NMC
Committed to Nuclear Excellence

Palisades Nuclear Plant
Operated by Nuclear Management Company, LLC

May 26, 2005

10 CFR 50.90

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

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

Supplement to License Amendment Request: One-Time Extension to Technical Specification Action Completion Time for Restoration of a Service Water Train to Operable Status

By letter dated April 1, 2005, Nuclear Management Company, LLC (NMC) requested Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment for the Palisades Nuclear Plant. After subsequent discussions with the NRC staff, it was determined that a supplement is necessary. Enclosure 1 contains the supplement.

A copy of this supplement has been provided to the designated representative of the State of Michigan.

Summary of Commitments

This letter contains one new commitment and no revisions to existing commitments.

> NMC will conduct fire tours hourly during the extended Technical Specification action completion time. Fire tours will be conducted in the following areas: Cable Spreading Room, 1D Switchgear Room, Turbine Building, 1C Switchgear Room, Auxiliary Building Corridor, Component Cooling Water Room, and Screen House.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 26, 2005.

W. [signature] FOR DAN J MALONE

Daniel J. Malone
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

A001

Enclosure (1)
Attachments (2)

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

ENCLOSURE 1
SUPPLEMENT TO LICENSE AMENDMENT REQUEST: ONE-TIME EXTENSION TO TECHNICAL SPECIFICATION ACTION COMPLETION TIME FOR RESTORATION OF A SERVICE WATER TRAIN TO OPERABLE STATUS

NRC Request

1. Service water pump P-7C trend history?

NMC Response

1. Service water pump P-7C differential pressure (dP) began a degrading trend shortly after rebuild in December 2001. Post maintenance test results provided a dP of approximately 99 psid at inservice test conditions. Degradation has been gradual and steady to the most recent test conducted April 14, 2005. The April test indicates P-7C is operating at approximately 96 psid under inservice test conditions. The inservice testing procedure for P-7C states that the following dP ranges are acceptable: $93.6 \leq \Delta P \leq 108.4$ (P-7C). The lower acceptance limit is 93.6 psid. When inservice test dP falls below this limit, P-7C is considered inoperable.

   It is anticipated that P-7C dP will degrade during the summer months. This expectation is based on Palisades' history and is due to the necessity to operate all three service water pumps on a continuous basis for most of the summertime period. The magnitude of this degradation is expected to be between 1 and 2 psid at inservice test conditions. Therefore, at the conclusion of the summer months, P-7C is expected to be operating between 94 and 95 psid at inservice test conditions. At these conditions normal test variations may cause the pump to become inoperable. Therefore, P-7C is currently scheduled for rebuild beginning the week of September 25, 2005.

NRC Request

2. Can the dP acceptance criteria be revised?

NMC Response

2. Acceptable, alert, and required action ranges for dP have been established in accordance with ASME OMa-1988 Part 6, Table 3b, with exceptions. The lower degradation limits are restricted as follows: P-7A - 96.7 psid, P-7B - 94.0 psid, P-7C - 93.6 psid. When operating at inservice test conditions, which are approximately 40% of accident assumptions, these dPs are the minimum necessary to support design basis accident analysis requirements.

The reference values for pump operating parameters are determined in accordance with test Code requirements from the initial inservice test performed when the pump is known to be operating acceptably. Reference values are at points of operation readily duplicated during subsequent tests.

If the particular parameter being measured or determined can be significantly influenced by other related conditions, then these conditions are analyzed. Additional sets of reference values may be established in order to facilitate pump testing under different plant conditions or equipment operating modes. These additional reference values are determined when the pump is known to be operating acceptably and do not conflict with the original reference data.

There are no identified circumstances that would allow the establishment of new reference values or a revised lower dP acceptance limit for service water pump P-7C that would meet the requirements of the inservice test Code applicable to Palisades.

NRC Request

3. Is a permanent change possible versus a one-time change?

NMC Response

3. Due to the complex compensatory measures necessary to have in place during the extended completion time, a permanent change is not practical.

NRC Request

4. Is it possible to operate with P-7C in constant standby until the 2006 refueling outage?

NMC Response

4. Section 4.0 of the License Amendment Request describes the Palisades service water system design.

   It would be possible to "minimize" P-7C operation, however, it would not be possible to place it in "constant" standby. As Lake Michigan water temperature and air temperatures in the turbine building and containment elevate during the summer months, service water loads require more cooling flow, which increases the likelihood that a third service water pump would be started in order to satisfy the plant's cooling flow needs.

