NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

October 21, 2005

10 CFR 54

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

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Response to NRC Requests for Additional Information Dated August 24, 2005, Relating to License Renewal for the Palisades Nuclear Plant

References:
1) NMC letter to NRC, "Application for Renewed Operating License," dated March 22, 2005
2) NRC letter to NMC, "Request for Additional Information (RAI) Regarding Severe Accident Mitigation Alternatives (SAMA) for Palisades Nuclear Plant," dated August 24, 2005 (ADAMS Accession No. ML052370327)

In a letter dated March 22, 2005, Nuclear Management Company, LLC (NMC) requested the renewal of the operating license for the Palisades Nuclear Plant. In a letter dated August 24, 2005, the NRC issued a request for additional information (RAI) concerning the analysis of Severe Accident Mitigation Alternatives (SAMAs) performed in support of the Palisades License Renewal Application. Enclosure 1 of this letter provides the NMC response to this request.

Please contact Mr. Darrel Turner, License Renewal Project Manager, at 269-764-2412, or Mr. Robert Vincent, License Renewal Licensing Lead, at 269-764-2559, if you require additional information.

Summary of Commitments

This letter contains no new commitments or changes to previous commitments.

A112

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 21, 2005.

Paul A. Harden
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosures (1)

CC Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
License Renewal Project Manager, Palisades, USNRC

ENCLOSURE 1

NMC Responses to NRC Request for Additional Information Dated August 24, 2005 Regarding Severe Accident Mitigation Alternatives

(45 Pages)

RAI 1.a

Provide the following regarding the PSA model:

a. Provide a list of the major plant and modeling changes incorporated within each of the PSA versions listed in Section E.2.3, with an emphasis on the most recent changes in the Level 1 and 2 analyses. Also, supplement the Table in Section E.2.3 to include large early release frequency (LERF) or applicable Level 2 metrics, as applicable. (Numerous plant and PSA modeling changes since the IPE are described within Sections E.2.3.1 through E.2.3.8, but it is not clear in which version of the PSA these changes were incorporated).

NMC Response to NRC RAI 1.a

Table 1.a-1 below describes some of the major changes made to each probabilistic safety analysis (PSA) model version listed in Section E.2.3.

Table 1.a-1

Section E.2.3 Listed Model	Major Changes Made to the Listed Versions
PSAR1	Moved the internal events CDF model from SETS to SAPHIRE.
PSAR1a	The AFW alternate steam supply line to AFW pump P-8B was removed from the model as a result of a plant modification.
	Updated selected Main Steam Line Break initiating event data as well as the SGTR initiating event value.
	Selected human error probabilities (HEPs) were updated.
PSAR1b	Selected common cause failure logic for control and solenoid valves were updated.
	A plant modification that swapped High Pressure Air power supplies from MCC-7 to MCC-8 was incorporated.
	Open circuit bus faults were added to the DC system logic.
	The summertime EDG HVAC success criteria was set to True for all nominal baseline calculations.
	The independent ATWS event trees were eliminated. Transfers from all event trees to a single ATWS event tree was created, taking advantage of SAPHIRE's event tree linking options.
	DC power demand logic was added.
PSAR1b-Modified	Corrected a conservative Shutdown Cooling Heat Exchanger modeling assumption.
PSAR1b-Modified w/HELB	The model was updated to account for main steam line breaks into the CCW room(s). Steam/feedwater line breaks in the CCW rooms with door 167 or door 167B to CCW room 123 open were included. A new initiating event (IE-MSLB-D-CCW) was created to represent the steam lines downstream of the MSIVs but in the CCW room as separate from remaining lines in the turbine building.
PSAR1c (SAMA)	Diesel generator repair/recovery logic corrected.
	PCP seal LOCA model added.
	The Recirculation Actuation System plant modification was incorporated.
	HEP dependency modeling was explicitly included.
	Removed modeling conservatism in the critical SW header valve logic.
	FPS makeup to P-8C was updated to include tank T-2 failure.
	Traveling screen logic under FPS was updated.
	The auto MSIV close logic 'CHP' and 'low SG pressure' were correlated to the correct initiating event categories.

Table 1.a-1 (continued)	
Section E.2.3 Listed Model	**Major Changes Made to the Listed Versions**
	Spurious bypass valve opening was added to both single and double steam generator blowdown models.
	The gland seal condenser or air ejector after condenser rupture logic was updated.
	EQ logic was added to CCW pumps P-52A, P-52B and P-52C.
	DC bus D11-2 logic was corrected.
	Diversion path failure modes were added to selected air/N2 sources.
	The ISLOCA logic was added to the CDF model.
	Inadvertent PCS safety relief valve opening was added to the model.
	Failure of the AFW flow control valves to close was added to the system logic.
	The plant instrument air compressor modification was added to the model.
	The common cause data were updated.

The containment event tree (CET) for the most part has been unchanged since the original Individual Plant Examination (IPE) submittal. The 1993 submittal characterized the plant damage states. For example, the late containment failure state was estimated to be 14.7%; the containment intact fraction was estimated to be 46.3%, etc. Refer to Table 1.a-2 below.

Table 1.a-2: 1993 CET Results	
Containment Failure Mode	% Contribution
Containment Bypass	5.7%
Containment Isolation Failure	0.4%
Early Containment Failure	1.5%
Early Core to the Aux Building	31%
Late Containment Failure	14.7%
Late Core to the Aux Building	0.4%
Containment Intact	46.3%

Subsequent to the 1993 submittal, a modification was implemented in 1995 to delay core relocation to the auxiliary building (AB). The 1993 containment event tree (CET) logic and event phenomenological probabilities (with exception of the events modeling early relocation to the AB) remain unchanged. The revised containment failure mode distribution is listed below (Table 1.a-3).

Table 1.a-3: 1995 Plant Modification	
Containment Failure Mode	% Contribution
Containment Bypass	5.7%
Containment Isolation Failure	0.4%
Early Containment Failure	1.5%
Early Core to the Aux Building	0%
Late Containment Failure	14.7%
Late Core to the Aux Building	31.4%
Containment Intact	46.3%

Next, in support of the SAMA analysis the CET models were converted to the SAPHIRE software. In addition to the conversion process, an independent review of the phenomenology and associated probabilities utilized in the model was conducted. A brief summary of this assessment is described in the following paragraphs.

As mentioned above, Palisades' Individual Plant Examination (IPE) was completed in the early 1990s. A review of the methods used to quantify the containment event tree in the IPE revealed that in general, the treatment of severe accident phenomena was unusually comprehensive compared to other analyses conducted during the same period. In most cases, updates to the containment event tree (CET) quantification were deemed unnecessary. However, the state of knowledge in a few areas of severe accident phenomenology has changed significantly in the past decade, and these areas are worthy of an updated assessment. Also, a noteworthy change was made to the plant since the time of the IPE (the aforementioned sump modification in 1995), and this too would affect the likelihood of selected events. In general, the recommended changes to the model included;

- A reduction of the probability of reactor cavity floor failure at all assessed reactor cavity pressures,
- A reduction of the probability of reactor cavity wall failure at all assessed reactor cavity pressures,
- An increase in the probability that the core debris is in a non-coolable configuration for both wet and dry cases,
- A reduction of the probability of heat transfer from molten core debris pool in the vessel lower head,
- An increase the probability of significant core debris location to the reactor cavity after initial vessel blow down,
- A reduction of the probability that a direct containment heating pressure rise fails containment,
- Changes to several events that influence the potential for creep rupture in the hot leg, surge line and steam generator tubes,

These recommended changes were incorporated into the CETs used in the SAMA analysis. In addition, the new model included setting the 1995 sump modification to True. This event precludes early core debris relocation to the auxiliary building. Table 1.a-4 lists the SAMA CET results.

Table 1.a-4: CET SAMA Analysis Results	
Containment Failure Mode	% Contribution
Containment Bypass	16.1
Containment Isolation Failure	0.5
Early Containment Failure	1.1
Early Core to the Aux Building	0.0
Late Containment Failure	10.0
Late Core to the Aux Building	38.5
Containment Intact	33.8

RAI 1.b

Provide the following regarding the PSA model:

b. Identify the PSA version reviewed under the CEOG Peer Review. Provide a general description of the 9 Level A findings and their resolution (including the PSA version(s) in which the findings were addressed), and the 8 unresolved Level B comments and their planned closeout.

NMC Response to NRC RAI 1.b

The PSAR1a model was the analysis version reviewed by the CEOG PEER Review team. Below is a listing of the Level A and the eight now-resolved level B issues and a discussion of how they were addressed.

<u>Level A Issues and Resolutions</u>

Issue 1. *Assumption 4.7: RCP seal LOCA is neglected. This is not consistent with CEOG standard (CE-NPSD-755, Rev. 1) which has a 9E-5 seal failure probability (4 stage seal) with pump shutdown within an hour (and a failure probability of 1.0 if pumps are not tripped within 1 hour).*

Issue 1 Resolution: The PSA seal LOCA model has been developed. The analysis is consistent with the most recent CEOG guidance and the resolution of two rounds of requests for additional information (RAI) from the NRC on the CEOG guidance. This modeling change was included in PSAR1c (SAMA) model.

Issue 2. *Diesel Generator Repair: DG repair is included in Palisades' model in two basis events: DG-REC-2HR and DG-REC-4HR. These events recover one of the two DGs to enable OTC (2 hours) or to enable continued AFW flow (4 hours – indication is maintained by recovering the DGs). The recovery is applied to all failures of both DGs. DG-REC-4HR has a RAW of 2.622 and is included in 559 cutsets. DG-REC-2HR is of less importance with a RAW of 1.001 and is included in 27 sequences.*

A failure probability of 0.17 is used for both basis events. This value is based on NSAC 161; Faulted System Recovery Experience dated May 1992. It includes 6 industry events of EDG failures with one event evaluated as not recoverable.

The two recovery basis events are used to recover all failures that fail the DG Top Event. This includes support systems. For example one of the significant sequences recovered is the common cause failure of the batteries. This failure is modeled as failing both DGs and is recovered by DG-REC-4HR basis event. NSAC 161 is a limited data set and is focused on direct failures of the DGs. Recovery of these support systems appears to be well beyond the scope of the data listed in the NSAC document. In addition, a plant specific evaluation of the applicability of the industry data to Palisades was not performed.

