EOP flow chart procedure would appear to be much less likely than the control room action to depressurize.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Ensure that the HEPs modeled in the PSA provide a consistent and realistic assessment of operator action error probability.

PLANT RESPONSE OR RESOLUTION

EC makeup is over a longer term than depressurization and does not require the operators to wait for specific conditions to be met. It is also not an action with a significant "down-side" as with blowdown (i.e., blowdown represents a significant containment challenge, reduces EC effectiveness, etc.)

Local action for the EC makeup case requires actions outside the control room over a short time. These two conditions bias the action toward a higher HEP. However, operators do not need to wait for a specific cue, they can act immediately, and there is no significant "down-side" to the action. These two conditions tend to bias toward a lower HEP. With this in mind the current HRA is judged adequate.. This could be revisited with additional simulator evaluation and/or review of other plant's analyses at some later date.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR **Subelement 12** The failure probability to initiate shutdown cooling does not seem to be consistent with other HRA values. The failure probability to align shutdown cooling within 10 hours (SD01) is .0016 as compared to the failure to immediate switch the reactor to REFUEL on a scram signal (CN01) which has a HEP of .0027 or the HEP for starting the second CRD pump (OC01 = .0014). This reviewer would generally expect that the failure probability for initiating shutdown cooling would be substantially lower than either of these events. The failure probability for inhibiting ADS during ATWS (2.3e-4) also appears to be very low. Although the operators are trained on this event, there is very little time available for the performance of this task.
LEVEL OF SIGNIFICANCE
B -- Some of the values could have a significant impact on the overall results of the PSA.
POSSIBLE RESOLUTION
Consider a comprehensive comparison of all HEPs to each other and perhaps to the results of another plant and PRA team.
PLANT RESPONSE OR RESOLUTION
HRA depends on performance factors not just time. Some important factors that influence the above is whether action is knowledge based, immediate memorized action, shows up in almost all legs of procedure (e.g., ADS inhibit during ATWS), practiced frequently. The case of shutdown cooling should not imply that the operator has 10 hours from the start of the event. The operator must wait for a low pressure interlock to clear and must follow a multi-step procedure. These conditions tend to bias the HEP higher than if shutdown cooling was easy and quick to initiate. The above was developed by a credible PRA HRA expert, these are not judged as Level B Significance. This could be revisited with additional simulator evaluation and/or review of other plant's analyses at some later date.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 15 OV2 The quantification of OV node appears to be performed regardless of the time available for operator action. The large LOCA with vapor suppression failure will have an over pressure challenge to containment early in the sequence (seconds). The medium LOCA and SORV with tailpipe failure will have a containment challenge on the order of 10 to 20 minutes. It would appear that the operator action to accomplish containment spray would be substantially different for these two sequences.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
The quantitative impact is not considered significant on CDF, but individual applications could be influenced. This would indicate that the HRA must be robust enough to support the application of it in the existing PSA model.
PLANT RESPONSE OR RESOLUTION
The timing was not judged significant and a more detailed OV analysis was not conducted because it is not risk important.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 15 ADS INHIBIT WITH FW AVAILABLE (AI01) The assertion in the HRA that ADS inhibit is called for in 2 separate steps and the single step value can be squared to produce the overall HEP appears to neglect the following: • time dependence of actions • dependence of the second error on the first error • the timing evaluation appears to neglect the parallel nature of the EOPs and the possibility that FW is tripped by the operators to terminate injection
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Re-evaluate the ADS inhibit action to address these issues.
PLANT RESPONSE OR RESOLUTION
Parallel nature provides redundancies in both conducting the action as well as the number of operators that will identify the action. Numerous operators failing in several legs of the procedure was simply treated as 2 separate independent events. This appears reasonable and has not been changed.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element HR Subelement 16

CONSISTENT WITH EOP

There appear to be a number of examples where the HEPs derived for use in the model are inconsistent with the EOPs. These include the following:

