Consumers Power
POWERING MICHIGAN'S PROGRESS
Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

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U.S. Nuclear Regulatory Commission
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DOCKET <u>50-255</u> - LICENSE <u>DPR-20</u> - PALISADES PLANT
GENERIC LETTER 88-20, SUPPLEMENT 4, INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS - REQUEST FOR ADDITIONAL INFORMATION

On June 30, 1995, Consumers Power Company submitted the response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events. On June 14, 1996, a request for additional information was received. This letter provides the requested information. The original request for additional information required a completion date of 60 days from the date of the NRC letter; however, due to the complexity of the questions, an extension of the response due date to September 30, 1996 was granted by the NRC Project Manager, Robert G Schaaf, per telephone conversation with Dale Engle of Consumers Power Company on July 11, 1996.

The attachment to this letter lists each of the individual requests for information and provides the Consumers Power Company response.

Enclosure 8 (Tables 3.5-1, 3.5-2, and 3.5-3) contains revised information that supersedes information on pages 3-36 through 3-42 of the IPEEE Report submitted on June 30, 1995. Therefore, pages 3-36 through 3-42 of the June 30, 1995 IPEEE Report are superseded in their entirety with enclosed tables 3.5-1, 3.5-2 and 3.5-3.

A011

SUMMARY OF COMMITMENTS

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

Robert A Vincent for.

Thomas C. Bordine
Manager, Licensing

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Attachment

ATTACHMENT

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
(NRC LETTER DATED JUNE 14, 1996)
GENERIC LETTER 88-20, SUPPLEMENT 4,
INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS

REQUEST FOR INFORMATION

PALISADES NUCLEAR PLANT
IPEEE REPORT

RESPONSES:

NRC letter dated June 14, 1996, requested additional information in respect to the Consumers Power Company's June 30, 1995 response to Generic Letter 88-20. Below is each request for additional information and the Consumers Power Company. response.

This document contains the responses to the NRC request for information on the Palisades IPEEE submitted June 1995. There was a revision submitted May 1996 that incorporates changes to the fire portion of the IPEEE that was not included in the NRC review. The responses note where the revised section contain the information requested. The responses are divided into two parts, seismic and fire.

A. Seismic

1. *Please provide (in a table) a complete list of anomalous conditions and outliers observed in the seismic walkdowns of all SPRA (seismic probabilistic risk assessment) equipment (including walkdowns for seismic-induced fires and floods). Anomalous conditions include anchorage concerns, interaction concerns, functional concerns, construction-adequacy concerns, seismic housekeeping concerns, etc. Please discuss the resolution of each of these items, noting any relevant plant improvements or analyses, and summarize their disposition status.*

Response:

Any equipment with an anomalous condition or considered an outlier was assigned a 0.1g High Confidence of Low Probability of Failure (HCLPF). Enclosure 1(Table A) contains all of the equipment with a screening value HCLPF of 0.1g. The table also contains the notes from walkdowns that were used to classify the component into the 0.1g screening value. Some of the components were not in the walkdowns and were assigned the 0.1g screening value to determine its importance on the Seismic Probability Risk Assessment (SPRA).

All equipment with a 0.1g screening value were determined to be acceptable as is, since none of these components significantly impacted the SPRA results (all had a risk achievement worth less than 1.15, see Enclosure 2 (Table B)). Therefore, no resolution is required.

2. *Table 3.5-1 lists only those SPRA components that did not screen out at 0.5g PGA (peak ground acceleration) HCLPF (high confidence of low probability of failure). In other words, the list excluded those components represented by means of the surrogate element. Please provide a list of all components that were addressed in plant seismic walkdowns, including those that were screened out at 0.5g PGA HCLPF. The result should be a complete table of all SPRA components. In this table, indicate which components were screened out at a HCLPF level of 0.5g PGA (i.e., those components represented by the surrogate element).*

<u>Response:</u>

Section 3.5 of the IPEEE report was revised to include three tables provided as Enclosure 8: Table 3.5-1; Table 3.5-2 and Table 3.5-3. Table 3.5-1 contains all components that were not represented by the surrogate event. Table 3.5-2 contains all components that were represented by the surrogate event. Table 3.5-3 contains fragility parameters for all seismic events used in the SPRA.

3. *Crude fragility assignments based on a 0.10g PGA HCLPF were made for a number of components in the SPRA. Some of the components modeled in this way are important to the overall plant capacity. Thus, calculations of actual HCLPF capacities for these components may reveal that the plant HCLPF is less than the reported value of 0.22g. In addition, NUREG-1407 guidelines specify that the fragilities "should be plant specific and rigorous to be able to identify dominant components and rank them."*

Please provide fragility calculation results for all components that were not screened out (i.e., for those components simply assigned a HCLPF capacity of 0.10g PGA) and for any additional components that have an important contribution to seismic core damage frequency (which may include some components screened out at a 0.3g PGA HCLPF level). Please modify Table 3.5-1 based on these calculations, and indicate in the table whether detailed or simplified fragility calculations were conducted. Please also modify Table 3.5-1 to show fragility parameters (median, β_c, HCLPF) for all components that were not screened out at 0.5g. (Please ensure that Table 3.5-1 is complete. For instance, it currently appears to be missing a fragility for loss-of-offsite power. In addition, no instance of "simplified" fragility is cited in the table, even though the submittal mentions that simplified fragility analysis has been employed. Also, the current table does not present fragility parameters produced from the detailed fragility calculations.) Please provide requantified accident sequence frequencies and seismic CDF (core damage frequency), using these refined fragilities, and a reassessment of the dominant risk contributors.

Response:

All equipment assigned a HCLPF of 0.1g is expected to have a much higher value if detailed fragility analysis were to be performed. Since detailed fragility analysis would result in higher fragilities for these components, the plant level HCLPF would be increased from its currently reported value of 0.22g.

Furthermore, none of the 0.1g HCLPF components were shown to be important to the seismic results. Enclosure 2 (Table B) contains the seismic basic events that appear in the cutset solution. Also included in this table are the importance measures for these seismic basic events. Defining important to the results as having a Risk Achievement Worth (RAW) ≥2.0 and Fussell-Vesely (FV) ≥5.0E-3, then only four seismic basic events are important to the results: J5PMSWS (Service Water System Pumps P-7A, P-7B, P-7C); J5PMP8C (Auxiliary Feedwater Pump P-8C); J5RE127D1 (Diesel Generator 1-1 UV relay 127D-1); and J3TKAB570 (High Pressure Air (HPA) Receiver Tanks T-9A&B). None of these four seismic basic events has a fragility of 0.1g. Also, detailed fragility calculations were developed for the first three seismic basic events. The HPA Receiver Tanks were screened at 0.3g HCLPF.