Issue 2 Resolution: This issue was self-identified prior to the conduct of the PEER Review and was an artifact of using the SETS code.

Several analyses addressed closure of this issue. The analyses included:

- the allowable timing requirements modeled in the recovery analyses, the human recovery events employed in the updated loss of offsite power (LOOP) analysis, the LOOP IE frequency, the LOOP recovery versus time and the EDG recovery versus time, and,
- the Bayesian analyses performed to assess the plant-to-plant variability using Monte-Carlo in calculating the LOOP initiating event frequency,

These modeling changes were included in PSAR1c (SAMA) model.

Issue 3: *Indication/equipment electrical support dependencies were not implemented into each human action.*

Issue 3 Resolution: Incorporation of control room instrumentation failures into the Palisades PSAR1B-Modified-EQ Fault Tree and assessing their Impact on core damage frequency (CDF) was assessed. Including hardware in the human error probability (HEP) analysis led to little increase in the CDF (<10%). Moreover, this increase could be further reduced by crediting available redundant instrumentation for CST makeup, makeup to the SGs and recognition that a SG was empty due to a stuck open ADV.

The results of this analysis were not included in PSAR1c (SAMA) model.

Issue 4: *An evaluation of the contributors to uncertainties has not been performed. This questions the capability of the PSA to adequately evaluate the results of the quantification. Without addressing the uncertainty contributors, there may not be enough of an information "base" from which to draw conclusions as to the accuracy of the quantification results (i.e. the cutset review).*

Issue 4 Resolution: A parametric uncertainty study was performed and reinforced the conclusions of the Level 1 internal events analysis. Dominant accident sequence types were station blackout and small LOCA with failure of recirculation. Important basic events in terms of their contribution to the total uncertainty associated with the core damage frequency were also found to be associated with these two accident sequence types. Dominant human actions were similar to those that are important to the Level 1 internal events analysis.

Issue 5: *An overall uncertainty assessment has not been performed. An example of this would be a Monte Carlo distribution that addresses the uncertainty bands of the modeled events.*

Issue 5 Resolution: A parametric uncertainty study was performed and reinforced the conclusions of the Level 1 internal events analysis. Dominant accident sequence types were station blackout and small LOCA with failure of recirculation. Important basic events in terms of their contribution to the total uncertainty associated with the core damage frequency were also found to be associated with these two accident sequence types. Dominant human actions were similar to those that are important to the Level 1 internal events analysis.

Issue 6: *REI Guideline 01 is the PSA model control document. REI Guideline 09 covers the PSA Issues Database. Guideline 01 discusses PSA update types and responsibilities and tracking issues and including them in updates. Guideline 09 basically covers only use of the issues database. Neither guideline addresses which information sources that should* *be used to identify plant changes or other information that need to be addressed in PSA model updates.*

PEER Review Comment Recommendations:

1. REI Guideline 01 should be expanded to identify specific information sources that need to be routinely reviewed to identify plant changes/ plant procedure changes or emerging industry issues that should be considered when selecting PSA Model Issues or Enhancements and identifying items to be addressed in model updates. These should include plant operating events, plant design changes, maintenance procedures changes, technical specification changes, EOP/AOP/NOP changes, industry operating experience feedback and plant equipment performance information from the maintenance rule system. Establish a process for interfacing with the site's corrective action program.

2. Establish an active list of applications identifying the PRA version and a process that provides a technique to review and document these applications following each update.

3. Establish a process where timely notification of significant PRA open items is made to owners of applications that used or are using the PRA and a process to ensure appropriate actions are taken.

4. Consider the development of a PRA quality indicator that measures the number of high and medium priority issues.

Issue 6 Resolution:

1. Attachment 1, Generic Information Sources for PSA Model Development/Maintenance was added to RIE Guideline 1. The attachment is a generic listing of information sources that should be evaluated for information that supports PSA model development or maintenance.

2. Attachment 2, PSA Applications was also added to RIE. The attachment is a listing of known PSA applications that require evaluation for the impact of changes to the PSA model.

3. Attachment 3, Notification of Potential Significant Impact was added to RIE. The notification identifies an issue(s) that has been discovered that could result in potential differences in the risk results calculated by the model.

4. A quality indicator will not be implemented.

Issue 7: *Palisades has an administrative procedure, 9.14, for control of computer software, and the Nuclear Fuels department has software quality assurance plan, SQAP-029 which implements 9.14. The PSA software are covered by these procedures but the appropriate documentation has not yet been prepared to include CAFTA and SAPHIRE within the scope of these programs. These two software packages should be captured under SQAP-29 immediately. In addition, the PSA Issues database is defined as a key input to the update process. Given that this is an Access database, it appears that it might be within the scope of 9.14 and SQAP-029. This is also true for the documents database covered in REI Guideline 08.*

Issue 7 Resolution: CAFTA and SAPHIRE have been added to the NMC Software Quality Assurance Program.

Issue 8: *Human Action Dependencies. H-ZZOA-OTC-INIT, Failure to initiate once-through-cooling shows up in cutsets that include:*

1) A-AVOA-AFWFLADJ: Failure to increase auxiliary feedwater flow when feeding one steam generator

2) A-AVOA-CV-2010: Failure to align and provide make-up to the condensate storage tank

3) H-OOOT-CSTMKUP: Operator fails to makeup to CST

The LOIA initiating event has a 2.258E-9 cutset that has 3 human action recoveries and no hardware failures. All the recoveries appear to be fully dependent. The actions include H-ZZOA-OC-INIT, A-AVOA-CV-2010 and H-OOOT-CSTMKUP.

Issue 8 Resolution: Palisades performed analyses to assess the impact of the CEOG comment. The analyses were conservative and bounding. The calculated change in CDF was less than 20%, with the applied conservatisms. The conservative dependencies were included in the PSAR1c (SAMA) model.

Issue 9: *In "Database 4 –Out of Service Assumptions" on page 4 of EA-PSA-DATA-99-0004, an out of service factor are calculated based on Palisades' actual on-line hours from 1994 to 1998. This factor is essentially that fraction of a calendar year represented by 1 operating hour, given the average availability (rhe value is 1.46E-4/hr.). This factor is to be used in conjunction with maintenance out-of-service hours to calculate the maintenance unavailabilities. EA-PSA-DATA-99-0011 documents the out-of-service hours for various components between 1994 and 1998. The document presents the total out-of-service hours for each component and then calculates the average out-of-service hours per year by dividing the total out-of-service hours by 5 years multiplied by the average annual availability of 0.78. The average annual out-of-service hours calculated in this manner are multiplied by the out-of-service factor determined in EA-PSA-DATA-99-0004 to calculate the maintenance unavailabilities. However, because the out-of-service factor determined in EA-PSA-DATA-99-0004 incorporates the average plant availability, this calculation effectively credits the plant availability twice. This is somewhat conservative.*

Issue 9 Resolution: An analysis determined the out of service probability for components in the PSA model based on the out of service data collected between August 21, 1995 and August 21, 2002 (7 years). The updated out of service failure data were added to SAPHIRE .BEI file for the next PSA model update.

These data changes were included in the PSAR1c (SAMA) model.

Level B Issues and Resolutions

Issue 1: *HVAC Modeling Palisades performed a plant specific room heat-up calculations for the ESF pumps rooms, EA-C-PAL-98-1574. This analysis supported their modeling decision not to model HVAC for the ESF pump room (this analysis was well prepared and documented.) However, Palisades did not perform room heat-up calculations for the Control room or the switchgear rooms. DBD 1.06, Rev. 4 did indicate that some sort of Control Room heatup calculations had been performed which showed that on loss of HVAC, the control room temperature would reach 110 °F in 13 hours and under SBO conditions it would reach 120 °F in 15 hours. Likewise, on loss of HVAC, the cable spreading room temperature would reach 104 °F in 6 hours. These results are not consistent with the decision not to model HVAC dependencies for the control room and the cable spreading room.*

Issue 1 Resolution: Palisades performed room heatup analysis the cable spreading room. Actual test data were available for the 1C switchgear room. Per the results of the analyses and review of the test data, modeling of HVAC for these rooms was considered unnecessary. Per review of the control room heat loads, the high design temperature limits of the components, the general conservative nature of the analysis, as well as the un-importance of HVAC in other PSA's, inclusion in the Palisades analysis was not considered warranted.

Issue 2: *The event tree analysis provides a pointer to references for success criteria utilizing thermal-hydraulic calculations (MAAP runs). The specific MAAP runs have not yet been approved or checked that they have indeed been run for the appropriate boundary conditions. The few human action analyses that were looked at had specific MAAP runs to justify the timing basis utilized in the human action. However, all human actions were not verified.*

Issue 2 Resolution: At the time of the PEER Review, the figures correlating the MAAP runs to the human error analyses were missing from the hard copy. Subsequent to the PEER Review, the figures were reconstituted. Moreover, the Risk Informed Engineering PSA Event Tree Notebook has been released. This document includes additional MAAP analyses that have been examined. The key data are noted in this document and are correlated to the human error models, success criteria etc.

Issue 3: *No guidance could be found applying to dependency determinations.*

Issue 3 Resolution: RIE Guideline 01 Control of the Plant PSA Model was revised (revision 7) to include;

1] guidance on the types of dependencies to be included in the PSA models,
2] establish how the dependencies are identified,
3] methodologies to generate probabilities for specific dependent events that are incorporated into the model, and
4] reference to other documents that implement dependency analyses (common cause analysis, human error analysis, etc.).

The guidance on dependency determinations was included as section 6.0 of the guideline. The revision of the guideline has been reviewed and approved.

Issue 4: *EA-PSA-DATA-99-0009 presents the Palisades specific failure data for mechanical components for the time interval from 1994 to August 1999. This document presents a total of 21 failures. This seems to be somewhat lower than would be expected given the component population and the time window. The data is based on the maintenance rule functional failure counts. Palisades may want to check to ensure that the maintenance rule functional failure definitions are consistent with the PSA failure definitions and confirm that failures were not missed based on definition differences.*

Issue 4 Resolution: A review of the Maintenance Rule data included a determination whether the failures in the Maintenance Rule data should have been considered for equipment that is included in the PRA or could affect the initiating event frequencies currently used in the PRA. The Maintenance Rule data included 286 failure events. Of these ARE identified 62 that were determined to be applicable for use in updating the PRA data. An additional 11 were identified as having a potential to impact PRA data, but further evaluation of the source of the information would be required to make a final determination.