Past plant operation has shown that three service water pumps are typically run during the summer months (approximately early-June through early-September). Additionally, Technical Specification LCO 3.6.5 limits containment average air temperature to less than or equal to 140 deg F. Therefore, the ability to run a third service water pump during the summer becomes increasingly important, and therefore, it would be highly unlikely that P-7C could be placed in constant standby mode during these months.

After summer operation, operation of P-7C could be minimized, however, periodic pump operation would be required to support technical specification surveillance testing activities and maintenance activities on P-7A or P-7B. Two service water pumps are required to furnish normal cooling water demand; the third pump will normally be in standby. Maintenance activities such as SW pump basket strainer cleanings and SW pump repacks require the associated SW pump to be removed from service for maintenance and the standby pump to be started to satisfy the two pump requirement for normal plant operation.

NRC Request

5. Degradation source of P-7C? Proposed Completion time of 7 days adequate?

NMC Response

5. The source of P-7C degradation is normal wear. Service water comes from Lake Michigan and contains a varying concentration of sand. The sand causes wear of rubber bearings, wear rings and other pump surfaces. Inservice testing data indicates that P-7C dP is currently in the 96 psid range and is showing a long-term, dP decreasing trend in pump performance.

   The current schedule for pump rebuild reflects 139 hours, which includes the post-maintenance testing.

NRC Request

6. The current plant core damage frequency (CDF), large early release frequency (LERF), delta CDF and delta LERF for the proposed change.

NMC Response

6. Baseline CDF for the model = 6.2E-05
   CDF with P-7C out-of-service (OOS) for 7 days = 6.3E-05E
   CDF with P-7C OOS for 7 days & Comp Measures = 5.7E-05
   Delta CDF = 1.0E-06

Baseline LERF = 3.3E-07
LERF with P-7C OOS = 3.4E-07
LERF with P-7C OOS for 7 days & Comp Measures = 3.4E-07
Delta LERF = 1.0E-08

NRC Request

7. Comparison of the above with the risk metrics in RG 1.174.

NMC Response

7. When the calculated increase in CDF is in the range of 10-6 per reactor year to 10-5 per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10-4 per reactor year (Region II).

   Comparison with the metrics of RG 1.174 shows that the delta CDF is at the criteria of 1.0E-06, with a total CDF below 1.0E-04. Delta LERF is below the criteria of 1.0E-07, and is considered a very small change. The delta CDF was also near the criteria of 1.0E-06, which would be considered a very small change in CDF.

NRC Request

8. Discussion of the "health" of the other two SW pumps: Are either of them showing a declining trend?

NMC Response

8. The inservice testing procedure for service water pumps P-7A and P-7B states that the following dP ranges are acceptable:
   P-7A - $96.9 \leq \Delta P \leq 112.2$; and P-7B - $94.0 \leq \Delta P \leq 108.1$. P-7A dP data is presently in the 99 psid range at test conditions and is stable with no observable trend. P-7B dP data is in the 97 psid range at test conditions and is stable with no observable trend. Hydraulic and mechanical data obtained during quarterly pump testing of both P-7A and P-7B indicates no declining trend.

NRC Request

9. Additional information on RG 1.177 Tier 1:

   PRA quality details, including category A and B findings and observations (F&O's) from the industry peer review, how dispositioned, and detailed justification that remaining open items do not impact the risk assessment.

NMC Response

9. The industry peer review identified nine Level A and fifty Level B F&O's. All nine Level A F&O's have been resolved. The peer review comments and resolution of the Level A F&O's is included as Attachment 1. Of the fifty Level B F&O's, seven remain to be resolved. The remaining seven are included in the plant corrective action process. The remaining items include documentation issues, use of generic versus plant specific data for electrical components, and a comment regarding the use of maintenance rule data. The remaining seven items do not impact the ability to use the model to calculate delta CDF and LERF values based on the removal of P-7C from service. A listing of the remaining Level B F&O's is included as Attachment 2.