There are no fragility calculation results for components not screened out and assigned a 0.1g HCLPF. These components are expected to have higher fragilities (if detailed fragility analysis was performed) so the use of 0.1g HCLPF is conservative. Also, the use of a screening value (either 0.3g or 0.5g HCLPF) for a component is used only if the component would be expected to have a much higher fragility if detailed fragility analysis was performed. Therefore, most equipment does not have a detailed fragility calculation and is conservatively modeled.

No fragility calculations were modified and Table 3.5-1 has not been modified because of this response.

The report has been revised to add Table 3.5-3, Enclosure 8. Table 3.5-3 contains fragility parameters for all HCLPFs used in the seismic analysis.

Section 3.5.2 of the IPEEE report discusses the use of simplified fragility analysis. A simplified method was used to develop fragilities for initial quantification of the seismic model. However, the final fragilities used were either screening or detailed fragilities. This allowed the team to focus their time on the potentially significant seismic contributors early in the process. Later in the evaluation, there was more time to calculate detailed fragilities for all components not screened out.

No requantification was performed as a result of question A.3.

4. *Please report the percent contribution to the seismic CDF that is due to failure of the surrogate element. Discuss the expected changes in the seismic CDF and in the failure contribution of the surrogate element if a separate surrogate element failure event were to be included (as an element in series with existing system fault tree logic) for each system in the SPRA model.*

<u>Response:</u>

As shown in Figure 3.6-4 of the IPEEE Report, the surrogate event contributes to 5.9% of the core damage frequency (CDF) for seismic initiated events.

Palisades' SPRA modeled one plant level surrogate event versus a system level surrogate event for each system fault tree. It is more conservative to model one plant level surrogate event that represents all equipment screened out at a 0.5g HCLPF. This way, all of the screened out equipment is totally coupled together - if one fails due to a seismic event then all equipment fails due to the same seismic event. If individual system surrogate events that are independent of each other (but all have the same fragility of 0.5g HCLPF) are modeled, then all of the equipment is not totally coupled together. If individual system surrogate events are modeled that are named the same, then the results would be exactly the same as the method employed. Therefore, no sensitivity was performed to identify the impact of modeling individual system level surrogate events since the method used is conservative.

5. *The submittal simply notes that seismic initiators/events, other than small break loss-of-coolant accident, loss-of-offsite power, and turbine building fires and floods, were screened out based on low (yet unreported) probabilities of occurrence. Please provide screening values and their bases that were used to exclude other potential seismic initiators/events that may be modeled in an SPRA. Describe the basis for assessing the bounding probabilities of occurrence for each initiator, and discuss any insights related to the conditional probabilities of core damage given occurrence of each of these events.*

<u>Response:</u>

There are several initiating events listed in NUREG/CR-4840 (vessel rupture; large, medium, small break LOCAs; transients with loss of off-site power; and transients with Primary Coolant System available). The small break LOCA probabilities were derived from NUREG/CR-4840. A detailed fragility was calculated for the loss of off-site power initiating event. Industry documents (EPRI NP-6041-SL and NUREG/CR-4334) conclude that large and medium break LOCAs and vessel rupture would occur at high ground motions such that all plant safety systems would also fail (in excess of 0.5g HCLPF). Therefore, no insights would be gained from including medium and large break LOCAs or vessel ruptures as initiating events in the SPRA.

6. *Please describe how the time history was generated for obtaining input for the soil-structure interaction analyses. Provide the following: (a) a plot of the acceleration time history; (b) a plot of the power spectrum of the time history; and (c) a plot of the response spectrum of the time history as compared to the target response spectrum.*

<u>Response:</u>

Enclosure 4 contains the requested information (including the plots). This attachment is Stevenson & Associates Calculation No. C-003, revision 1, dated 1/17/94.

7. *Please provide a discussion of the treatment of mission times, failure dependencies (e.g., of similar, co-located equipment), and of other inter-related failure effects (that were not discussed in the IPEEE report) within the SPRA model. What are the relevant numerical values used in the analysis pertaining to these effects? How were these values obtained, and how were they used in the SPRA model? What are their impacts on the SPRA results?*

<u>Response:</u>

Mission times remain the same as in the Individual Plant Examination (IPE) - 24 hours. Failure dependencies and inter-related failure effects are treated in the fault trees. Components of similar design located in the same building and the same floor elevation were given the same seismic basic event. This essentially acts as a totally dependent seismic failure between similar components where seismic failure of one component leads to seismic failure of all similar components. For example, Service Water System (SWS) Pumps (P-7A/B/C) were all assigned the same seismic basic event J5PMSWS, with a HCLPF of 0.52g. A similar method was used for modeling potential interactions (such as block walls). Each affected component was assigned the same seismic basic event that represented the seismic failure of the interaction and thereby failing all of the affected components simultaneously upon failure of the interaction. These types of seismic basic events were evaluated similar to component seismic basic events for calculating fragilities and for inserting into the seismic fault trees. The seismic results contain the common seismic basic event. Enclosure 3 (Table C) contains the list of all seismic basic events that represent more than one component and the components they represent.

8. *Please list the human actions modeled in the SPRA and their associated IPE human error probabilities and their seismic fragilities. Please indicate when and where each human action is required.*

Response:

The SPRA used the same human actions that were credited in the IPE for the same accident sequences. No new operator actions were credited in the SPRA. Only the post-accident human error probabilities (HEPs) were modified based on the timing and location of the operator action. IPEEE Report Section 3.6.5.2.2 discusses the treatment of operator actions in the SPRA. The operator actions were divided into three categories: performed in the Control Room within one hour; performed outside the Control Room within one hour; and performed after one hour (regardless of where the action is performed). Non-lognormal fragilities were assigned to all of the post-accident operator actions.

The post-accident operator actions to be performed were assumed to be unaffected by ground motions up to the design basis (0.2g). Therefore, no HEP's from the IPE were modified for ground motions ≤0.2g. For ground motions greater than the design basis, short-term operator actions (required within one hour) were likely to be affected, with actions performed outside the Control Room affected more. Long-term operator actions (not required within one hour) were assumed not to be affected by ground motion and used the IPE HEP's. Based on this reasoning, the following non-lognormal HEPs were used in the SPRA:

Category 1 HEPs (in Control Room within one hour) were modified to use the IPE HEP up to a 0.2g earthquake (design basis). From 0.2g to 0.4g, the HEP linearly increased from the IPE HEP to 0.5. From 0.4g to 0.6g, the HEP linearly increased from 0.5 to 1.0. Above .6g, the HEP was set to 1.0.