As a result of the review, the issue identified in the PEER Review is not the concern. The issue is why all of the indicated failure information was not used in the PRA data update. A review the analysis determined that there is no information included to justify any information not used or why only certain failure rates were updated.

As a result there is no reason to believe that the Maintenance Rule Functional Failure reports do not provide a reasonable source of information for the PRA data updates. It is also understood that the Maintenance Rule Functional Failure information would not necessarily be all encompassing and that other sources (e.g. corrective action, work order history, etc) should also be examined to determine whether other failure information exists that should be included in the data update. Data not used in EA-PSA-DATA-99-0009 will be included in the next PSA data update. No further action regarding the PEER Review finding was determined necessary.

Issue 5: *Currently there is no plant specific data for electrical type components. There is generic electrical failure data.*

Issue 5 Resolution: Use of plant specific data is not required by any of the guidance documents (e.g. ASME RA-S-2002, AMSE Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, or NRC RG 1.200, An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities) for conducting PRAs. The ASME Standard states, "The parameter estimates shall be based on relevant generic industry or plant specific evidence. Where feasible, generic and plant specific evidence shall be integrated using acceptable methods to obtain plant specific parameter estimates. Parameter estimates for the important parameters shall be accompanied by a characterization of the uncertainty." The NRC guidance in Regulatory Guide 1.200 included an evaluation of the current guidance including the ASME standard. The evaluation of the standard did not identify any objections to the statements presented above regarding use of data in risk assessments.

Use of plant specific data is recommended in particular for cases where component performance is not within the expected range of the industry data. However, the poor performance could also be accommodated in the model by adjusting generic data to reflect the extent that component performance deviates from the mean of the generic industry data. Palisades has used plant specific electrical data in the past for certain components (e.g. diesel generator output breakers). In general, there has been no recent indication that electrical component performance within specific groups (e.g. 480VAC breakers, 2400VAC breakers, etc.) is substantially different from industry data. The replacement in recent years of many breakers to important plant equipment represents a condition in which the initial performance of the breakers is best represented by generic data until the equipment has been in place for a sufficient time to establish a component specific performance history.

An update of the parameters used in the PRA is planned. In the process of completing the data update consideration will be given to the need for development of plant specific data for electrical components.

Issue 6: *The electronic documents identify the code files used for sets solution, others identify the Definitions and Basis for Event Tree Top Headings for quantifying the model with SAPHIRE, but no single document provides clear instruction for the quantification process.*

Issue 6 Resolution: Palisades documents the files used in developing and quantifying the PSA logic models according to plant administrative procedures. In addition the Risk Informed Engineering Quantification Notebook has been developed that describes the quantification process.

Issue 7: *In Attachment 1 to EA-PSA-DATA-99-0004, Rev 00 (draft), two component failure modes are given for each component failure listed in the attachment. For example, for ADMT, the failure modes are "Fails to Run" and "Loss of Function/fails to perform function". It appears that the second failure mode description is intended to provide more detail for the first failure mode. In most cases, the first failure mode seems appropriate and adequately descriptive. However, in a number of instances, the second failure mode description contains inappropriate failure modes for the component failure of concern. For example, CRMJ pertains to "Fittings, cooler", a mechanical component. The first failure mode is "External Leakage/Rupture". However, the second failure mode is "Short circuit/line to ground/leak/rupture". The "short circuit/line to ground" is inappropriate for the specific component type and should be deleted. The document does show a strikeout for this. The second failure mode should be reviewed for all component failures to ensure that all of the cited failure modes are appropriate for the component type.*

Issue 7 Resolution: The original PRA development included a listing of component type designators (2 letter code) and failure mode designators (2 letter code). The combination of component type code, failure mode code, system code designators and component specific identifiers allowed unique basic event identifiers to be created for use in the PRA model. In addition, listings of generic data to be used in conducting risk analyses were generated by creating a listing that included a combination of component type codes and failure mode codes with applicable data to be used. The failure mode codes were developed to be dual purpose codes and many had two descriptions. The same code could represent an electrical component failure mode or a mechanical component failure mode. Since the listing of generic data or the creation of basic event identifiers was accomplished manually it was the individual analyst's responsibility to assure that the appropriate description was used. In EA-PSA-DATA-99-04 an attempt was made to automate some of the data generation process via a database. However, as noted there were many instances where incorrect descriptions were applied to the developed listing of generic data and were not corrected. The listing of generic data was extracted from the database used in EA-PSA-DATA-99-04 and a corrected table was created and included as attachment 1 to Risk Informed Engineering Guideline 03, Generic Data, revision 1.

Issue 8: *The PSA documents are covered by Administrative Procedure 9.11. This procedure requires sign-off of Engineering Analyses by the initiator, the technical reviewer and an approver. This is the procedure applied to all engineering calculations. Not all of the*

PSA calculation documents have been completed and thus do not have the appropriate signatures in place. The completed calculations reviewed did have the signatures.

Issue 8 Resolution: The problem was not that there were documents without all required signatures completed at the time. The normal process of updating the PSA and implementing the engineering analysis procedure will result in cases where there are analyses that are in process for a particular update that do not currently have all signatures. In addition, there can be analyses that do not directly support a model change. Sensitivity analyses may be conducted to demonstrate that issues raised would not require a change. The concern should be that the analyses that support a particular change or update to the PSA model should be complete with all signatures in place prior to finalizing the model and approving it for use. It is unclear from the comment what analyses did not have all signatures and whether they were for the approved model at the time of the review or for an ongoing update to the model. Presently, the analyses that support the current version of the model have been approved. Analyses that support the ongoing update to the model are complete or in the review process. The analyses in the review process that support the ongoing update are scheduled for completion prior to implementing the update. Therefore this issue no longer exists and is considered resolved.

RAI 1.c

Provide the following regarding the PSA model:

c. Provide a breakdown of the population dose (person-rem per year within 50 miles) by containment release mode in the following form, or equivalent

NMC Response to NRC RAI 1.c

Table 1.c below lists the requested information:

Table 1.c

Containment Release Mode	Population Dose	% Contribution
SGTR	7.5	23.9
ISLOCA	9.70E-07	≈0.0
Early Failure	1.6	5
Intermediate Failure	0	0
Late Failure	0.26	0.9
No Failure	0.57	1.8
Basement Failure[1]	21.3	67.8
Containment Isolation Failure	0.19	0.6

[1]NOTE: Although this was discussed during the teleconference with the NRC Staff and recognized as a typo nevertheless the "Basement" failure mode was repeated in the above formal request. For purposes of this evaluation, it is considered that the question intent was with respect to a "Basemat" failure mode. Moreover, this would only be an issue if the high-intermediate releases that occur due to delayed sump failure were eliminated. Therefore this failure mode is interpreted as a delayed sump failure.

RAI 1.d

Provide the following regarding the PSA model:

d. The baseline frequency for release category "L-L" appears to be erroneously reported as 4.37E-8 per year in Section 2.5.5.5 and Tables E.3-4 and E.3-5. The correct value appears to be 4.37E-6 per year, as reported in the individual tables in Sections E.6 and E.7. Confirm the correct value and address any impacts on the SAMA analysis.

Resolved via August 9, 2005, teleconference: The applicant confirmed that 4.37E-6 per year is the correct baseline frequency value. No further response is required.

NMC Response to NRC RAI 1.d

Included for completeness. No additional response is required.

RAI 2.a

Provide the following concerning the MELCOR Accident Consequences Code System (MACCS2) analyses:

a. In Section E.3.5 it is stated that plant-specific data was used based on ORIGEN2.1 calculations. Please elaborate on how plant-specific data was used. If Palisades specific fuel burnup/management data was not used, provide an evaluation of the impact on population dose and on the SAMA screening and dispositioning if the SAMA analysis were based on the fission product inventory for the highest burn-up, fuel enrichment and power level expected at Palisades during the renewal period.

NMC Response to NRC RAI 2.a

New ORIGEN data developed in 2004 were used in support of NRC GL-2003-01. The data was subsequently applied to the MACCS2 analysis. Best estimate, 23 GWD/MTU (796 days ~ 2.6 months) data were used. The Palisades cycle duration is expected to remain at 18 months.

RAI 2.b

Provide the following concerning the MELCOR Accident Consequences Code System (MACCS2) analyses:

b. Identify and briefly discuss the key MACCS2 input assumptions or other factors that contribute to the offsite economic cost risk at Palisades, e.g., per diem cost for relocated individuals, the costs to relocate an individual, and the value of farm and non-farm wealth.

NMC Response to NRC RAI 2.b

Table 2.b below lists the Sample MACCS2 Economic Parameters:

Table 2.b Sample MACCS2 Economic Parameters

Variable	Description	Palisades Values
DPRATE(1)	Property depreciation rate (per yr)	0.2
DSRATE(1)	Investment rate of return (per yr)	0.12
EVACST(2)	Daily cost for a person who has been evacuated ($/person-day)	41.58
POPCST(2)	Population relocation cost ($/person)	7700
RELCST(2)	Daily cost for a person who is relocated ($/person-day)	41.58
CDFRM0(2)	Cost of farm decontamination for various levels of decontamination ($/hectare)	866.25 1925
CDNFRM(2)	Cost of non-farm decontamination per resident person for various levels of decontamination ($/person)	4620 12320
DLBCST(2)	Average cost of decontamination labor ($/man-year)	53900
VALWF0(3)	Value of farm wealth ($/hectare)	2273
VALWNF(3)	Value of non-farm wealth ($/person)	95129

(1) DPRATE and DSRATE are based on NUREG/CR-4551 value
(2) These parameters for Palisades use the NUREG/CR-4551 value and updates them to the 2000 CPI value
(3) VALWF0 and VALWNF are based on SECPOP2000 values for Palisades

RAI 3.a

Provide the following regarding the SAMA identification process:

a. Table A-1 of the Addendum to Appendix E of the ER provides a list of 266 potential SAMAs that were used to help identify potential enhancements for selected functions at Palisades. However, it is not clear from Appendix E of the ER how this list of SAMAs was specifically used to identify candidate SAMAs for evaluation. Briefly describe how the information in Table A-1 was used in the identification of SAMAs, including the rationale or criteria for eliminating each of the items in from consideration as a Palisades SAMA.