NRC Request

10. Description of PRA models to a greater level of detail than provided. For example, are internal floods included in the PRA model?

NMC Response

10. The model used in this analysis was the current PRA, which was a Level 3 internal event model. It consisted of detailed moderate size event trees with linked fault trees. The model did not include external event models developed for the Individual Plant Examination of External Events (IPEEE). An ongoing update to the model does include an updated flooding analysis. The flooding analysis results represent a contribution to the CDF in the low to mid E-07 range. In addition, the area the service water pump is in (screen house) is a small contributor to the flood CDF because of its location and the out flow areas from the room.

The fire update will begin this year with an expected completion in 2006. The original fire PRA results were of the same order of magnitude as the internal events PRA. The IPEEE analysis was conservative in that for many areas a fire was assumed to fail all equipment in the area.

The IPEEE Report identified five risk significant fire areas. The areas are:

1) Control Room,
2) Cable Spreading Room,
3) 1D Switchgear Room,
4) Turbine Building, and
5) 1C Switchgear Room.

NRC Request

11. Overview of the risk assessment methodology, including assumptions employed (only the ICCDP and ICLERP results are provided).

NMC Response

11. With respect to the analysis for this proposed license amendment request (LAR), the only assumption was that P-7C was unavailable for the 7-day (168 Hour) duration and the configuration of pumps in service versus standby was modified to reflect the actual condition. The methodology used to support the proposed LAR was a full requantification of the PRA model using the model of record. Quantification was truncated at 1E-09.

NRC Request

12. A more detailed estimate of fire risk or justification of why fire risk is not significant. For example: A fire that results in loss of one of the remaining SW pumps during the planned maintenance could lead to core damage for scenarios where operators fail to isolate SW to the containment and non-critical header.

NMC Response

12. As noted above, the current model does not include an updated fire model. The five high-risk fire areas were identified in response to item 10. above.

NRC Request

13. The seismic discussion should explicitly state a conclusion regarding the judged risk impact.

NMC Response

13. The principal contributors to loss of secondary heat removal and failure of once through cooling during the injection phase and recirculation phase following a seismic event are associated with makeup to the condensate storage tank (CST). The CST nominally has capacity to provide makeup of the steam generators to account for approximately 6 hours of decay heat removal. Normal on-site makeup supplies (such as from demin water storage) rely on offsite power for transfer to the CST. Alternate sources of

makeup include fire protection system (FPS) makeup to the suction of auxiliary feedwater (AFW) pumps, P-8A&B, or the CST (which can then supply suction to AFW pump P-8C). Additionally, AFW pump P-8C suction can be aligned from the service water system (SWS).

Dominant seismic contributors to these sources of makeup include the day tanks for the diesel fire pumps, FPS control cabinet (EC-137) and transformer EX-13 (power for electric fire pump P-9A). Loss of this equipment is assumed to lead to failure of the FPS leaving only P-8C with SWS as the principle long term suction source. There are no significant vulnerabilities of the SWS to a seismic event. However, the SWS feeds only the suction of the AFW pump P-8C. No credit is taken for the SWS cross-tie to the FPS to allow the SWS to supply suction water to AFW pumps P-8A&B. The seismically induced failures of the FPS effectively leaves one train of AFW (pump P-8C) for long term makeup to the steam generators. Loss of this train due to random failure leads to initiation of once through cooling with either train of HPSI pumps and at least one PORV. There are no seismic failures that significantly impact the operation of equipment required for once through cooling during the injection phase.

There are two dominant random event groups contributing to the failure of AFW pump P-8C: failures associated with the pump and power supply; and failures associated with the manual supply valves to align service water as an alternate suction source. The failure rate for these two groups is 2.36E-02/yr and 3.65E-02/yr, respectively, for a total of 6.01E-02/yr. P-8C operation becomes important for long term makeup to the steam generators should other means of supplying water to the CST or AFW pumps P-8A&B become unavailable following a seismic event.

As noted in the IPEEE report the importance of the service water system was due to the seismic induced failures of the fire protection system to provide an alternate suction source to the AFW pumps for continued long term secondary cooling. Under these conditions AFW pump P-8C becomes important because it can be aligned to the service water system for which there were no significant seismic vulnerabilities identified. Also identified in the IPEEE was the fact that that the primary contributor to the inability to SWS flow to P-8C suction was based on the probability of failure to align the manual valves that connect flow from the SWS header to the suction of P-8C. Therefore, the removal of P-7C from service for the stated period would not result in a significant change in the IPEEE results for this function and not have any significant impact on the calculated delta CDF or LERF.