Category 2 HEPs (outside Control Room within one hour) were modified to use the IPE HEP up to a 0.2g earthquake (design basis). From 0.2g to 0.4g, the HEP linearly increased from the IPE HEP to 1.0. Above .4g, the HEP was set to 1.0.

Category 3 HEPs (not required within first hour) were not modified and used the IPE HEPs for all earthquakes.

9. *Please provide HCLPF calculations and results, completed screening evaluation work sheets (SEWS), walkdown notes/checklists and photographs for the following SPRA-significant components:*

- *CST [Condensate Storage Tank]*
- *SIRWT [Safety Injection and Refueling Water Storage Tank]*
- *Control Panel for Fire Pumps*
- *Diesel Day Tanks (T-24 and T-40) for Fire Pumps*
- *Block Walls Supporting the Diesel Day Tanks for the Fire Pumps*
- *Station Transformer 13*

* *MSIVs [Main Steam Isolation Valves]*
* *Diesel Generator (DG) Fuel Oil Tank (T-10)*
* *DG 1-2 Undervoltage*

Please respond to question A.3 above before providing this information.

Response:

Enclosure 5 contains copies of the following supporting documentation:

1) CST (T-2): outlier seismic verification sheet (OSVS), screening evaluation work sheet (SEWS) and S&A calculation EA-POC0007899-T2, Rev 1, dated 12/7/95;
2) SIRWT (T-58): OSVS, SEWS and S&A calculation C-019, Rev 0, dated 5/19/95;
3) Control Panel for Fire Pumps (EC-137): as-built drawing (URS/JOHN A. BLUME Dwg. No. 6 dated 4-11-80), see notes for EC-137 in Table A;
4) Diesel Day Tanks (T-24 and T-40) for Fire Pumps: T-24 and T-40 are exactly the same, see notes for T-40 in Table A;
5) Block Walls Supporting the Diesel Day Tanks for the Fire Pumps: see notes for T-40 in Table A;
6) Station Transformer 13 (EX-13): see notes for EX-13 in Table A;
7) MSIVs (CV-0501/0510): SEWS;
8) Diesel Generator (DG) Fuel Oil Tank (T-10): OSVS, SEWS and S&A calculation C-001, Rev 0 dated 8/20/93;
9) DG 1-2 Undervoltage (127D-2 in cabinet EC-26): SEWS.

10. *The submittal's discussion in the seismic-fire interaction evaluation does not adequately address the relevant concerns for seismic degradation of fire suppression systems. The discussion focuses only on potential interactions of FPS (fire protection system) components with essential equipment. The evaluation should also include an examination of potential loss of FPS capability itself due to seismic events, especially since credit for this system is taken in the SPRA model. Examples of items found in past studies include (but are not limited to):*

* *Unanchored CO_2 tanks or bottles*
* *Sprinkler standoffs penetrating suspended ceilings*
* *Weak or unanchored 480V or 600V (nonsafety-related) electrical cabinets (as potential fire sources) in close proximity to essential safety equipment (e.g., cables in cable spreading room)*
* *Fire pumps unanchored or on vibration isolation mounts*
* *Mercury or "bad actors" relays in fire protection system (FPS) actuation circuitry*
* *Use of cast iron fire mains to provide fire water to fire pumps*

NUREG-1407 suggests a walkdown as a means of identifying any such items.

Please provide the results of your seismic-fire interaction study pertaining specifically to seismic degradation of FPS capability. Also, include the guidelines given to walkdown personnel for evaluating the foregoing issues (if they exist).

Response:

Three areas were studied as part of the seismic-fire interaction evaluation: seismically induced fires; inadvertent actuation of Fire Suppression Systems; and seismic degradation of Fire Suppression Systems.

Seismically induced fires were evaluated by assembling a list of significant combustible sources (significant quantities with ignition points below about 500°F) and performing a walkdown to assess whether the sources are both significant hazards and seismically vulnerable. All potential fire sources were walked down in the power block buildings. Combustible sources such as fuel oil tanks, waste gas tanks, hydrogen gas bottles, flammable liquid storage cabinets, and hydrogen piping were assessed.

Inadvertent actuation of fire suppression was evaluated via a walkdown and relay review. Walkdown personnel were instructed to observe the potential for spray-down or release of fire suppression media due to seismic interaction. No such instances were observed at Palisades. No written instructions were provided to the walkdown team members, however all team members were trained and received certification by completion of the SQUG/EPRI sponsored training course entitled "SQUG Walkdown Screening and Seismic Evaluation Training Course." In addition, fire control equipment (panels and cabinets) were walked down to ensure they were properly anchored and not subject to potential seismic interactions. A functional relay review was performed to identify low seismic capacity "bad actor" relays which might not perform well in seismic events which could potentially lead to inadvertent suppression actuation. No such instances were observed at Palisades.

Seismic degradation of Fire Suppression Systems was reviewed by walking down fire piping and looking for poor structural design features or potential interactions with equipment. This was routinely performed for each equipment item during the walkdown phase. No such potential interactions were noted.

11. *Please report the final plant HCLPF capacity after responding to the preceding questions. Include plots of the plant HCLPF spectrum and the SSE (safe shutdown earthquake) spectrum on the same graph. Please justify the spectral shape used for reporting the plant HCLPF spectrum.*

Response:

The final plant HCLPF did not change based on the response to the seismic questions presented in this document.

The spectral shape used for reporting the plant HCLPF was from NUREG-1488. NUREG-1407 approves of the use of the 10,000 year median spectral shape from NUREG-1488.

The plot containing the plant HCLPF and the Safe Shutdown Earthquake (SSE) spectrum is provided as Enclosure 6.

B. Fire

NOTE: A revision to the fire IPEEE analysis was submitted to the NRC in May 1996. This revised report includes some of the information requested here. The information referenced in the revised report is specifically noted in the responses.