- ZOV01—Initiate containment sprays given vapor suppression failure(p. 199). This action is to initiate sprays in order to control DW pressure when above 13 psig. The HEP appears to be evaluated based on the DW/T leg of the EOPs and does not address the need to initiated the sprays in a timely manner, especially for a large LOCA with vapor suppression failure. (PCP-7)
- SLC INITIATION LATE(P. 3 and assumption 4 on p. 45 of the Tier 2 HRA) The cognitive failure probability is said to be 0.0 when UL (lowering RPV level) is successful. This appears to be an incorrect assumption because the lowering of level is directed in a separate leg from the boron injection. This means that boron injection can be completely missed and is really contingent on the stress and confusion which may be present in the scenario.
- Z-CH01—Operator terminates and prevents LPI under failure to scram conditions. This HEP is quantified taking credit for multiple (2) steps in the EOPs specifying terminating injection to the RPV. There are two difficulties identified:
  1. Not all sequences that result in Emergency depressurization pass through the assumed 2 steps in the EOPs (i.e., RL-15 of EOP-3 may not have been exercised yet). This is possible for high DW/T or high torus temperature.
  2. There may be depressurization due to SORV such that neither step has been entered.

LEVEL OF SIGNIFICANCE

B

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

POSSIBLE RESOLUTION
The proposed resolutions include the following: • ZOV01—Both a qualitative and quantitative fix is required to the model. • Reconsider the HEP for the late poison to include a cognitive error for SLC initiation. • Z-CH01 – appears to have a disconnect between HRA and the sequence development. The HRA is not sufficiently robust to address the variations in ATWS sequences. Create an HEP for the above postulated scenarios.
PLANT RESPONSE OR RESOLUTION

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

ZOV01 – operators would spray if DWT>300F or if DWP>13 psig. Steps for spraying are similar and involve checking the drywell spray initiation curve which is a function of both DWT and DWP. While the documentation is not clear, the HEP quantification would not be significantly affected such that this is not considered a priority for the current update. However, this should be considered for clarification in a future update.

SLC – This is an insightful comment, however, the process of ATWS mitigation is not as "leg dependent" as the question implies. Recognition of an ATWS, as mentioned relative to lowering level, is the key cognitive factor that drives operators to EOP-3 which directs ATWS response. Operators are trained to process all EOP-3 legs in parallel which reduces the "missing a leg" contribution. Also, operators are conditioned to couple ATWS with SLC, thus further reducing the potential for additional cognitive errors once the ATWS is diagnosed. While the documentation could be clarified, the HEP quantification would not be significantly affected such that this is not considered a priority for the current update. However, this should be considered for clarification in a future update.

ZCH01 – Current EOPs have operators preventing core spray injection immediately upon entry. Also, all direction for blowdown is handled by entrance to EOP-8. EOP-8 has a specific sub-leg for blowdown under ATWS conditions which directs operators to terminate and prevent all injection except Boron and CRD. Thus, the current HEP calculation is considered adequate. For SORV, it is unlikely that significant depressurization occurs with a coincident ATWS unless power is very low. Thus either the event is a low power ATWS, Boron has been injected, or level has been reduced and it is late in the scenario. In any case core spray should have been prevented. If level has been successfully reduced, feedwater has been prevented and significant injection is unlikely, given the early direction to prevent core spray. In the case of boron, it is unlikely that boron has much effect on power unless the event is low power or level has first been reduced. For low power, the effect of injection is minor and for cases where level has first been reduced, feedwater must already be controlled. Thus the SORV case is unlikely to affect the model significantly beyond that currently included in the model.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element HR Subelement 16

LOW PRESSURE INJECTION CONTROL-ATWS (IM)

There are two potential items related to IM which are not described in the text of the IPE as if IM fails. These include the following:

- There is not a discussion of the excessive flow event where core spray is initiated but level is controlled too high and power instability occurs leading to boron wash out or prompt criticality.
- There does not appear to be a sequence related to SC=F(SORV) where depressurization will occur inadvertently and boron washout can occur if the CS and condensate pumps were not placed in PTL as might be assumed.