NRC Request

14. Additional information on RG 1.177 Tier 2:

Discuss how "high risk configurations" during the SW pump maintenance were identified, and show how the commitments and compensatory measures relate to those scenarios.

NMC Response

14. A list of proposed compensatory measures were identified by a NMC employee responsible for implementation of the risk management process (Maintenance Rule (a)(4)) utilizing the equipment out of service (EOOS) program. The proposed actions were assessed by requantifying the PRA model with P-7C removed from service and model changes to reflect the impact of the proposed actions (if they were explicitly represented in the model). Subsequently a third quantification was performed only crediting the actions believed most likely to have a significant impact on offsetting the increased risk from P-7C removal from service. The analysis demonstrated that a subset of the actions were responsible the reduction in risk.

NRC Request

15. Depending on the more detailed consideration of fire risk, consider additional compensatory measures (e.g., roving fire watch in risk-significant fire areas).

NMC Response

15. NMC will commit to performing fire tours in the areas other than the control room (due to continuous occupation).

NRC Request

16. Additional information discussed in RG 1.174: (1) compliance with existing regulations; (2) defense-in-depth; and, (3) adequate margin.

NMC Response

16. (1) Compliance with existing regulations

With the implementation of the proposed LAR at Palisades Nuclear Plant, NMC continues to meet the applicable design criteria. The proposed LAR is a one-time extension to the Technical Specification action completion time. It does not affect the design basis of the plant. In addition, NMC will remain within the scope of the Technical Specification Limiting Conditions for Operation and is still subject to the requirements of the action statements.

(2) Defense-in-depth

The proposed change remains consistent with the defense in depth philosophy based on the following considerations.

1. The change is not a permanent change.

2. Current technical specifications allow a service water pump to be out of service.

3. System redundancy, independence and diversity are maintained commensurate with the expected frequency, consequences of challenges to the system, and technical specification allowances.

4. The proposed change is an extension of the allowed outage time and is not expected to introduce new common cause failure mechanisms or changes in human error due to changed maintenance practices or responses to system malfunctions.

5. Appropriate compensatory actions have been identified.

6. Equipment that will be maintained in service has been identified.

7. The process to ensure that risk will be maintained at acceptable levels and to ensure that redundant or diverse equipment will not be removed from service is in place as discussed in the submittal.

8. Prevention of core damage and containment failure, and consequence mitigation have been demonstrated by the results of the PRA analysis.

(3) Adequate margin

The proposed amendment does not involve a significant reduction in a margin of safety. With service water pump P-7C inoperable, 100% of the required post-accident service water system cooling capability remains available with the redundant train maintained operable. Therefore, there is no significant reduction in the margin of safety.

Based on the availability of redundant systems, the compensatory measures that will be taken, and the low probability of an accident that could not be mitigated by the available systems, the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the PRA evaluation and the planned compensatory measures, NMC has concluded that risk levels are consistent with RG 1.174 and RG 1.177 criteria, and remain manageable with sufficient margin to allow remedial and corrective actions to be implemented in the event unplanned equipment outages occur.

NRC Request

17. A more detailed estimate of the fire risk or a justification of why fire risk is not important: The proposed response merely states that the licensee does not have a fire PRA and lists the fire contributors identified in the IPEEE. This is not sufficient. A fire PRA is not needed. However, the licensee needs to look at fire initiating event frequencies (from the IPEEE) and do a conscientious look for scenarios that could be exacerbated when the SWP is out of service. Experience on several other dockets indicates that fires can become a significant risk contributor during service water pump outages.