1. *The study assumed that only a single control room cabinet would be affected by a suppressed fire. In fact, it is assumed a particular cabinet (C01) would be affected. It is typical for plants to have cabinets with open sides which would allow propagation of fire (or smoke) damage into another cabinet. This might occur before operators are able to act to suppress the fire. control rooms are also susceptible to fires that start from other sources such as waste baskets and kitchen areas. Therefore, this assumption may actually underestimate core damage frequency. Furthermore, the assumption that C01 is the damaged cabinet only allows vulnerabilities to be discovered with respect to failures in that cabinet. The state-of-the-art assessment includes analysis of fires postulated to initiate from each fire source in the control room.*

   a. *Please provide a discussion of the potential for inter-cabinet fire propagation owing to open-sided cabinets at Palisades before operators can suppress the fire.*

   b. *Please provide a discussion of how the dominant sequences would be affected by assuming the fire initiates in other control room cabinets. For each cabinet in the control room, include a discussion of the equipment that is affected and the sequences that are most significant to the conditional core damage probability.*

   c. *Similarly, please provide a discussion of the potential of fire growth from other fire sources in the control room area. Include the potential to* *propagate to overhead cables, computers, cabinets, and consoles. For each fire source in the control room, include a discussion of the equipment that is affected and the sequences that are most significant to the conditional core damage probability.*

Response:

a) Section 4.10.2 in the revised report discusses the method for evaluating Control Room fires. The walkthrough cabinets in the Control Room have a walkway along the back of the cabinet, but no significant combustible material is present that would propagate the fire between cabinets before the operators would have time to extinguish the fire. Other cabinets in the Control Room do not have open sides and will not propagate fires between cabinets.

b) Section 4.10.2 in the revised report discusses the method for evaluating Control Room fires. Each cabinet in the Control Room that contained equipment or cables credited in the SPRA was evaluated for a fire in the revised fire analysis. Each cabinet fire initiating frequency was assigned a value equal to the cumulative fire initiating frequency for all cabinets in the Control Room. The cabinet with the highest core damage frequency is the bounding cabinet fire analysis for the Control Room. Only the results of the cabinet with the highest core damage frequency was then reported as the core damage frequency for cabinet fires.

c) Section 4.10.2 in the revised report discusses the method for evaluating Control Room fires. The two types of fires evaluated in the revised fire analysis in the Control Room are cabinet fires and exposure fires. Exposure fires are fires that are initiated by sources outside of cabinets.

The revised fire analysis assumes that even if an exposure fire was successfully suppressed, the fire would damage one entire system. This is conservative because each system and each channel in a system is sufficiently separated that only one channel from one system is likely to fail. Two systems were evaluated for damage: Auxiliary Feedwater (AFW) and High Pressure Safety Injection (HPSI). These two systems were chosen because they have the greatest impact on core damage frequency. Each of these systems was evaluated with the one that contributes the most to core damage frequency chosen as the reported failed system.

If the fire is not successfully suppressed, then the entire Control Room is assumed to be engulfed in the fire and everything (including the cabinets) is assumed to have fire damage and fail all equipment and cabling in the Control Room.

The results of the revised fire analysis are shown in Table 4.11-1 of the revised IPEEE report.

2. *The probability of manual suppression before damage is a function of both the probability of fire damage, as a function of time, and the probability of successfully completing fire suppression activities which is a also a function of time. For example, FIVE (fire induced vulnerability evaluation) methodology suggests that if a critical combustible loading is present, then the time to damage, t_{crit}, is calculated. The probability of non-suppression depends on the relationship between t_{crit} and the demonstrated fire brigade response and extinguishment times. In contrast, the study appears to simply have assigned a value of 0.01 for the probability of failure of manual suppression in the control room. Either (a) provide an explanation of the analysis performed to develop the control room manual non-suppression probability, and demonstrate that 0.01 is a realistic estimate, or (b) discuss the effect of using a more typical number (e.g., 0.1).*

Response:

Section 4.8.3.2 in the revised report discusses the reasons for assigning the lower value for failure to manually suppress a fire in the Control Room. The reasons are:

1) Failure to detect a fire in the Control Room is negligibly small due to the redundancy and diversity of cues and due to the continuous staffing of the Control Room.

2) All Control Room operators are trained in fire suppression techniques, therefore, very early detection and immediate action to suppress a fire is very likely.

3) There is minimal combustible material (loading) in the Control Room outside of the cabinets.

If a more conservative value of 0.1 were used, the core damage frequency for fires would rise approximately 36% to 4.54E-5/yr from 3.32E-5/yr. The bulk of this increase would be in Accident Class 1A, which is dominated by sequences where the Control Room is abandoned. No conclusions would change as a result of using the more conservative value, except that the Control Room would be a higher contributor to CDF than the Cable Spreading Room (currently the highest contributor to CDF).

3. *Sandia has performed experiments to investigate a reasonable range of times to operator abandonment of the control room. These indicate that poor visibility can force abandonment within 6 to 8 minutes from the time flame is visible in a*

cabinet. The Palisades IPEEE, however, simply assumed that unsuppressed fires would require abandonment and suppressed fires would not, without regard to timing. Please discuss how suppression will be achieved before operators would be forced to abandon the control room. What is the probability of suppression before abandonment? What is the probability that smoke will force abandonment even if suppression is successful?

Response:

The response to fire question 2 above provides the reasons for using a lower value for the manual suppression failure probability. These reasons also apply to timing and abandonment of the Control Room. A fire in the Control Room is expected to be detected and suppressed before conditions would lead to abandonment of the Control Room. There are smoke detectors/alarms in the tall cabinets, and the Control Room is continuously occupied to provide early visual detection. Also, portable fire extinguishers (for which all Control Room operators are trained to use) are located at various places around the Control Room for quick response. The probability used for failure to manually detect and suppress a fire in the Control Room includes the failure to detect and suppress the fire prior to forcing abandonment of the Control Room.

4. *The study assumed that any fire in the cable spreading and switchgear rooms would be limited to a single system, the auxiliary feedwater system, if suppression were successful. The assumption that only the auxiliary feedwater is damaged limits discovery of vulnerabilities related to these failures. It is not obvious that this assumption is bounding. Please provide a discussion of how the dominant sequences would be affected by assuming the fire initiates in other fire sources (e.g., cabinets, MCCs (motor control centers), panels, motor generator sets). For each source in these rooms, include a discussion of the equipment that is affected and the sequences that are most significant to the conditional core damage probability.*

Response:

Section 4.10.3 in the revised report discusses the method for evaluating cable spreading or switchgear room fires (similar to control room cable fires). The two types of fires evaluated in the revised fire analysis in the Cable Spreading and Switchgear Rooms are cabinet fires and exposure fires. Exposure fires are fires that are initiated by sources outside of cabinets.

The revised fire analysis assumes that even if an exposure fire was successfully suppressed, the fire would damage one entire system. This is conservative because each system and each channel in a system is sufficiently separated that only one channel from one system is likely to fail. Two systems were evaluated for damage: AFW and HPSI. These two systems were chosen because each system and each channel in a system is sufficiently separated that only one channel from one system is likely to fail. Two systems were evaluated for damage: AFW and HPSI. These two systems were chosen because they have the greatest impact on core damage frequency. Each of these systems was evaluated with the one that contributes the most to core damage frequency chosen as the reported failed system.