In addition, the accident class for these overflow failures would likely be Class IV not Class IC as currently identified.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

Either (1) add a node or (2) address overfill in this node and under flow in the WL node of the ATWS event tree.

PLANT RESPONSE OR RESOLUTION

This is an insightful comment. Currently, top event IM represents all failures in restoring RPV level after level has been reduced and boron injected. The model assumes the consequence is dominated by cases where operators do not allow sufficient flow. As the certification observation points out, this can be optimistic for the fraction of IM failures that represent overfill and thus overpower failures. A review of results shows that IM=F occurs in 2.1E-8 of CDF sequences. If the overfill failure mode of IM represents a significant fraction of IM failures, contribution to LERF could be underestimated by approximately 2E-8/yr. This is a small fraction of LERF and is not considered to represent a significant priority for the current update but this should be better modeled during subsequent updates. Contribution is likely less than 2E-8/yr since the overfill failure mode is a fraction of IM failure rate and a significant fraction, although not all, Class IC scenarios go to LERF anyway.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 17 CH02 The performance shaping factors (PSF), indications, and training are not generally included and specifically in this HEP assessment.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Include the PSFs that impact the HRA.
PLANT RESPONSE OR RESOLUTION
Performance shaping factors were treated using judgment and those that were determined to be significant were accounted for in the analysis. However, a discussion of all PSFs was not presented and the documentation does not make clear why some were judged important and others not. This is not considered a significant priority for this update but could be considered future updates (documentation issue only).

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element HR Subelement 17

Z OM02 - REOPEN MSIV IN ATWS

There appears to be substantial credit given in the ATWS evaluation for reopening the MSIVs and reestablishing the main condenser as a heat sink.

The HRA treatment of the action appears to use simple "skip a step" procedure. This would appear to be inconsistent with the EOPs which require the thorough and extensive determination of :

- fuel failure
- steam line failure
- condenser availability

All of these may be very difficult and time consuming to assess when RPV level is near or below TAF, condenser is isolated from the RPV, and MSIVs have closed. The process of reopening the MSIVs is not given any time required or time allowed.

LEVEL OF SIGNIFICANCE

B

POSSIBLE RESOLUTION

The HRA appears simplistic and in need of documented evidence that the Operations Department agrees with the timing assessment and the actions required under the conditions of the sequence. In addition, the time available and time required are vital to the model assessment.

PLANT RESPONSE OR RESOLUTION

Per the HRA Tier 2, operators were interviewed and they estimated that the action took 3 minutes. Per Table 3.2.1.4-1, time window is on the order of hours. Thus, time is not a significant factor in the HEP quantification. Also, it is noted that the RAW for Top Event OM is1.0 and this effort is viewed as a low priority.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element HR **Subelement 18**

The timing of events does not appear to be traced to calculations for the timing of events. Although the values generally appear reasonable and the level of precision appears to indicate that plant specific calculations may have been used in the analysis, it is difficult to confirm this judgement based on the available documentation.

LEVEL OF SIGNIFICANCE

B -- There is potential for errors in assumptions regarding the timing of transient progression if the traceability to specific calculations is not apparent.

POSSIBLE RESOLUTION

Review the HRA documentation and add references to thermal-hydraulic calculations, either typical for NMP1 or plant specific.