NMC Response to NRC RAI 3.a

Table A-1 of the Addendum to Appendix E of the environmental report (ER) is not part of the Palisades SAMA screening process. The plant improvements listed in Table A-1 were used only as a source of ideas for the types of enhancements that could be proposed to address the plant specific insights that were identified for Palisades.

For example, review of the PSA demonstrated that station blackout (SBO) was a large contributor to the Palisades risk profile. Table A-1 was reviewed to determine if the industry had already devised a means of reducing SBO risk that would address the issues that made SBO important for Palisades. In this case, the industry had developed multiple methods of reducing SBO risk and several of these were applicable to the Palisades issues identified by the plant specific PSA review. These plant enhancements were then adopted as Palisades SAMAs and included in the SAMA screening process. For the Palisades issues that were not adequately addressed by items included in Table A-1, the NMC staff developed new methods of reducing plant risk and included them on the Palisades SAMA list for evaluation (e.g., insulating the EDG exhaust ducts).

The use of Table A-1 is considered to be beneficial because it reduces the resources required to develop the Palisades specific SAMA list and it provides access to innovative plant enhancements that might not have been derived independently.

RAI 3.b

Provide the following regarding the SAMA identification process:

b. Two events in ER Table E.5-1 have a very large risk achievement worth (RAW), as estimated by the staff (i.e., RXC-MECH-FAULTS and RXC-ELECFAULTS). In the case of the mechanical faults, the staff estimates that an order-of-magnitude increase in this event alone would increase the CDF to 6.8x10-5. Provide an assessment of the value of ensuring that these RPS subsystems do not degrade with time, and whether or not a SAMA is warranted to ensure these subsystems do not degrade.

NMC Response to NRC RAI 3.b

The Palisades reactor protection system (RPS) is the single most important plant critical safety function, with controls and tests too numerous to mention. Given the numerous surveillances, etc., any observable system/component degradation should be detected early.

The values used for these events are generic data developed for Combustion Engineering Plants. The value used for RXC-MECH-FAULTS is the upper bound of the CE recommended values for mechanical failures. Current and historical plant performance has not indicated that there is any reason to believe that the plant performance would be expected to be different from the CE fleet performance. While there may be demonstrable value in assuring that there is no degradation in performance over time, these components are routinely tested to assure that they are capable of performing their design function. Again, observable degradation should be discovered as a result of the testing process. Therefore what would be considered appropriate as a SAMA is considered to be in place.

RAI 3.c

Provide the following regarding the SAMA identification process:

c. Appendix E of the ER indicates that SAMAs from Calvert Cliffs, also a Combustion Engineering plant, were reviewed for applicability to Palisades. However, none of the potentially cost-beneficial SAMAs identified in the Calvert Cliffs evaluation (NUREG-1437, Supplement 1) made it out of the generic list of industry SAMAs (Table A-1) and onto the list of Phase I SAMAs (Table E.5-3). The three potentially cost-beneficial SAMAs for Calvert Cliffs are:

i. Change undervoltage, AFW actuation signal (AFA) block, and high pressurizer pressure actuation signals to 3-out-of-4, instead of 2-out-of-4 logic (SAMA 112 in Table A-1, SAMA 48a in Calvert Cliffs license renewal application).
ii. Implement internal flood prevention and mitigation enhancement (e.g., watertight doors) to prevent flood propagation (SAMA 155 in Table A-1, SAMA 66b in Calvert Cliffs license renewal application).
iii. Automate demineralizer water make-up to the CST and provide a dedicated diesel generator for this purpose (SAMA 172 in Table A-1, SAMA 74 in Calvert Cliffs license renewal application).

NMC Response to NRC RAI 3.c

The evaluation of these potentially cost beneficial SAMAs is discussed in the following:

i. The importance list did not disclose any instances where the level of redundancy for actuation logic was an issue.
ii. The Palisades flood CDF update was E-07. This Calvert Cliffs issue is considered not applicable to Palisades.
iii. Current Palisades design does include automatic makeup from the demineralized water tank (T-939) to the condensate storage tank (CST) (T-2). The makeup function provides assurance that the CST will be maintained at its required level. However, the equipment to support this function is not supplied by a bus that is supplied by a diesel generator. It is possible at the time of CST depletion to align the power to this equipment to a safety related power supply that would be supplied by a diesel generator.

RAI 4.a

Provide the following regarding the Phase I screening:

a. Page 4-31 of the ER indicates that 16 candidate SAMAs remained after the Phase I screening, whereas Table E.5-4 identifies only 9 SAMAs. Resolve the discrepancy.

Resolved via August 9, 2005, teleconference: The applicant indicated that the text on page 4-31 is in error. Nine SAMAs remained after Phase I screening, as indicated in Table E.5-4. No further response is required.

NMC Response to NRC RAI 4.a

Included for completeness. No additional response is required.

RAI 4.b

Provide the following regarding the Phase I screening:

b. SAMA 12 addresses several events in the Importance List Review tables (Tables E.5-1 and E.5-2) but was not retained for Phase II analysis due to it being a BWR mitigation feature. However, this SAMA proposes modifying the existing CVCS injection system to automatically operate during ATWS, and would seem to have applicability to a PWR. Provide further discussion of why SAMA 12 is not retained, including a cost estimate. In this discussion, consider the collective impact of all items in the Importance List Review tables that refer to SAMA 12.

NMC Response to NRC RAI 4.b

SAMA 12 was identified as a potential means to reduce risk at Palisades based on the importance of three events and the cutsets that include them:

- IE_TRANS-WC: Transient with main condenser available
- RXC-MECH-FAULTS: RPS mechanical failure – CRDS
- RXC-ELEC-FAULTS: RPS electrical failure

As indicated in the ER submittal, the frequency and failure probabilities for these events were overestimated in the SAPHIRE PSA SAMA model, which artificially inflates the importance of the events. In addition, some hardware changes that have been performed at the plant are not credited in the SAMA model. When these facts are considered together, no ATWS related SAMAs are considered to be potentially cost beneficial for Palisades, as discussed below.

Several items have been identified that would reduce the estimated importance of ATWS if they were incorporated into the PSA. These items include both data updates and hardware changes:

- Volume 10 of NUREG-5500 provides updated information on the reliability of the "Group 1" RPS design that exists at Palisades. Based on the results presented in Table 3-4 of that document, the failure probability of the mechanical fault portion of the RPS (RXC-MECH-FAULTS) is 8.4E-7 compared with the 1.0E-5 used in the Palisades SAMA analysis.

- The failure probability of the electrical component of the RPS (RXC-ELEC-FAULTS) provided in NUREG-5500 is 4.81E-6, which is larger than the 2.0E-6 failure probability used in the Palisades SAMA analysis. However, this revised estimate does not account for the incorporation of the alternate scram capability implemented at Palisades in the 1990s. While NUREG-5500 assesses the impact of the manual scram function, it does not credit manual scram for Group 1 RPS because the dominant failure mode of the electrical component of RPS also disables manual scram actuation (common cause failure of the trip contactors prevents both the automatic scram signal and the manual scram signal from removing power to the clutch power supplies). The Palisades ATWS modification included installation of diverse scram circuitry such that the trip contactors are not required for manual scram success. Table 3-4 of NUREG-5500 shows that crediting manual scram reduces the failure probability of non-Group 1 RPS by factors which range from about 5.8 to 7.4. If similar credit is assumed to be available for the Palisades diverse manual scram system, the event RXC-ELEC-FAULTS could be reduced by a factor of 7 from 4.81E-6 to 6.87E-7.

- The Palisades initiating event frequencies have been reevaluated since the performance of the SAMA analysis. The initiating event frequency for transients with the main condenser available, which is the relevant event for SAMA 12, was intentionally set to a conservatively high value of 2.0/yr. The current best estimate for this value is 1.97E-1/yr, which is an order of magnitude lower than the previous assessment.

When these values are applied to the PSA, the risk reduction worth (RRW) estimates of the events discussed above are greatly reduced compared to the values presented in the ER submittal. Table 4.b following, summarizes these changes.

Table 4.b

Basic Event ID	Description	ER Submittal RRW	Revised RRW
IE_TRANS-WC	Transient with main condenser available	1.081	1.002
RXC-MECH-FAULTS	RPS mechanical failure – CRDS	1.081	1.003
RXC-ELEC-FAULTS	RPS electrical failure	1.039	1.007

Based on the revised RRW importance calculations, none of the events are above the 1.01 RRW review threshold applied in the SAMA analysis. As demonstrated in the ER, the RRWs for these events indicate that further efforts to reduce ATWS risk would yield averted cost-risks of not more than about $50,000.

It should also be noted that the turbine driven AFW system was modified as part of the ATWS modification to initiate on loss of DC power such that it would be available in an ATWS even if DC power is lost during the accident. This change, in conjunction with the addition of the diverse scram circuitry, represents the considerable effort expended by Palisades to address the largest contributors to ATWS sequences. No additional plant enhancements have been identified that would result in any meaningful reduction in ATWS risk.

RAI 4.c

Provide the following regarding the Phase I screening:

c. The discussion of Phase I SAMA 19 defers to Phase I SAMA 17. SAMA 17 addresses the failure of valves CV-3070 and CV-3071 due to filter plugging. It is not clear that filter plugging is the dominant initiating failure of these valves in SAMA 19. Please discuss.

NMC Response to NRC RAI 4.c

The plugging of filters in the air supply to these valves is the dominant failure mode of the valves in all cases. Since they are included as separate events (from the valves) they were addressed separately in SAMA 17. SAMA 19 describes the valves failure mode that show up lower in the importance listing. SAMA 19 addresses the design that each valve is aligned to a particular pump, and that including a cross-tie to allow either valve to supply either pump could be an improvement. However, the valves are still subject to the failure of the air supplies due to plugging of the filters. In addition, the current results do include the failure of both valves, and the only real benefit from a cross-tie would be to eliminate the combinations where a valve to one train fails and the other train fails for other causes.

RAI 4.d

Provide the following regarding the Phase I screening:

d. For Phase I SAMA 20, existing procedures to prevent traveling screen failure are assumed to be adequate. Re-evaluate the RRW given the operator action to ensure this event can be eliminated from consideration for a SAMA and address again accordingly.