If the fire is not successfully suppressed, then the entire fire area is assumed to be engulfed in the fire and everything (including the cabinets) is assumed to have fire damage and fail all equipment and cabling in the fire area.

Each cabinet in the fire area that contained equipment or cables credited in the SPRA was evaluated for a fire. For each fire area, each cabinet fire initiating frequency was assigned a value equal to the cumulative fire initiating frequency for all cabinets in the fire area. The cabinet with the highest core damage frequency is the bounding cabinet fire for the fire area. Only the results of the cabinet with the highest core damage frequency was then reported as the core damage frequency for cabinet fires in that fire area.

The results of the revised fire analysis are shown in Table 4.11-1 of the revised IPEEE report.

5. *Initiating events seem to be limited to transients with loss of the power conversion system. Loss-of-offsite power owing to turbine building or switchgear room fires has emerged as an important contributor in other studies but does not appear to have been considered in this study. The submittal claims that no fire initiator was identified that could credibly lead to a LOCA (loss-of-coolant accident). The process used to search for such initiators was not provided. Furthermore, the transient event tree used does not include transient induce LOCAs, such as stuck open relief valves. Although fire-induced hot shorts are typically unlikely to cause LOCAs, this potential was not mentioned in the submittal. Please,*

   a. *Discuss the potential for loss-of-offsite power to safe shutdown systems because of fires in the turbine building and auxiliary building.*

   b. *Discuss the potential for fires to create a LOCA (such as a stuck open relief valve), or interfacing LOCA (such as opening of shutdown cooling system isolation valves), particularly considering the potential of fire-induced hot shorts to initiate valve motion.*

   c. *Explain how transient-induced LOCAs, such as stuck open relief valves, are treated in the study. If they were not treated, discuss how the dominant sequences would be affected by including them.*

Response:

a) The potential for loss of off-site power was explicitly evaluated in the accident sequence quantification. The equipment in the power distribution system is included in the fault tree models. Off-site power would be lost if power distribution equipment were located in the fire area evaluated and were affected by a fire. For example, cable from the safeguards transformer (fed by off-site power) to one of the safety buses and cable for the fast transfer to startup transformers is routed through the Turbine Building. A fire in the turbine building is assumed to short these cables to ground and disable the safeguards transformer and result in no fast transfer to the startup transformer. This results in the two safeguards buses being fed from the diesel generators.

   This type of modeling captured all power related failures in the fault tree so no assumptions were made as to whether off-site power was lost.

b) Fires are assumed not to contribute to LOCAs initiated by pipe breaks as the piping material would not be affected by a fire. However, the potential for LOCAs due to equipment failures was considered (such as stuck open relief valves).

   The Palisades Plant is operated with the block valves from the PORV's normally isolated. For a fire to result in a PORV LOCA a false signal would have to be generated simultaneously to both a PORV and its corresponding block valve.

   The suction of the Shutdown Cooling System is isolated from the Primary Coolant System by two normally closed isolation valves. The discharge is isolated by a normally closed isolation valve and a check valve.

   For a fire to initiate a LOCA, multiple and simultaneous hot shorts must occur or a hot short must occur with random failure of a redundant component. When combined with the frequency of a fire, it was concluded that these initiators had a sufficiently low probability of occurrence not to be modeled explicitly in the fire PRA.

c) LOCAs due to random failures of equipment (such as a stuck open relief valve) are built into the fault and event trees. As an example, the PORVs do not get a demand unless the steam generators are lost as a heat sink. Loss of secondary heat removal is an explicit heading in the event trees leading to the need to open the PORVs and block valves for the purpose of initiating once through cooling. Demands on the safety relief valves and the potential for their failing open are anticipated only during ATWS conditions.

6. *The failure probability for automatic suppression used the FIVE values. This data is acceptable for systems that have been designed, installed, and maintained in accordance with appropriate industry standards, such as those published by the National Fire Protection Association. It is not clear that the assumption, used in the study, that automatic suppression is capable of limiting fires to a single system is conservative in all case. Please,*

   a. *Describe the survey or walkdown that was performed to determine if sprinklers are installed in accordance with industry standards.*

   b. *Provide the estimate of delay time for sprinkler actuation and fire suppression in these areas.*

   c. *Describe the analysis or evaluations that determined that automatic suppression would limit damage to a single system (e.g., in the cable spreading room) or single power bus (e.g., in the switchgear rooms).*

<u>Response:</u>

a) Automatic sprinklers were credited only for the Cable Spreading and Switchgear Rooms. The sprinklers in these rooms were installed in accordance with NFPA 13 (1968 - which is the code of record).

b) Delay time estimates were not used at Palisades. Palisades conservatively assumes that even if the fire was successfully suppressed, the fire would burn long enough to damage one system. This is conservative because each system and each channel in a system is sufficiently separated that only one channel from one system is likely to fail. As noted previously, two systems were evaluated for damage: AFW and HPSI. These two systems were chosen because they have the greatest impact on core damage frequency. The system that contributes the most to core damage frequency chosen as the reported failed system.

c) Section 4.10.3 in the revised report discusses the method for evaluating Cable Spreading or Switchgear Room fires. The two types of fires evaluated in the revised fire analysis in the Cable Spreading and Switchgear Rooms are cabinet fires and exposure fires. Exposure fires are fires that are initiated by sources outside of cabinets.

   The only time that one system is assumed to be failed due to the fire is a successfully suppressed exposure fire (see response 6a above). For all other types of fire, all equipment in the area with the fire is assumed to fail (all equipment in the fire area for exposure fires not suppressed and all equipment in a cabinet being evaluated).

7. *Although the IPEEE study recognized that operators may be required to abandon the control room because of a fire in the control room, it did not appear to recognize that they may have to abandon the control room because of a large fire in the cable spreading room which renders the controls in the control room inoperable. Please,*

   a. *Discuss how operators would respond to a fire in the cable spreading room that disables controls of key safe shutdown functions from the control room.*

   b. *Describe the affect on the dominant sequences and core damage frequency owing to fires in the cable spreading room including the potential for operator abandonment of the control room.*

   Response:

   a) Section 4.10.3 in the revised report discusses the method for evaluating cable spreading room fires. The two types of fires evaluated in the revised fire analysis in the Cable Spreading Room are cabinet fires and exposure fires. Exposure fires are fires that are initiated by sources outside of cabinets. In the event that there is an exposure fire that is not successfully suppressed, the revised fire analysis assumes that the operator would abandon the Control Room.

   b) The potential for abandoning the Control Room for Cable Spreading or Switchgear Room fires is included in the revised fire analysis as discussed in Section 4.10.3 in the revised IPEEE report.