PLANT RESPONSE OR RESOLUTION

A listing of MAAP runs has been developed for Section 3.4. Timing was checked and found consistent with MAAP calculations. As pointed out, these are not adequately linked in the HRA and improvement would be beneficial. However, there is traceability to be found, with some effort, using event tree sections and available MAAP calculations. Therefore, this is considered a documentation issue with lower priority for this update. This should be worked into the workload of future updates.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 19 SBO Depressurize The assignment of split fractions for depressurization in SBO event tree uses the same probability of 3E-3 regardless of the timing conditions. This appears to relegate the time dependence of the action to a subsidiary role. This is judged an area that could use additional investigation: Shortest time could be a sequence of events with no EC, seal LOCA, SORV estimated time to core damage is 0.4 hrs (see TODD1 MAAP run). Longest time could be sequences that extend to 8 hrs. before AC recovery. It would appear inconsistent with many HRA models to find that the HEP does not vary over and 8 hr span.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Reassess the HEP used in SBO.
PLANT RESPONSE OR RESOLUTION
Blowdown is an action that takes less than 1 minute to accomplish and with a span of 25 minutes to 8 hours, the operators should feel little time pressure. It should be noted that the HEP is based on a 25-4It should be noted that the HEP is based on a 25-45 minute time window. There is ample opportunity for recovery in even the 25 minute case as numerous cues would exist (i.e., Lo-lo, Lo-Lo-Lo, ADS timer). A bigger contribution would come from either distraction (i.e., EDG recovery) or the wish to avoid downsides of blowdown (i.e., limits EC effectiveness should it be recoverable, PC challenge). There may be some additional recovery to be applied to the longer cases as TSC should be staffed, etc. but this is viewed as minor, conservative, and a low priority.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 20 The time required to complete an action is not generally stated, though by implication there is a long time available for the completion of most actions as compared to the time required. Additionally, for those cases where the time is stated, screening values appear to have been used.
LEVEL OF SIGNIFICANCE
B -- The lack of detailed information on the timing of so many actions allows a considerable opportunity for non-conservatisms in the HEPs to be introduced.
POSSIBLE RESOLUTION
Complete the timing blanks in the HEP calculation forms for all HEPs and assess the impact on the calculated HEPs.
PLANT RESPONSE OR RESOLUTION
A listing of MAAP runs has been developed for Section 3.4. Timing used for the HRA was checked and found consistent with MAAP calculations. As pointed out, these are not adequately linked in the HRA and improvement would be beneficial. However, there is traceability to be found, with some effort, using event tree sections and available MAAP calculations. Therefore, this is considered a documentation issue with lower priority for this update. This should be worked into the workload of future updates. Also, it should be noted that when the time available was long relative to time required, less attention was paid to timing issues, especially issues relating to documenting the timing issues in detail.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element DE Subelement 4 Dependency - OM The restoration of the main condenser is apparently treated on a single numerical estimate of 1.9E-2 for all transients. This value appears to be applied to: • Turbine trips • MSIV closure • Loss of condenser vacuum • Loss of service water • Loss of intake • Loss of air • Loss of TBCCW It is judged that a single estimate is not applicable across the board for these initiators and that the variation would be valuable to represent in realistically portraying the ability to reopen MSIVs and reestablish the condenser vacuum. (See QU-18)
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Provide discussion of the condenser restoration modeling for different accident pre-conditions to justify the applied recoveries. (See Shoreham PRA for typical variations.)
PLANT RESPONSE OR RESOLUTION
No, the condenser is only recovered in the IPE for the MSIV closure initiator. The event rules are complex and it is difficult to make every issue easily transparent to the reviewer. MACROs for top events OD and OM ensure guaranteed failure if condenser fails when asked in event tree, when support systems fail, for LOCAs, and loss of condenser IE. No changes necessary.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION

Element L2 Subelement 7

The binning of accidents into the level 2 analysis appears to have some weaknesses related to the usefulness of the groups in the level 2 analysis. For example:

- All SBO sequences are binned into IB. A review of these sequences indicate that many of them have adequate core cooling however, the containment heat removal fails which subsequently leads to core damage.
- The success criteria for depressurization in the L1 trees includes the use of 2 ECs (or ECs in combination with other systems). However, if the ECs become unavailable as the accident progresses, the vessel will repressurize before the vessel fails. Thus, the containment may be subjected to HPME/DCH.
- Binning of the accident sequences does not address the addition of water from outside the containment. If a substantial amount of water is added, the containment performance will be very different than if no water was added from outside containment.

LEVEL OF SIGNIFICANCE

B -- The calculated LERF may be impacted by these and other similar assumptions.

POSSIBLE RESOLUTION

Reexamine the binning of events based on variations in containment performance.