NMC Response to NRC RAI 4.d

There is not currently a developed human error to include in the analysis. A sensitivity calculation was conducted assuming a screening value of 0.3 for human error. Since only the common cause term for the traveling screens is present in the results and is the basis for the SAMA evaluation, the probability of the common cause term was adjusted to reflect the impact of being combined with the human error. The results were reevaluated with the modified probability, and the RRW for the event fell below the criteria (RRW >= 1.01E+00) for SAMA consideration.

RAI 5.a

Provide the following regarding the SAMA cost estimates:

a. Provide a brief description of the methodology and major cost elements used to develop the cost estimates for the 23 Phase I SAMAs (e.g., was the estimate developed by Palisades or obtained from another source, does the estimate include the cost of replacement power during extended outages required to implement the modifications, does the estimate include recurring maintenance and surveillance costs or contingency costs associated with unforeseen implementation obstacles).

NMC Response to NRC RAI 5.a

An engineering firm was contracted to determine cost estimates as described below. Cost estimates for potential plant modifications identified in the SAMA analysis have been developed as order-of-magnitude cost estimates. Project descriptions from the SAMA analysis were expanded to develop cost estimates which applied considerations, assumptions and criteria as discussed below.

Each cost estimate was broken down into relevant work activities across the following major project phases: Study, Analysis, Design, Implementation, and Life Cycle. These estimates do not include replacement power costs.

A top-down approach was used based on past experience in providing proposals and estimated costs for plant modifications at several nuclear power plants. A bottom-up approach was not generally needed for order-of-magnitude type estimates. At the early stage of project conception, a bottom-up approach is typically not practicable due to the extent of uncertainties inherent in the specific project descriptions. Such uncertainties are normally characterized and reconciled as part of an initial study effort for each potential design change.

Work activities associated with the various project phases as described below were considered with respect to the expanded SAMA project descriptions.

Cost estimates for the 'Study' phase of each project account for consideration of physical design change alternatives, identification of stakeholders, pre-conceptual design, assessment of impact on plant procedures, processes and programs and a draft safety evaluation or licensing / permitting assessment.

Estimates for the 'Analysis' phase of each project account for evaluations, calculations and analyses required to support the basis for the project such as revisions to the heat balance or plant accident analysis.

Estimates for the 'Engineering and Design' phase of each project account for conceptual design, preliminary design and final design. This involves preparation, review and approval of drawings, specifications, data sheets, design change packages, as well as various discipline engineering elements and engineering program elements. Also included are evaluations, calculations and analyses required to support the implementation of the design change such as piping analysis, pipe support calculations, structural load analyses, electrical circuit analyses and loading, cable tray loading, etc.

Estimates for the 'Implementation' phase of each project account for procurement, materials management, work planning, installation, contingencies, testing, return to operations and closeout. This involves maintenance services, construction services, craft labor, design engineering support, program engineering support and procurement services.

Estimates for the 'Life Cycle' phase of each project include surveillance and recurring maintenance costs, and account for labor and materials required for maintaining plant equipment in operable condition for 20 years.

RAI 5.b

Provide the following regarding the SAMA cost estimates:

b. The cost of implementing Phase I SAMA 21 is given as $7,000,000. The cost of implementing a similar SAMA at Brunswick was estimated to be $100,000. Provide a further explanation for the significant cost associated with this SAMA.

NMC Response to NRC RAI 5.b

At Brunswick this was a procedure change. At Palisades this is a major modification. The Palisades cost estimate for this modification is described in the following Table 5.b.

Table 5.b: Order-of-Magnitude Cost Estimate				
Phase	**Item**	**Resource**	**Functional Area**	**Dollars (2005)**
Study / Analysis	1	Labor	Engineering Design	$150,000
	2	Support	Engineering Programs	$50,000
			Subtotal	**$200,000**
Design	3	Contract Labor	Engineering Design – Mech	$600,000
	4	Contract Labor	Engineering Design – Mech MCR	$100,000
	5	Contract Labor	Engineering Design – I&C	$300,000
	6	Contract Labor	Engineering Design – Elec	$300,000
	7	Contract Labor	Engineering Design – Elec MCR	$200,000
	8	Contract Labor	Engineering Design – Structural	$400,000
	9	Labor	Engineering Programs	$100,000
			Subtotal	**$2,000,000**
Implement	10	Labor	Maint. & Constr. Services	$2,300,000
	11	Labor	Maint. & Constr. Services - MCR	$900,000
	12	Materials and Trans	Materials & Material Management	$800,000
	13	Contract Labor	Engineering Design (Constr. Support)	$200,000
	14	Support	Engineering Programs	$100,000
Life Cycle	15	Labor & Materials	Periodic inspection and repair for 20 years	$500,000
Grand Total			Exceeds MMACR of $5,630,000	$7,000,000

RAI 6

For Phase II SAMA 3 and SAMA 4, provide a more detailed description of the PSA model changes made to reflect the SAMA implementation. Include the original and modified failure probability values for each component assumed to be impacted by the SAMA.

NMC Response to NRC RAI 6

SAMA 3 and SAMA 4 are discussed in detail below.

For SAMA 3, the model was modified by reducing the existing pump failure probabilities to simulate the addition of a direct drive diesel injection pump (DDDIP) considering that the failure rate for the DDDIP is comparable to the existing pumps. In addition, the pump common cause failure (CCF) terms and random system failures were removed to represent the independence of the pump. Table 6.a summarizes these changes.

Table 6.a

Basic Event ID	Description	Original Probability	SAMA 3 Probability
A-PMME-P-8A	AFW PUMP P-8A FAILS TO START	1.65E-3	1.9895E-4
A-PMME-P-8B	AFW TURBINE PUMP P-8B FAILS TO START	1.72E-3	1.9895E-4
A-PMME-P-8C	AFW PUMP P-8C FAILS TO START	1.65E-3	1.9895E-4
A-PMMG-P-8A	AFW PUMP P-8A FAILS TO RUN	2.13E-3	1.1346E-4
A-PMMG-P-8B	AFW TURBINE PUMP P-8B FAILS TO RUN	8.11E-4	1.1346E-4
A-PMMG-P-8C	AFW PUMP P-8C FAILS TO RUN	2.13E-3	1.1346E-4
A-PMCC-P8AB-ME	COMMON CAUSE FAILURE OF AFW PUMPS P-8A AND P-8B TO START	7.40E-6	0.0E0
A-PMCC-P8AB-MG	COMMON CAUSE FAILURE OF AFW PUMPS P-8A AND P-8B TO RUN	9.91E-6	0.0E0
A-PMCC-P8ABC-ME	COMMON CAUSE FAILURE OF ALL 3 AFW PUMPS P-8A/B/C TO START	2.41E-6	0.0E0
A-PMCC-P8ABC-MG	COMMON CAUSE FAILURE OF ALL 3 AFW PUMPS P-8A/B/C TO RUN	1.49E-6	0.0E0
A-PMCC-P8AC-ME	COMMON CAUSE FAILURE OF AFW PUMPS P-8A AND P-8C TO START	7.40E-6	0.0E0
A-PMCC-P8AC-MG	COMMON CAUSE FAILURE OF AFW PUMPS P-8A AND P-8C TO RUN	9.91E-6	0.0E0
A-PMCC-P8BC-ME	COMMON CAUSE FAILURE OF AFW PUMPS P-8B AND P-8C TO START	7.40E-6	0.0E0
A-PMCC-P8BC-MG	COMMON CAUSE FAILURE OF AFW PUMPS P-8B AND P-8C TO RUN	9.91E-6	0.0E0
A-PMOO-P-8A	AFW PUMP P-8A OUT OF SERVICE	1.46E-4	0.0E0
A-PMOO-P-8B	AFW TURBINE PUMP P-8B OUT OF SERVICE	1.46E-4	0.0E0
A-PMOO-P-8C	AFW PUMP P-8C OUT OF SERVICE	1.46E-4	0.0E0
A-AVCC-AFW-4-MA	ALL 4 AFW AOV'S CCAUSE FTO CV-0727/CV-073 6/CV-0736A/CV-0749	1.49E-6	0.0E0
A-AVMA-CV-0522B	AFW STEAM SUPPLY FROM SG A CV-0522BFAILS TO OPEN	8.46E-4	0.0E0
A-AVMA-CV-0727	AFW A/B TO SG B AIR OPERATED VALVE CV-0727 FAILS TO OPEN	8.46E-4	0.0E0
A-AVMA-CV-0749	AFW A/B TO SG A AIR OPERATED VALVE CV-	8.46E-4	0.0E0