8. *The study assumed that fire barriers would always be effective, as rated, at limiting fires and smoke to a single area. However, it is not clear that the study considered active fire barriers (e.g., a normally open fire door closed by a fusible link, or a similarly actuated open damper). Please provide an analysis of the effects on the results (i.e., dominant sequences and dominant areas):*

   a. *If the potential for the failure of active barriers is considered in all areas, and*

   b. *If the potential for cross-zone fire propagation is considered for high hazard areas such as the turbine building, diesel generator room, switchgear rooms, and lube oil storage areas.*

   Response:

   a) The fire barriers were identified and evaluated during the Appendix R evaluation. The potential for active fire barrier failure was not considered for any fire area/zone. Active fire barriers were installed and are periodically tested and surveilled in accordance with the applicable codes and standards. Furthermore, the Appendix R program performed walkdowns and hydraulic calculations that revalidated that the active fire barriers meet or exceed the original design specifications.

b) Fire areas and zones were evaluated in detail to provide a high confidence that fire would not propagate from one fire area/zone to another. The fire areas are bounded by three hour rated walls and doors and are defined in the Appendix R analysis. Some fire areas were divided into smaller fire zones for the fire analysis. These fire zone divisions were performed in accordance with the FIVE methodology.

9. *The submittal used the FIVE methodology and database for fire initiation frequencies but two areas were shown in the submittal with frequencies lower than the base frequencies found in Table 10.2 of the "Fire Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide," TR-100370, April 1992. The FIVE methodology would give an initiation rate of least $3x10^{-2}$ per year for each diesel generator room. The submittal (Table 4.1.7.3) shows the frequency of each room as approximately $1.7x10^{-2}$ per year for each diesel generator room. Please provide the calculation details and explanation for the fire frequency in these areas.*

<u>Response:</u>

Enclosure 7 contains the Ignition Source Data Sheets (ISDS) for the Diesel Generator Rooms (fire areas 5 & 6). The ISDS were completed in accordance with the FIVE methodology and result in the fire initiating frequency for each diesel generator room to be $1.7x10^{-2}$ per year.

10. *Human action are identified as important to core damage frequency estimates. However, no details are provided regarding how the human error probabilities were assessed. Please provide a description of how fire event recovery actions (e.g., control room abandonment and use of the alternate shutdown panel, local manual operation of auxiliary feedwater pumps, opening of atmospheric dump valves, initiation and control of once through cooling) were assessed. Include how factors such as timing and environmental stressors (e.g., reduced visibility, impaired communications, impaired accessibility) were considered. If IPE values were used, how were they adjusted to reflect the fire-related environmental stressors? If IPE values were not adjusted, provide the rationale for not having adjusted the values.*

Response:

The IPE reduced the reliance on operators to mitigate accidents by crediting only Control Room operator actions and a minimum number of risk significant operator actions outside the Control Room. The performance shaping factors for the Control Room operator actions were reviewed and determined to be acceptable and the IPE HEPs were used in the fire PRA. The operator actions performed outside the Control Room were not required to be performed within the first hour. Therefore, their performance shaping factors were determined to not be affected and the IPE HEPs were used in the fire PRA.

The operator actions credited in the fire PRA but not credited in the IPE were specifically evaluated for each fire area. The performance shaping factors for each fire area were reviewed to determine the HEP for the operator action or a screening value of 0.1 was used.

11. *The study stated on Page 4-7 that it used the FIVE method for screening out inter-zonal fire propagation. The control area is divided into three zones as described on Pages 4-17 and 4-18. It appears from Table 4.1.6.1 that each of these zones was retained for analysis and analyzed as individual zones without the potential for inter-zonal fire propagation. In view of the fact that 30% of the separating wall is ordinary glass, which does not constitute a fire barrier, please (a) explain why these zones were not considered as a single area in the analysis, and (b) discuss the effect on the results (e.g., dominant sequences and core damage frequency) of considering Zones 1A, 1B and 1C as a single entity.*

Response:

The original fire analysis assumed that a fire would propagate between the three fire zones identified for the Control Room (i.e., a fire in zone 1B leads to a fire in zone 1A) effectively evaluating them as one fire area. Therefore, there is no effect on the results for fire area 1 (Control Room).

12. *Page 4-14, Assumption 1 states that engineering analysis concluded that fire spread between transformers and the turbine building is not credible. In light of the occurrence of such fires, and the obvious potential of fire spread between a large station power transformer and the turbine building, please provide the referenced analysis (Ref. 4-6 of the submittal).*

Response:

There are two sets of transformers near the turbine building: startup transformers; and station power transformers. An Appendix R evaluation was performed that concluded that a fire in one set of transformers would not spread to the other set of transformers or the turbine building. The analysis conclusion is based on three factors. Each set of transformers has a separate deluge system to extinguish and prevent fire spread. Also, there is a fire wall between each set of transformers and the turbine building to prevent fire spread. In addition, each transformer is on a rock bed that acts as a collection barrier for any leaking oil or deluge water which prevents fire spread due to burning oil. Based on these factors, fire spread between these transformer sets and the turbine building was deemed to not be credible and was not considered in the fire analysis.

ENCLOSURE 1

TABLE A,
LOW CAPACITY COMPONENT NOTES

ENCLOSURE 1

TABLE A
Low Capacity Component Notes

Equipment ID	Field Notes
C-50A	Screen at 0.1g - anchorage unknown
C-50B	Screen at 0.1g - anchorage unknown
CV-0501	REF: 1) CPCo Drwg. # M120, Sh. 1, Rev. 10A. 2) CPCo Drwg. # M120, Sh. 20, Rev. 2. 3) CPCo Drwg. # M120, Sh. 21, Rev. 1. 4) CPCo Drwg. # M120, Sh. 22, Rev. 2. Piston-operated valve. BSCav2: The valve body is cast steel A 216 WCB. (REF 1). BSCav3: Combination cylinder cap and support yoke is cast steel A 216 WCB (REF. 1). BSCav4: Pipe diam. is 30" (REF 1). BSCav6: Verified using Figure B7.2. Offset is 36" (REF 1) and estimated operator weight by Seismic Review Team (SRT) is less than 750# < 100" and 750# (max. acceptable for a 30" pipe diam.).
CV-0510	REF: 1) CPCo Drwg. # M120, Sh. 1, Rev. 10A. 2) CPCo Drwg. # M120, Sh. 20, Rev. 2. 3) CPCo Drwg. # M120, Sh. 21, Rev. 1. 4) CPCo Drwg. # M120, Sh. 22, Rev. 2. Piston-operated valve. BSCav2: The valve body is cast steel A-216 WCB (REF 1). BSCav3: Combination cylinder cap and support yoke is cast steel A 216 WCB (REF. 1). BSCav4: Pipe diameter is 30" (REF 1). BSCav6: Verified using Figure B7.2. Offset is 36" (REF 1) and estimated weight by SRT is less than 750# < 100" and 750# (max. acceptable for a 30" pipe diam.).
CV-0511	BSCav5: Verified using Figure B7.1. Measurement Offset is 40" < 45" (maximum acceptable for a small pipe diameter greater than or equal to 1"). Interactions: OUTLIER - Grating overhead is unsecured.