PLANT RESPONSE OR RESOLUTION

All SBO sequences go to ClassIB in the level 1. This is due to the fact that failure is dominated by cases that occur before 8 hours and involve combinations of EC failure, AC recovery failure, and operator actions. These scenarios have loss of injection type factors. However, the level 2 has macros (see CDEARLY) which bin the sequences appropriately independent of the Class1B assignment. In other words, the assignment of Class1B in L1 does not automatically direct the sequence to LERF.

The success criteria only allows ECs to depressurize under LOCA conditions and the possibility of re-pressurization under LOCA conditions is less. Also, the system analysis for ECs requires 24 hours of operation which would present re-pressurization.

Water addition to the PC is controlled by EOPs which define torus water level. In the case of PC flooding, water level is raised beyond EOP control values and the level 2 handles the cases accordingly.

No actions are required for this observation.

ATTACHMENT 4

Nine Mile Point Nuclear Station

Level B Facts and Observations That Have Not Been Incorporated into the Nine Mile Point Unit 2 PRA Model To Date

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element AS Subelement 5 The evaluation of accident sequence response using RCIC can be strongly influenced by the plant specific feature at NMP-2 of the Dikkers SRVs. The Dikkers SRVs have characteristics associated with them that result in RPV depressurization to very low pressures when the EOP direction is followed to open all ADS SRVs. Following the emergency depressurization directions results in the RPV pressure reduced to well below the pressure required for RCIC operation whether or not the low pressure trip is bypassed. This effect is to make RCIC unavailable whenever emergency depressurization is directed by the EOPs.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
The SBO evaluation and any other sequences relying on RCIC that could be threatened by implementing emergency depressurization due to requirements such as containment conditions should be reevaluated to ensure that RCIC is not credited as available in sequences where emergency depressurization would be directed.
PLANT RESPONSE OR RESOLUTION
SBO procedure N2-SOP-01 Rev 4 Cautions the Operators that "operating with RPV pressure less than 200 psig can jeopardize RCIC availability. Also, most recent EOPs (1/1/99) provide new direction (EOP-6, Attachment 29) so that depressurization does not necessarily make RCIC unavailable. Also, MAAP calculations indicate that it takes at least 4-6 hours without containment heat removal (per EOPs and operator training RPV pressure is maintained below HCTL and other containment limits) before eventual emergency depressurization may occur. Since the SBO analysis ends at 8 hours this is not an important issue.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element AS **Subelement 7** DFP The diesel fire pump alignment under SBO when essential lighting has been shed appears to be a difficult process. There is questionable evidence that the alignment can be performed and the LPCI valve opened under SBO conditions.
LEVEL OF SIGNIFICANCE
B - Verification o f the viability of the alignment within the time and environmental constraints is required to give credit in the PSA.
POSSIBLE RESOLUTION
Perform a test that demonstrates the viability under the postulated conditions or eliminate credit for the DFP.
PLANT RESPONSE OR RESOLUTION
The present model only allows a 0.5 probability of success (0.2 operator action failure). The most recent EOPs (1/1/99) ensure that the DFP will be aligned early (level below scram set point and stops in the EOPs have been removed); the operators will not wait. SBO model only allows DFP success if RCIC was successful for 2 hours. The operators practice the physical alignment and the LPCI MOVs are accessible. This has not been practiced in a SBO condition where the operators have to use flashlights. However, given that this would be done by sending operations personnel out in pairs, the above EOP changes, and timing in the SBO model, a 0.2 probability of failure is judged reasonable if not conservative. We may pursue taking more credit for the operator in the future. A separate open item was whether the DFP can protect the core 2 hours after event initiation (0.3 failure probability). Preliminary MAAP calculations indicate that a diesel fire pump with 1 of 2 injection paths is marginal. Therefore, the 0.5 S1 failure probability may not be conservative, but is still considered reasonable given our present state of knowledge. This will be considered further relative to risk management and future updates.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element SY **Subelement 7** LOSP load shedding diesel start sequence and reloading not modeled.
LEVEL OF SIGNIFICANCE
B (estimated as a non-trivial contributor)
POSSIBLE RESOLUTION
Include in update models or provide technical basis for not addressing.
PLANT RESPONSE OR RESOLUTION
This was considered during the PRA update. Failure of diesel generator load sequencing is assumed to be included in the basic events for EDG start, MOV supply operation, and circuit breaker demand. The failure of the load sequencing is considered a small contributor in comparison to the other failure modes (Section 4.2.6.11).