Table 6.a			
Basic Event ID	Description	Original Probability	SAMA 3 Probability
	0749 FAILS TO OPEN		
A-AVMA-CV-2010	CV-2010 FAIL TO OPEN	8.46E-4	0.0E0
A-AVMB-CV-0727	AFW A/B TO SG B CV-0727 FTC	2.72E-4	0.0E0
A-AVMB-CV-0736A	AFW C TO SG B CV-0736A FTC	4.42E-3	0.0E0
A-AVMB-CV-0737A	AFW C TO SG A CV-0737A FTC	4.42E-3	0.0E0
A-AVMB-CV-0749	AFW A/B TO SG A CV-0749 FTC	2.72E-3	0.0E0
A-AVOA-AFWFLADJ	OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FAILURE OF ONE HDR	1.49E-3	0.0E0
A-AVOA-AFWSTEAM	OPERATOR FAILS TO OPEN CV-2010 FOR T-939 MAKEUP TO CST	2.59E-3	0.0E0
A-AVOA-CV-2010	OPERATOR FAILS TO OPEN CV-2010 FOR T-939 MAKEUP TO CST	2.59E-3	0.0E0
A-AVOA-CV-2010-CDTNL-HEP	CONDITIONAL HEP: A-AVOA-CV-2010 / L-ZZOA-SDC-INIT	1.43E-1	0.0E0
A-AVOA-MISCALADJ	OPERATOR FAILS TO ADJUST AFW FLOW GIVEN FLOW INSTRUMENT MISC	1.45E-3	0.0E0
A-AVOA-THROTTLE-FCV	OPERATOR FAILS TO THROTTLE AFW FCV'S GIVEN LOSS OF PNUEMATICS	1.50E-3	0.0E0
A-C2MB-152-104	AFW PUMP P-8A CIRCUIT BREAKER 152-104 FAILS TO CLOSE	1.61E-3	0.0E0
A-CBMC-52-9631	BREAKER 52-9631 FTRC	2.23E-4	0.0E0
A-CBMC-52-9749	BREAKER 52-9749 FTRC	2.23E-4	0.0E0
A-CEPO-POC-0522B	AIR TO AFW STEAM SUPPLY CV-0522B CONTROLLER POC-0522B FAILS	2.82E-3	0.0E0
A-CVMA-CK-DMW400	CK-DMW400 FTO	2.69E-4	0.0E0
A-CVMA-CK-FW743	AFW PUMP P-8B DISCHARGE CHECK VALVE CK-FW0743 FAILS TO OPEN	1.84E-4	0.0E0
A-CVMA-CK-MS402	AFW STEAM SUPPLY FROM SG A CHECK VALVE CK-MS402 FTO	1.84E-4	0.0E0
A-CVMA-CKDMW1036	CK-DMW1036 FTO	2.69E-4	0.0E0
A-CVMA-CKDMW1802	CK-DMW1802 FTO	2.69E-4	0.0E0
A-FLMK-F-P936	P-936 SUCTION STRAINER PLUGS	1.76E-3	0.0E0
A-HCMT-HIC-0727	AFW A/B TO SG B HAND INDICATING CONTROLLER HIC-0727 FAILURE	2.82E-3	0.0E0
A-HCMT-HIC-0749	AFW A/B TO SG A HAND INDICATING CONTROLLER HIC-0749 FAILURE	2.82E-3	0.0E0
A-HSMC-HS-8950A	HS-8950A CP#1 FTRC	3.49E-4	0.0E0
A-HSMC-HS-8950B	HS-8950B CP#1 FTRC	3.49E-4	0.0E0
A-IPMT-IP-0727	AFW A/B TO SG B CURRENT-PNEUMATIC CONTROLLER I/P-0727 FAILS	8.15E-4	0.0E0
A-IPMT-IP-0749	AFW A/B TO SG A CURRENT-PNEUMATIC CONTROLLER I/P-0749 FAILS	8.15E-4	0.0E0
A-ISOH-AFW-HDR3	MISCALIBRATION OF ALL FLOW INSTRUMENTS ON ALL HEADERS	1.30E-4	0.0E0
A-ISOH-AFW-HDRAB	MISCAL OF ALL AFW FLOW INSTRUMENTS ON TRAIN A/B HEADER	1.30E-4	0.0E0
A-KVMB-SV-2010	CST MAKEUP CV-2010 SOLENOID SV-2010 FTE	3.93E-3	0.0E0
A-LMMB-LMS-2010	CV-2010 LIMIT SWITCH FAILS TO CLOSE	6.71E-5	0.0E0
A-OLMK-49-9631	1/3 THERMAL OVERLOADS CIRCUITS FOR 49-9631 FTRC	3.31E-4	0.0E0
A-OOOT-CSTMK-CDTNL-HEP-1	CONDITIONAL HEP: A-OOT-CSTMKUP / A-AVOA-CV-2010	4.99E-1	0.0E0
A-OOOT-CSTMK-CDTNL-HEP-2	CONDITIONAL HEP: A-OOOT-CSTMKUP / L-ZZOA-SDC-INIT	1.43E1	0.0E0
A-OOOT-CSTMKUP	OPERATOR FAILS TO MAKEUP TO CST	2.66E-3	0.0E0

Table 6.a			
Basic Event ID	Description	Original Probability	SAMA 3 Probability
A-PBMC-PB-P936	LOCAL STOP PUSHBUTTON FTRC	3.49E-4	0.0E0
A-PCMT-PC-0522B	PRESSURE CONTROLLER PC-0522B FAILS TO FUNCTION	3.76E-3	0.0E0
A-PMME-P-936	P-936 FAILS TO START	3.29E-3	0.0E0
A-PMMG-P-936	P-936 FAILS TO RUN	8.20E-4	0.0E0
A-PMOE-AFW-PPMAN	OPERATOR FAILS TO MANUALLY START AN AFW PUMP	3.38E-2	0.0E0
A-REMA-SSX-3P8AB	AFW A/B INJECTION VALVES OPEN RELAY SSX-3/P8A/B FTD	2.41E-4	0.0E0
A-REMB-42-9631	RELAY 42-9631 FAILS TO ENERGIZE	2.41E-4	0.0E0
A-REMB-LS-8946X	RELAY LS-8946X FAIL TO ENERGIZE	2.41E-4	0.0E0
A-REMD-SSX-3P8AB	AFW A/B INJECTION VALVES OPEN RELAY SSX-3/P8A/B FTRD	2.40E-5	0.0E0
A-RVMC-RV-0783	AFW PUMP A/B DISCHARGE RELIEF VALVE RV-0783 FAILS TO REMAIN	3.82E-5	0.0E0

For SAMA 4, the changes made to the model to simulate the installation of an additional HPSI pump were also performed by manipulating the failure probabilities of existing basic events (refer to Table 6.b below). Given that the previously existing third HPSI pump was supported by the "A" power division, the "A" HPSI pump events were used to capture the shared power dependence.

The independent failures of the "A" train were squared to account for the presence of the additional pump and the fact that only 1 pump train of three is required for success, as shown in the table below. The common cause failure term was decreased by an order of magnitude to account for the change from a group of 2 to a group of 3. These changes were considered to capture the major contributors to the failure of the additional HPSI pump.

Table 6.b			
Basic Event ID	Description	Original Probability	SAMA 4 Probability
H-C2CC-HPSIPP-MB	BOTH HPSI PUMP BKRS 152-113 & 152-207 COMMON CAUSE FTC	1.13E-4	1.13E-5
H-C2MB-152-207	AC CIRCUIT BREAKER 152-207 (2400V) FAILS TO CLOSED	1.61E-3	2.60E-6
H-C2MC-152-207	AC CIRCUIT BREAKER 152-207 (2400V) FAILS TO REMAIN CLOSED	1.98E-5	3.95E-10
H-CSMD-152-207CS	CONTROL SWITCH 152-207/CS FAILS TO REMAIN OPEN	1.06E-3	1.12E-6
H-CVMA-CK-ES3186	CHECK VALVE CK-ES3186 FAILS TO OPEN	2.53E-4	6.40E-8
H-CVMA-CK-ES3340	CHECK VALVE CK-ES3340 FAILS TO OPEN	2.53E-4	6.40E-8
H-CVMD-CK-ES3183	CHECK VALVE CK-ES3183 FAILS TO REMAIN OPEN	2.48E-7	6.15E-14
H-PMME-P-66A	HPSI PUMP P-66A FAILS TO START	2.03E-3	4.00E-6
H-PMMG-P-66A	HPSI PUMP P-66A FAILS TO RUN	8.13E-4	6.61E-7
H-PMOO-P-66A	HPSI PUMP P-66A OUT OF SERVICE FOR MAINTENANCE	2.63E-3	6.90E-6
H-RVMC-RV-3267	RELIEF VALVE RV-3267 FAILS TO REMAIN CLOSED	1.16E-4	1.35E-8

RAI 7.a

Provide the following regarding lower cost alternatives to some of the SAMAs considered:

a. Phase I SAMA 1 (Additional Diesel Generator) is estimated in Table E.5-3 to cost more than $20M. This is presumably a safety grade installation with permanent connections to the E-buses. Address the viability and costs of providing a non-safety grade installation with more expedient connections as an alternative. For example, the Palisades site has a co-located (nearby) gas turbine generating facility. Address the viability and costs of providing non-safety grade backup power from this facility.

NMC Response to NRC RAI 7.a

Palisades has previously informally looked into the possibility of an agreement with the gas facility to reduce plant risk from SBO events. However, it was determined unlikely that the gas facility could support Palisades under SBO conditions. The facility is not a continuously operating plant. It is operated as a 'peaker' unit and is online only when there is a demand for power. For example, the gas plant is on-line when the Midwest Independent System Operator (MISO) dispatches the facility due to their need somewhere in the MISO footprint or when the station has contractual obligations to serve a customer. Since the plant is new, operational data is very limited.

Based on prior informal discussions, the facility does not have a black start capability. The facility is reliant on station power from its connection to the transmission system in the Palisades Switchyard for start up. Under loss of offsite power (LOOP) conditions, startup power would become unavailable to the gas plant at the same time power becomes unavailable to Palisades.

The ability of the gas plant to support Palisades would be restricted to the occasions in which the facility were operating, and would require that:

a. the facility were able to withstand the event that caused Palisades to lose offsite power and keep the unit on line and,
b. the Palisades switchyard connections (breakers) to the various transmission lines were open (except the connection to the gas facility) to ensure that Palisades would be the only load until the event was over or an onsite ac power source was recovered.

RAI 7.b

Provide the following regarding lower cost alternatives to some of the SAMAs considered:

b. Phase I SAMA 2 (portable generator for DC support) is screened out on the basis that it is less desirable and less cost-effective than the procedural changes considered in Phase I SAMA 10, which was retained. However, the evaluation of Phase I SAMA 3 (direct-drive diesel injection pump), which was also retained, indicates that a portable generator should be included for long-term SBO with the direct-drive diesel injection pump. Discuss whether a single portable generator could perform the functions required for both SAMA 2 and SAMA 3, in which case the benefits would be about \$2.5M (\$1.7M for SAMA 2/10 + \$0.8M for SAMA 3) for a cost of less than \$1.4M (\$0.3M for SAMA 2 + \$1.1M for SAMA 3).