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
CV-0857	BSCav4: Mounted on a large line BSCav5: Verified using Figure B7.1. Meas. Offset is 36" < 45" (max. acceptable for a small pipe diam.). Therefore, offset must be acceptable for large pipe diam. There is a positioner (part of the valve) that is about 1/4" from an reinforced concrete wall - outlier due to potential impact.
CV-0944A	REF: 1) CPCo Manual # M-354, Sheet 51, Rev. 0. 2) CPCo. EA-SP-03312-01, Rev. 1. BSCav2: Valve body is not cast iron by SRT inspection. BSCav4: Meas. pipe diam. is 12". BSCav6: Verified using Figure B7.2. Meas. Offset is 32" and weight (REF 1) is 270# < 80" and 750# (max. acceptable for a 12" pipe diam.). Interaction. Attached solenoid is near a structural member, pipe may move enough to shear off solenoid. SRSS pipe displacement during SSE is about 0.1" (REF 2). Judged OK by SRT.
CV-1037	2" away from CV-1101 on very flexible pipe. They will collide in moderate earthquake. Check piping displacement if available to establish HCLPF. Screen at 0.1g
CV-1045	Interactions: Valve bearing against Containment wall. May fail in seismic event. Screen at 0.1g
CV-1101	2" away from CV-1037 on very flexible pipe. They will collide in moderate earthquake. Check piping displacement if available to establish HCLPF. Screen at 0.1g
CV-3001	REF: 1) Palisades Dwg. 5935-M233-71-2. 2) Palisades Dwg. 950W22-M233-72-1. BSCav2: Valve body is ASTM-A351 Type 316, Gr. CF8M (REF 1). BSCav4: The valve is 6" 300# B. W. Sch. 40 (REF 1). BSCav5: The valve operator is 32-3/4" long (REF 2) < 60" for a 6" valve per Fig. B.7-1. Interaction: Operator diaphragm 1" away from another pipe which, through impact, may cause valve failure. Screen at 0.1g

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
E-1/2/3/4/5/6A/B	Not seen - screen at marginal 0.1g value
E-10	Not seen - screen at marginal 0.1g value
E/P-0511	Small device on floor-to-ceiling strut frame. Base anchorage corroded but overall, frame is solid based on tug testing and since it has a fundamental frequency > 8 Hz, it is judged OK by inspection. OUTLIER- Grating overhead is unsecured.
EB-03	Screen at 0.1g - anchorage unknown
E-14	Not seen - screen at marginal 0.1g value
EB-77	Screen at 0.1g - anchorage unknown
EB-79	Screen at 0.1g - anchorage unknown
EC-137	REF: URS/JOHN A. BLUME & ASSOC., Job # 8013, Drwg. # 6, Sh. 1, Rev. 1. Equipment is located in 136 (Turbine Building) which has been seismically qualified. Outlier: Base entirely corroded and appears to be unanchored. It is attached at base (but not above base) to EC-131.
ED-36A/B	Screen at 0.3g OUTLIER - Batteries are not confined by Styrofoam spacers. Anchored by 4-1/2" anchors, total.
ED-36C/D	OUTLIER - Batteries are not confined by Styrofoam spacers. Anchored by 4-1/2" anchors, total. Screen at 0.3g

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
ED-38A/B	REF: 1) URS/JOHN A. BLUME & ASSOC., Job # 8013, Document (Consultation for Safety-Related Electrical Equipment at Palisades Nuclear Power Plant), 5/20/81. 2) Field sketch by D. Engle 8/2/94 (see document). Equipment is located in 136 (Turbine Building) which has been seismically qualified. Batteries are Lead Plate type (DELCO REMY 761A) and battery is replaced as cell fails. OUTLIER - Batteries are not confined by Styrofoam spacers and battery rack does not have longitudinal cross bracing. Anchorage: Anchored by 4-1/2" expansion anchors, total. Weight is 612 # (REF 1). Anchorage was installed in accordance with CPCo Spec. C-97(Q), Rev. 17, "Furnishing, Installation, and Inspection of Expansion-type Concrete Anchors". SRT has reviewed samples of anchor bolt inspection records and found acceptable. Thus, no tightness check of bolts was performed during SRT inspection. Screen at 0.3g
ED-38C/D	Identical to ED-38A/B Screen at 0.3g Equipment is located in 136 (Turbine Building) which has been seismically qualified. OUTLIER - Batteries are not confined by Styrofoam spacers and battery rack does not have longitudinal cross bracing. Anchorage: Same as ED-38A/B
EX-13	Attached to load centers 13 and 14 and EX-14 on other end. All basically one unit sitting on thin, 1/4" plate resting on grating. No apparent anchorage to plate. Plate is marginally tack welded to grating. Basically, unit is unanchored. Set HCLPF accordingly low.
EX-77	Screen at 0.1g - anchorage unknown
HIC-0826	Mounted on strut frame which has one floor-to-ceiling strut and the other post is attached neither to the ceiling or floor and can swing freely at a frequency << 8 Hz. may be seismically vulnerable as it is a poor design. Screen at 0.1g