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 12 FW FLOW CONTROL DURING ATWS Re-establishing feedwater between 25 sec after feedwater runback ("lockout" time) and 83 sec when Level 1 is passed isolating the condenser hotwell due to MSIV closure appears to be given too much credit at 0.5.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Re-evaluate the HEP for feedwater control under ATWS conditions to preserve the condenser as a heat sink.
PLANT RESPONSE OR RESOLUTION
Re-establishing feedwater does not have to occur in the time frame suggested and it was judged that there was some chance of success. We do not believe on using 1.0 when there is an opportunity for success (based on HRA and interviews). We judge that the 0.5 value is appropriate.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element: HR Subelement: 16 The HRA analyst used a "cause based" analysis procedure (EPRI-TR-100259) for developing HHA1. This is a stress related event and the EPRI procedure is judged not to be effective in differentiating between stress and non-stress sequences. Therefore this HEP may be lower than the sequence can justify.
LEVEL OF \SIGNIFICANCE
B - This HEP should be re-analyzed early in the update of the IPE and used in update evaluations.
POSSIBLE RESOLUTION
Reanalyze this sequence before the next evaluation of the IPE model.
PLANT RESPONSE OR RESOLUTION
Disagree. There are numerous hours to perform this local action when time permits. Even the TSC could perform the action. If anything the value is considered conservative.

FACT/OBSERVATION REGARDING PSA TECHNICAL ELEMENTS

OBSERVATION
Element HR Subelement 16 LPCI/LPCS FLOW CONTROL UNDER ATWS EOP-6 Throttle ECCS Attachment 3 This appears difficult to implement and is not the procedure evaluated as part of the HRA for this action.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Re-evaluate the associated HEP and provide documentation regarding the referenced EOP and other procedures and their effect on the calculated HEP.
PLANT RESPONSE OR RESOLUTION
This action is performed after emergency depressurization and the EOPs (1/1/99) utilize LPCI A and B as the preferred ECCS trains. Throttling is available in the control room from these trains which makes the task much easier than having to apply EOP-6 Attachment 3. Even if EOP-6 Attachment 3 is needed, it is relatively easy and is performed in the control building. A re-evaluated HEP is judged unnecessary at this time.

FACT/OBSERVATION REGARDING
PSA TECHNICAL ELEMENTS

OBSERVATION
Element DE Subelement 4 Appears that the dependency matrix was constructed with plant design basis in mind, rather than the realistic (as modeled) basis for the PRA. This may be somewhat confusing for future users. Example: Noted dependency of RHR on normal AC, TBCLC and Service Water for pump seal cooling. System discussion notes assumption that seal cooling is not needed. Component Block Description tables (in system portion of the report) are good in that they define failure mode, initial state, actuated state, support system and state on loss of support. Matrix should relate to this better.
LEVEL OF SIGNIFICANCE
B
POSSIBLE RESOLUTION
Use additional notes to clarify when design basis dependencies are not required and not modeled.
PLANT RESPONSE OR RESOLUTION
No, the dependency matrix was intended to address all dependencies that the engineers could identify during the PRA development without necessarily burdening them on whether they were needed in the model. Note that seal cooing during shutdown cooling is a dependency, but shutdown cooling has not yet been added to the PRA. The Systems Analysis (Section 4.2) identifies the dependencies that are modeled and why some may not be modeled. For the time being, this will retained, although we may decide to update the dependency matrix as suggested. We are not sure whether it will add or detract from clarity; it will definitely be redundant.