NMC Response to NRC RAI 7.b

SAMAs 2, 3 and 10 all include alternatives that would reduce the importance of SBO events. The principle scenario driving the SBO results are the sequences in which the turbine-driven AFW pump is operating and supplying at least one steam generator. Water is available in the Condensate Storage Tank, and the diesel driven fire pumps are available as an alternate source. At four hours the station batteries are depleted and the analysis assumes that continued heat removal fails because it cannot be assured that continued flow to the steam generator(s) will not result in overfill of a steam generator and failure of the turbine driver. SAMA 2 was intended to address this scenario by providing an alternate power source to allow sufficient instrumentation to provide the operators information regarding the steam generator level and AFW flow, and assurance that the generators would not be overfilled. SAMA 3 was intended to address the broader class of sequences in which all secondary cooling is failed or degraded by partial failures and other methods of heat removal are failed (e.g., once-through-cooling, shutdown cooling, etc.). It was subsequently determined that the addition of the diesel generator discussed in SAMA 2 would allow SAMA 3 to also address the SBO scenario. SAMA 10 was included as an alternative to SAMA 3. SAMA 10 would provide procedural guidance on the means of controlling AFW flow after battery depletion that assures that steam generator overfill would not occur and that the turbine-driven pump could continue to operate for the remaining mission time. It is possible that the benefits of SAMA 3 could be reduced by combining the pump and generator on a single diesel unit. However, the current benefit of having a separate alternate pumping capability and separate electrical source would be reduced by the probability of the diesel engine failure which would fail both pump and electrical source. For SBO events this would require retaining the assumption that the turbine-driven pump fails at battery depletion. Alternately, having a separate pump and electrical source allows the diesel-driven pump to be redundant to the turbine-driven pump. The diesel generator could provide instrumentation to allow either pump to be operated after battery depletion. Failure of the diesel generator would fail both the turbine-driven and diesel-driven pumps. Therefore SAMA 3 is considered to a better alternative to address SBO and other loss of heat removal scenarios. SAMA 10 is the most cost effective alternative of reducing the risk associated with the unavailability of continued heat removal in SBO events after battery depletion.

RAI 7.c

Provide the following regarding lower cost alternatives to some of the SAMAs considered:

c. Phase I SAMA 15 is screened from further consideration based on the "potential leakage paths for contaminated sump water back to the SIRWT," and hence increased dose to the control room requiring modifications to the control room HVAC. Clarify the operation during the re-circ phase as the words "potential" and "leakage paths" imply the by-pass pathway may or may not be present for this accident (i.e., implementation requires a failure check valve in by-pass lines). Also, address the possibility of "locking" open one or more valves in the return lines to provide the same flow that would be provided by the by-pass lines, avoiding the excess flow. Provide a cost estimate for the following alternatives to this SAMA: (1) adding by-pass lines with no modifications to the control room HVAC, and (2) locking-open return line valve(s) with no modification to the control room HVAC.

NMC Response to NRC RAI 7.c

The recirculation line from the ESF pumps to the SIRW tank contains two valves (in series) that perform two distinct and competing functions. During the injection phase these valves are normally open and are required to stay open during events in which the initial PCS pressure is above the shut-off head of the HPSI pumps. Failure of either valve to remain open during this period will result in damage to any operating pump. There are two events, one for each valve, for failure to remain open that have a RRW of 1.04. Slightly lower in the importance listing is the common cause term for failure of both valves to close when a recirculation actuation signal (RAS) is generated with a RRW of 1.02. When the SIRW tank level falls to the low level set point, a RAS is generated to transfer the HPSI pump suction from the SIRW tank to the containment sump. At this point the recirculation valves must close to avoid pump runout condition as the present available NPSH at the time of RAS is very limited, and to prevent dose problems for the control room due to the introduction of contaminated water into the SIRW tank.

The addition of a bypass line around the two recirculation valves would improve the probability that a recirculation path would be open during the injection phase. However, the bypass line would have to meet the same criteria to isolate flow during the recirculation phase. The plant has had problems in the past meeting the leakage limits and has gone to great lengths to assure that the valves close and that leakage is minimized during recirculation. So, as stated, the bypass line would introduce an increased probability of failure of the isolation function. Failure of the bypass line to provide isolation would be additive to the failure of the existing valves. Since the valves are required to close, locking one open is not an option. As indicated, improving the probability to perform one of the valve functions will introduce failures that reduce the probability of performing the other function.

Palisades is currently in a study phase with respect to regulatory issues GL-2003-01 and GSI-191 where the performance of these valves is being analyzed. Since any required actions in response to these issues would override any changes considered solely for SAMA, a cost estimate for SAMA considerations is not provided.

RAI 7.d

Provide the following regarding lower cost alternatives to some of the SAMAs considered:

d. Phase I SAMA 18 is screened from further consideration due to the cost of a dedicated pump and line for EDG cooling. The description implies that the FPS as a backup would not function if the SW cooling line fails. Explain why an additional line or a temporary connection could not be installed directly from the FPS (by-passing the SW lines) as a lower cost alternative. If feasible, assess the impact on the SAMA identification and evaluation process.

NMC Response to NRC RAI 7.d

The installation of a new dedicated cooling loop to serve as the primary source of EDG cooling, with the existing Service Water cooling supply and associated FPS backup re-configured to serve as a redundant backup flow path has been evaluated. This assessment includes an engineering study to determine the source of water for the new cooling loop, the space allocation within existing structures, safety classification of the SSC's, and the need for main control room (MCR) modifications (although the cost of remote manual operation from the MCR is not included); or consideration that an operating EDG would be shut off and require restart. The engineering, design, and implementation efforts include installation of a new pump, piping, pipe supports, equipment foundations, valves, instrumentation, power feeds, motor control centers and electrical raceways. Table 7.d provides a summary of this cost estimate.

Table 7.d: Order-of-Magnitude Cost Estimate

Phase	Item	Resource	Functional Area	Dollars (2005)
Study / Analyses	1	Labor	Engineering Design	$100,000
	2	Support	Engineering Programs	$50,000
			Subtotal	**$150,000**
Design	3	Contract Labor	Engineering Design – Mech	$500,000
	4	Contract Labor	(Engineering Design – Mech MCR)	0
	5	Contract Labor	Engineering Design – I&C	$250,000
	6	Contract Labor	Engineering Design – Elec	$300,000
	7	Contract Labor	(Engineering Design – Elec MCR)	0
	8	Contract Labor	Engineering Design – Structural	$350,000
	9	Labor	Engineering Programs	$100,000
			Subtotal	**$1,500,000**
Implement	10	Labor	Maint. & Constr. Services	$1,800,000
	11	Labor	(Maint. & Constr. Services – MCR)	0
	12	Materials and Trans	Materials & Material Management	$800,000
	13	Contract Labor	Engineering Design (Constr. Support)	$200,000
	14	Support	Engineering Programs	$50,000

Table 7.d: Order-of-Magnitude Cost Estimate				
Phase	**Item**	**Resource**	**Functional Area**	**Dollars (2005)**
Life Cycle	15	Labor & Materials	Periodic inspection and repair for 20 years	$300,000
Grand Total				$4,800,000

RAI 7.e

Provide the following regarding lower cost alternatives to some of the SAMAs considered:

e. Several low cost alternatives to major enhancements have been identified as potentially cost-beneficial in previous and current license renewal applications and might be applicable to Palisades. For the following SAMAs, provide a brief statement regarding the applicability/feasibility of the alternative for Palisades, and a further evaluation of the impact on the SAMA identification and evaluation process if the alternative could be potentially cost-beneficial at Palisades:

i. Modify procedures to conserve or prolong the inventory in the SIRWT during SGTR events (Ft. Calhoun, SAMA 92)
ii. Add accumulators or modify procedures on SIRWT bubblers and recirculation valves to avert/recover from premature recirculation actuation signal (Ft. Calhoun, SAMA 181)
iii. Provide portable power supply as backup to open PORVs during/following core damage (Ft. Calhoun, SAMA 183)
iv. Add capability to flash the field on the EDG to enhance SBO recovery (Ft. Calhoun, SAMA 184)
v. Modify procedures and/or make hardware changes to provide alternate capability to increase heat removal from the RCS and accelerate RCS cool down (Ft. Calhoun, SAMA 186)
vi. Modify procedures and enhance training to reduce human error associated with recovery following SBO (ANO-2, SAMA AC/DC-16)
vii. Modify procedures to shed CCW loads on loss of essential raw cooling water to extend component cooling water heat-up time (ANO-2, SAMA CW-06)
viii. Install backwash filters in place of existing service water pump discharge strainers to reduce probability of common cause failures (ANO-2, SAMA CW-27)
ix. Replace a containment sump valve(s) with air-operated valve(s) to reduce common cause failures (ANO-2, SAMA CC-20)

NMC Response to NRC RAI 7.e – i

Per Omaha Public Power District (OPPD) APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE, the following information was provided regarding SAMA 92:

> *Description:*
> *Modify procedures to conserve or prolong the inventory in the Borated Water Storage Tank (Safety Injection Refueling Water Storage Tank, or SIRWT) during SGTRs. At FCS this SAMA would be implemented by providing procedures to*

refill the SIRWT with borated water and ensuring that the necessary boration and water sources are available.

SAMA Benefits:
An increased supply of borated water would reduce the potential for a SGTR to result in core damage. Revision 3 of the FCS probabilistic risk assessment (PRA) model conservatively assumes that once the initial SIRWT inventory is depleted the event will progress to core damage.

Evaluation:
The evaluation assumed procedures and additional sources of borated water would eliminate failures associated with depletion of the SIRWT inventory during ISLOCAs and SGTRs.

Palisades ISLOCA Events

As directed in plant EOPs, the operator would initiate actions to makeup to the SIRWT. The operator is directed to either:

- SOP-2A, "Chemical & Volume Control System Charging & Letdown", or
- SOP-17A, "Clean Radioactive Waste System"

Palisades SGTR Events

Neither EOP-5.0 nor CEN-152 discuss makeup to the SIRW tank or discuss recirculation actuation signal (RAS). Nevertheless, given that the SIRW level would be lowering and given indications that primary coolant is leaking from somewhere outside containment, the operators would make the determination that the in-use EOP-5.0 SGTR Recovery procedure is not adequate. They would transition to EOP-9.0 Functional Recovery Procedure. IC-2 Step 14 covers SIRW tank lowering without a corresponding rise in containment sump level. The operator is again referred to either,

- SOP-2A, "Chemical & Volume Control System Charging & Letdown", or
- SOP-17A, "Clean Radioactive Waste System"

regarding makeup to the SIRW tank.

If makeup to the SIRW tank were not provided, the low level RAS switchover setpoint would be reached and makeup to the PCS would continue. The Palisades PSA model credits the RAS function for selected sequences during an SGTR event, unlike Ft. Calhoun.

In summary, the Palisades EOPs explicitly cite SIRWT makeup in the LOCA recovery procedure and indirectly address SIRWT makeup in the Functional Recovery Procedure given an SGTR event.