NMC Response

17. An evaluation of the impact of fire on the availability of service water pumps was conducted in support of the proposed LAR to extend the LCO time for service water pump P-7C to allow for a rebuild of the pump. The input data used to conduct the IPEEE for Palisades was reviewed to identify the fire areas that were identified as having an impact on the service water pumps. A review of the data indicates that the fire impacts could be grouped into three categories. Category 1 represents those areas that were determined to impact all three pumps. These areas include; the main control room (fire area 1), the cable spreading room (fire area 2), the screen house (location of the service water pumps) (fire area 9) and a minor contribution from the manholes from the Bus 1C switchgear room (fire area 34). A fire in any of these areas is assumed to fail all three pumps. Under these conditions there would be no change in the delta risk from a fire. Pumps P-7A and P-7B would be failed by the fire and P-7C would either be failed by the fire or be out of service.

    Category 2 represents the fire areas that were determined to impact both pumps P-7A and P-7C. Since these pumps are on the same electrical division, their cables are routed through common areas. The fire areas in category 2 are: bus 1D switchgear room (fire area 3), auxiliary building corridor (fire area 13), and the Component Cooling Water Room (fire area

16). Under these conditions the fire results in the loss of the two pumps on the same electrical division. Again the impact of fire is that there is no change in the delta risk. P-7B remains unaffected by the fire and is subject only to its random failures. Pumps P-7A and P-7C are either both failed by the fire or the fire fails P-7A and P-7C is out of service for maintenance.

Category 3 represents the fire area that was determined to impact only pump P-7B. The one area that affects only pump P-7B is the bus 1C switchgear room (fire area 4). A fire in this area would have the most impact on the availability of the operating pumps with pump P-7C out of service since it has no impact on pump P-7C. However, the initiating event frequency for this fire area is only slightly higher than the random failure of the pump and therefore will only have a minor impact on risk related to pump failure and in particular the delta CDF.

These conditions were evaluated with the model using two methods. First, a transient event tree was quantified using the combined initiating event frequency for each category as the initiating event frequency for the event tree. Affected pumps were assumed to be failed by the fire. Quantifications were conducted with P-7C available and with P-7C out of service. Second, the fire initiating event frequencies for the categories were converted to probabilities/day of a fire. These probabilities were added to the current probability of a basic event in the model that represented the pump combinations failed by the fire. The internal events model was then quantified for each category with P-7C available and with P-7C out of service.

The result of these quantifications support the conclusions stated above. The results from the first method demonstrate that for category one and two the delta CDF is zero. Category three results were below the truncation value (1E-09) used in the analysis. In the second method there was a small increase (~1.4E-07) in the CDF for both the baseline (P-7C available) and with P-7C out of service. However, the delta CDF remained constant indicating that the delta was not impacted by the addition of the fire probabilities.

Based on the fire impact evaluation, NMC is making an additional commitment to perform fire tours as follows:

> NMC will conduct fire tours hourly during the extended Technical Specification action completion time. Fire tours will be conducted in the following areas: Cable Spreading Room, 1D Switchgear Room, Turbine Building, 1C Switchgear Room, Auxiliary Building Corridor, Component Cooling Water Room, and Screen House.

Attachment 1
Level A Findings and Observations with Resolution

AS-05 CA017243
Update PSA to Include primary coolant pump (PCP) seal model
Track closure of CEOG PSA peer review comment PEER-2000 AS-05, PCP Seal Model (PSA issue #171).

Peer Review Comment
Assumption 4.7: Reactor Coolant Pump seal loss of coolant accident (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).

Resolution
The PSA seal LOCA model has been developed and is documented in PA-03-004-01, Palisades Reactor Coolant Pump Seal LOCA Analysis, revision 0, dated 2/2/7/04. 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.

AS-09 CA017237
Track Closure of CEOG PEER-2000 AS-09 EDG repair
Track closure of CEOG PSA peer review comment PEER-2000 AS-09, Diesel Generator Repair/Recovery (PSA issue #114).

Peer Review Comment

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 diesel generators (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.

Issue

The two recovery basis events are used to recover all failures that fail the DG Top Event. This includes supports 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.

Resolution
The following engineering analyses address closure of this issue:

EA-PSA-LOSDC-03-11 r0. This EA defines the allowable timing requirements modeled in the recovery analyses.

EA-PSA-DG-REC-03-14 r0. This EA documents the four human recovery events employed in the updated Loss of Offsite Power (LOOP) analysis.

PA-02-001-01 r0. This EA documents the LOOP IE frequency, the LOOP recovery versus time and the EDG recovery versus time.