NMC Response to NRC RAI 7.e – ii

Per Omaha Public Power District (OPPD) APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE, the following information was provided regarding SAMA 181:

> *Description*
> *This SAMA would involve adding the capability to prevent an early Recirculation Actuation Signal (RAS) following the loss of instrument air. Depletion of the SIRWT bubblers will result in a low-level indication in the SIRWT and cause a premature RAS. This may cause the Emergency Core Cooling System (ECCS) and spray pumps to take suction from a sump with inadequate net positive suction head (NPSH). Pump damage and failure are possible. The options considered by this SAMA are: (1) procurement and installation of additional accumulators to extend the instrument measurement time; (2) replacement of the existing accumulators with larger ones; or (3) implementation of procedural guidance (and the associated engineering analyses and training) to support operator actions to avert and/or recover from the premature RAS.*
>
> *SAMA Benefits*
> *This SAMA would significantly reduce the potential for a premature RAS resulting from the depletion of the SIRWT level indication air bubblers. Currently the bubblers will last 13 hours. Several events, such as SGTRs and smaller LOCAs, may require extended feeding from the SIRWT. Extending the capability of the bubblers and/or increasing the guidance documents (EOPs /AOPs) to alert the operator to the potential inadvertent RAS will reduce the potential for or mitigate the consequences of premature RAS.*

At Palisades, the method of detecting low water level in the SIRW tank for RAS initiation is by conductivity probes hung in the tank; therefore the details of the above identified failures do not apply.

NMC Response to NRC RAI 7.e – iii

Per Omaha Public Power District (OPPD) APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE, the following information was provided regarding SAMA 183:

> *Description:*
> *This SAMA would provide a portable power source, inverter, associated implementing cables, and necessary operating and implementation instructions for use as a backup power supply for opening the power-operated relief valve(s)*

[PORV(s)]. Guidance for use of this backup supply will be provided in the FCS SAMG."

SAMA Benefits:
This SAMA is primarily directed at mitigating severe accident releases following a core damage event with RCS release paths (or potential release paths) to the environment. These events include ISLOCAs and some SGTRs. Opening a PORV during a core damage event would reduce the potential for a TI-SGTR, lower RCS pressure while potentially averting a high-pressure melt ejection challenge to the Containment, and retain RCS fission products within the Containment.

At Palisades, the PORVs have dedicated 1E power. Only one valve is required for once through cooling (OTC). If the valves don't have power, neither do the HPSI pumps. There is no benefit unless a portable power supply is provided for the HPSI and CSS pumps as well. The CSS pumps are required for the HPSI piggyback configuration so that adequate pump NPSH can be maintained.

Palisades does include opening a PORV as a Level 2 mitigation strategy addressing high temperature creep rupture thermally induced failures of the steam generator tubes. Thermally induced failure of steam generator tubes did not meet the criteria for consideration of SAMA for Palisades.

NMC Response to NRC RAI 7.e – iv

Per Omaha Public Power District (OPPD) APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE, the following information was provided regarding SAMA 184:

Description
This SAMA is intended to increase the capability of FCS to cope with an SBO event when one or more emergency diesel generator (EDG) fails to start or an EDG failure occurs and restart is required after battery depletion. This SAMA would require hardware modification and operational changes. The hardware modification includes the addition of a power supply to flash the field. Operational changes include the development of procedures for restoring the affected EDGs to operability and the associated operator training.

SAMA Benefits
This SAMA enhances EDG recovery for SBO accident sequences involving the unavailability of one or more EDG following a loss of offsite power event. This SAMA will enhance safety by reducing the probability of core damage due to certain SBO events.

This SAMA was not originally evaluated. NMC will investigate the benefits of developing a procedure to allow alternate battery supplies to provide field flashing to allow a restart of a recovered EDG. If warranted, the procedure would be added to the present Palisades Severe Accident Management Guidelines.

NMC Response to NRC RAI 7.e – v

Per Omaha Public Power District (OPPD) APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE, the following information was provided regarding SAMA 186:

> *Description*
> *This SAMA involves performing specific procedural and/or hardware changes to give the plant the alternate capability to increase heat removal from the RCS and accelerate RCS cooldown. Introducing an alternate cooldown pathway will increase the capability of the plant to cope with ISLOCAs, SGTRs, and long-term SBOs.*
>
> *This modification is designed to facilitate reducing RCS temperature and pressure to mitigate ISLOCAs and RCS SGTRs. ISLOCAs are often complicated by equipment failures due to flooding in the AB, which preclude normal cool down methods such as HCV-1040 or steam dump and bypass. This modification may involve nitrogen backup to open the Main Steam (MS) valves, MS-291 and -292 (and leave them open) while continuing to feed both steam generators. This would also facilitate rapid RCS temperature reduction to preclude RCP seal LOCAs during prolonged SBO.*
>
> *SAMA Benefits*
> *These changes would both avert core damage and reduce potentially high releases of radioactivity by extending the time until core uncovery following an SBO-induced RCP seal LOCA. Efficient depressurization of the RCS to below 200 pounds per square inch atmospheric (psia) may terminate the small ISLOCA. RCS heatups that result from SGTRs may also be cooled down more quickly, allowing the potential for reaching safe shutdown cooling (SDC).*

The Palisades identified SAMA 23 addresses accelerated PCS cooldown to alleviate the stress on the PCP seals.

NMC Response to NRC RAI 7.e – vi

Per Arkansas Nuclear One – Unit 2 APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE – Attachment E, the following information was provided regarding SAMA ANO-2, SAMA AC/DC-16:

> *Potential enhancement*
> *Emphasize steps in plant recovery following a station blackout event.*
>
> *Discussion*
> *Reduce human error associated with recovery of station blackout events through enhanced training and procedural guidance.*

The Palisade's SAMAs 2, 3, and 10 address different means of reducing the impact from a SBO event. SAMA 10 targets the human error contribution during a SBO event by providing detailed guidance regarding proper operation of the turbine-driven AFW pump after the station batteries have depleted. The PSA model currently assumes that the human error for controlling AFW flow after loss of instrumentation due to station battery depletion would result in failure of the turbine-driven AFW pump. The failure would result from overfilling the steam generator and the resulting low quality steam subsequently failing the turbine-driver. If SAMA 10 is implemented, training would be required to provide guidance on the proper means of controlling AFW flow just prior to and following battery depletion in a way that assures that the turbine-driven pump can be operated successfully without instrumentation and preclude overfill of the steam generator. Therefore it is concluded that Palisades has addressed the intent of the ANO identified SAMA, given that the Palisades SAMA10 will be evaluated.

NMC Response to NRC RAI 7.e – vii

Per Arkansas Nuclear One – Unit 2 APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE – Attachment E, the following information was provided regarding SAMA ANO-2, SAMA CW-06:

> *Potential enhancement*
> *On loss of essential raw cooling water, proceduralize shedding component cooling water loads to extend the component cooling water heatup time.*
>
> *Discussion*
> *Increase time before reactor coolant pump seal failure during loss of service water sequences.*

Palisades has procedures in place that provide guidance on the isolation of components to reduce the heat loads on the system. ONP 6.2. "Loss of Component Cooling" includes guidance on isolation and restoration of loads given a loss of component cooling water flow or inventory. ONP 6.1, "Loss of Service Water" provide guidance on the isolation and restoration of service water loads and directs the user to ONP 6.2 given a loss of service water flow or inventory. Therefore it is concluded that Palisades already addresses the intent of ANO SAMA CW-06.

NMC Response to NRC RAI 7.e – viii

Per Arkansas Nuclear One – Unit 2 APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE – Attachment E, the following information was provided regarding SAMA ANO-2, SAMA CW-27:

> *Potential enhancement*
> *Replace current service water pump discharge strainers with backwash filters.*
>
> *Discussion*
> *Reduce the failure frequency of the service water system. This SAMA would install backwash filters in place of the existing strainers, reducing the probability of a common cause failure.*

The Palisades PSA model includes the service water pumps and their discharge basket strainers. The service water system is not a significant contributor to core damage. The only events that met the criteria for consideration of the Palisades SAMA were the out-of-service event for one of the pumps and the common cause failure of the traveling screens. The failure of the basket strainers did not meet the criteria for SAMA consideration, therefore ANO SAMA CW-06 is not considered to warrant any further review.

NMC Response to NRC RAI 7.e – ix

Per Arkansas Nuclear One – Unit 2 APPENDIX E APPLICANT'S ENVIRONMENTAL REPORT OPERATING LICENSE RENEWAL STAGE – Attachment E, the following information was provided regarding SAMA ANO-2, SAMA CC-20:

> *Potential enhancement*
> *Make containment sump recirculation outlet valve motor-operated valves 2CV-5649-1 and 2CV-5650-2 diverse from one another.*
>
> *Discussion*
> *Replace either containment sump valve 2CV-5649-1 or 2CV-5650-2 with an air operated valve. This would reduce the potential for common cause failure of these valves.*

Palisades reviewed the cost of replacing the actuator on one of the existing 24-inch AOVs with a motor operator, including Class 1E circuit and power modifications, control modifications, simulator modifications, plant procedure changes and training. This review assumed the present containment sump discharge valve controls as designed to provide proper response to automatic control signals, as well as manual commands from both the main control room (MCR) and the redundant engineered safeguards control panel. However, significant study effort is required for the power supply conceptual design, since no initial design margin exists with respect to emergency diesel generator (EDG) loading after a design basis accident. Measures, such as increasing the EDG capacity is not included in the cost estimate presented below.

It is considered that the MCR modifications involve re-labeling of existing switches and indication; most control wiring is reused. Table 7.e-ix provides a summary of this cost estimate.

Table 7.e–ix: Order-of-Magnitude Cost Estimate

Phase	Item	Resource	Functional Area	Dollars (2005)
Study / Analyses	1	Contract Labor	Engineering Design / Analyses / Studies	$40,000
	2	Support	Engineering Programs	$20,000
			Subtotal	**60,000**
Design	3	Contract Labor	Engr Design – Mech	$20,000
	4	Contract Labor	Engr Design – Elec	$60,000
	5	Contract Labor	Engr Design – I&C	$30,000
	6	Contract Labor	Engr Design – Structural	$10,000
			Subtotal	**$120,000**
Implement	7	Labor	Maintenance & Construction Services	$150,000
	8	Contract Labor	Construction Contractor	$50,000
	9	Materials and Trans	Materials & Material Management	$50,000
	10	Contract Labor	Engr Design (Const Support)	$10,000
	11	Support	Engineering Programs & Simulator	$20,000
Life Cycle	12	Labor & Materials	Maintenance for 20 years.	$40,000
Grand Total				$500,000