EA-PSA-LOSDC-03-13 r0. This EA documents the double bayesian analyses performed to assess the plant-to-plant variability in calculating the LOOP initiating event frequency.

EA-PSA-LOOP-EVAL-03-12 r0. This EA evaluates the overall impact of the updated LOOP information, contained in the above referenced EAs, on the baseline core damage frequency employing the PSAR1B-Modified wEQ model.

DA-07 EA-PSA-DATA-02-12

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. (The 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.

Resolution
The EA was revised and the identified corrections made.

HR-10 EA-PSA-HRA-DEP-01-33

Human Action Dependencies

H-ZZOA-OTC-INIT, Failure to initiate once-through-cooling shows up in cutsets that include:

- *A-AVOA-AFWFLADJ: Failure to increase auxiliary feedwater flow when feeding one steam generator*
- *A-AVOA-CV-2010: Failure to align and provide make-up to the condensate storage tank*
- *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.

Resolution
Contractor TENERA (now Applied Reliability Engineering - AREI) prepared and completed a series of HRA calculations addressing the HEP used in the Palisades PSA model. The analysis has received a owners review and approval under EA-PSA-2001-015 Revision 0. The HRA process including the initiators is described in the vendor document. The dependencies are addressed by EA-PSA-HRA-DEP-01-33 (CALC 01-33).

HR-13 CA017242

CEOG PSA peer review comment PEER-2000 HR-13
Track closure of CEOG PSA peer review comment PEER-2000 HR-13 Instrumentation Supporting Human Actions (PSA issue #166).

Peer Review comment
Indication/equipment electrical support dependencies were not implemented into each human action.

Resolution
Calculation PA-02-002-1, "Incorporation of Control Room Instrumentation Failures into the Palisades PSAR1B-Modified-wEQ Fault Tree and assessing their Impact on Core Damage Frequency (CDF)," has been completed.

QU-25 CA017245
Track Closure of CEOG PEER-2000 QU-05
Track closure of CEOG PSA peer review comment PEER-2000 QU-05, Uncertainty Guidance Document (PSA issue #200).

Peer Review comment
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).

Resolution
PA-03-004-1, "Parameteric Uncertainty Analysis for the Palisades Level 1 Internal Events" was completed.

QU-08CA017246
Track closure of CEOG PEER-2000 QU-08 Track closure of CEOG PSA peer review comment PEER-2000 QU-08, Uncertainty Analysis (PSA issue #203).

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.

Resolution
PA-03-004-1, "Parameteric Uncertainty Analysis for the Palisades Level 1 Internal Events" was completed.

MU-04 CA017241
Implement a PSA Documentation & Control Process
Track closure of CEOG PSA peer review comment PEER-2000 MU-01 Documentation & Control Process (PSA issue #152).

Peer Review Comment

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.

Resolution
Revision 1 of RIE Guideline 01, dated 11/14/03, was issued with the resolution of peer review comments included.

MU-07 CA017240
Implement PSA Software Control Process Track closure of CEOG PSA peer review comment PEER-2000 MU-03, Software Control (PSA issue #147).

Peer Review Comment

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.*

Resolution
CAFTA and SAPHIRE have been added to the SQAP-029 Controlled Dataset Book and the Admin 9.14 software summary sheets are filled out.

Attachment 2
Remaining Level B F&O's To Be Resolved

CA017297 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 TH-01, MAAP Documentation (PSA issue #170).
Due 5/30/2005

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.

CA017255 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 DE-01, Dependency Determination Guideline (PSA issue #187).
Due 6/1/2005

No guidance could be found applying to dependency determinations.

CA017251 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 DA-03, Equipment Failure Documentation (PSA issue #134).
Due 7/15/2005

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.

CA017253 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 DA-04, Electrical Component Failures (PSA issue #160).
Due 7/31/2005

Currently there is no plant specific data for electrical type components. There is generic electrical failure data.

CA017250 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 DA-02, Failure Mode Documentation (PSA issue #133).
Due 8/1/2005

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.*

CA017273 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 IE-09, Documentation Completeness (PSA issue #142).
Due 8/30/2005

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

CA017281 Level B Peer Review Comments for the Palisades PSA
Track closure of CEOG PSA peer review comment PEER-2000 QU-01, Quantification Guideline (PSA issue #198).
Due 7/31/2005

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.