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
HIC-0881	Mounted on strut frame which has one floor-to-ceiling strut and the other post is attached neither to the ceiling or floor and can swing freely at a frequency << 8 Hz. may be seismically vulnerable as it is a poor design. Screen at 0.1g
HIC-0882	Mounted on strut frame which has one floor-to-ceiling strut and the other post is attached neither to the ceiling or floor and can swing freely at a frequency << 8 Hz. may be seismically vulnerable as it is a poor design. Screen at 0.1g
LS-0204	Seismic interaction (at risk from) tank T-81
LS-2019	Seismic interaction (at risk from) tank T-81
M-59A	See field sketch. Large unit anchored by 12 - 7/8" CIP anchors (6 per side). Do anchorage analysis to obtain HCLPF. Anchors in 3 groupings per side that are 6" apart. A Evaporator is adjacent to unqualified block wall.
M-59B	See field sketch. Large unit anchored by 12 - 7/8" CIP anchors (6 per side). Do anchorage analysis to obtain HCLPF. Anchors in 3 groupings per side that are 6" apart. A Evaporator is adjacent to unqualified block wall.
N-06	Located in tower cooling bldg in with 3 unqualified block walls. Also need to check 20' rule for vertical pumps. Anchorage adequate with 4 - 1" CIP anchors. Screen at 0.1g
P-5	Located in tower cooling bldg in with 3 unqualified block walls. Also need to check 20' rule for vertical pumps. Anchorage adequate with 4 - 1" CIP anchors. Screen at 0.1g
P-52A	REF: 1) CPCo Drwg. M-34, Sheet 10, Rev. 8 2) Pad and anchor bolt drwg. by D. E. Engle 6/22/94. (see document) 3) CPCo Doc. # EA-SP-03313-01, Sh. 19, Rev. 0. 4) CPCo Doc. # EA-SP-03311-01, Pg. 36, Rev. 0. Anchorage: Anchored w/ 4 ½" cast-in-place anchors and 4 5/8" cast-in-place anchors. Steel skid is grout filled per REF 2. Nozzle loads from large bore piping attached to these pumps appears to be about 15'. Nozzle loads (REFs 3 & 4) and see document for load combination.

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
P-52B	REF: 1) CPCo Drwg. M-34, Sheet 10, Rev. 8 2) CPCo Doc. # EA-SP-03313-01, Sh. 20, Rev. 0. 3) CPCo Doc. # EA-SP-03311-01, Pg. 37, Rev. 0. Anchorage: Same as P-52A. Anchored w/ 4 ½" cast-in-place anchors and 4 5/8" cast-in-place anchors. Nozzle loads have the order of magnitude of those in P-52A (REFs 2 & 3).
P-52C	REF: 1) CPCo Drwg. M-34, Sheet 10, Rev. 8 2) CPCo Doc. # EA-SP-03313-01, Sh. 21, Rev. 0. 3) CPCo Doc. # EA-SP-03311-01, Pg. 38, Rev. 0. Anchorage: Same as P-52A. Anchored w/ 4 ½" cast-in-place anchors and 4 5/8" cast-in-place anchors. Nozzle loads have the order of magnitude of those in P-52A (REFs 2 & 3).
P-9A	Check 20' casing length rule by drawing review. Drawing indicates lead cinch anchors for anchorage - therefore, outlier. Screen at 0.1g
PCV-2274	Securely mounted to what appears to be unmodified, unqualified block wall. Set HCLPF accordingly low.
PT-0510	Located on unqualified block wall and adjacent to unanchored rack. Screen at 0.1g
RV-2274	Securely mounted to what appears to be unmodified, unqualified block wall. Set HCLPF accordingly low.
T-77	Not seen - screen at marginal 0.1g value
T-82A-D	Not seen - screen at marginal 0.1g value
T-9C	Not seen - screen at marginal 0.1g value
T-24	Not seen - screen at marginal 0.1g value
T-13A&B	No positive longitudinal load path - screen at 0.1g value
T-3	Anchored on 4 angle legs anchored by 1 - 1" CIP anchor on each leg Get drawing for analysis purposes. Screen at 0.3g
T-40	Small fuel oil tank sitting on a 96" high block wall support enclosed in a small block building. The tank does not appear to be positively attached to the block wall. Outlier by inspection.
T-54	Not seen - screen at 0.1g

TABLE A Low Capacity Component Notes	
Equipment ID	Field Notes
T-58	REF: 1) CPCo. Drwg. #C-18, Sh.62, 8/20/68 , Sh. 65, 8/20/68 and Sh. 67, Rev. 6 (Tank General Configuration). 2) CPCo. Drwg. #C-38, Rev. 5 and #C-101, Rev. 2 (Tank Foundation and Anchor Bolt Schedule) The SIRW Tank contains water and is about 24 feet tall and 46 feet in diameter. Its normal maximum water level (250,000 gals) is 21 feet. The tank shell material is 5454-O aluminum. It is founded at floor slab located at Auxiliary Building Elevation 643'. See REFs 1 & 2 for tank configuration, bolt schedule and foundation. Anchorage: REF 2 shows the tank seated on 10" high concrete pad and anchored by 52 1-1/2" diameter 90 degree J-bolts. A continuous ring at top of anchor bolt chair is used. The J-bolt has depth of 10" and hook length of 3-1/4". CPCo. concrete detail drawings do not indicate any reinforcement rebars inside the concrete pad, nor a connection between the pad and the concrete floor. The J-bolt has 10" depth which is equal to the height of the pad. The J-bolt has not extended into the concrete floor. Outlier due to inadequately documented tank anchorage. For future reference, following calculation number will be used to address the seismic capacity of the tank. S&A Calculation # C-019 or CPCo engineering analysis # EA-POC0007899-T58.
T-81	Marginally anchored tank - screen at 0.1g
TC-0216	Mercoid switch - screen at marginal 0.1g value

ENCLOSURE 2

TABLE B,
IMPORTANCE MEASURES FOR SEISMIC COMPONENTS IN THE CUTSET SOLUTION

TABLE B

Importance Measures for Seismic Components in the Cutset Solution				
Event	Equipment ID	Equipment Description	FusVes	AchW
J1AVAB590	CV-0944A	SPENT FUEL POOL CLG ISOL.	2.54E-04	1.00E+00
J1AVAB590	CV-1045	P-69A/B SUCTION	2.54E-04	1.00E+00
J1AVAB602	CV-1037	CLEAN WASTE RECEIVER TANK	1.33E-03	1.00E+00
J1AVAB602	CV-1101	T-67 INLET VENT HEADER	1.33E-03	1.00E+00
J1AVAB602	CV-3001	CS HEADER ISOLATION	1.33E-03	1.00E+00
J1AVAB607	CV-0501	MAIN STEAM ISOLATION E-50B	4.15E-02	1.07E+00
J1AVAB607	CV-0510	MAIN STEAM ISOLATION E-50A	4.15E-02	1.07E+00
J1EC137	EC-137	P-41 ACTUATING PANEL	3.28E-01	1.07E+00
J1EC137	PS-5350	P-41 DISCHARGE PS	3.28E-01	1.07E+00
J1ERAB590	M-59A	A EVAPORATOR	2.07E-08	1.00E+00
J1HCAB590	HIC-0826	CCW HX SW OUTLET	2.43E-02	1.00E+00
J1HCAB590	HIC-0881	CCW HX SW OUTLET	2.43E-02	1.00E+00
J1HCAB590	HIC-0882	CCW HX SW OUTLET	2.43E-02	1.00E+00
J1PV2274	PCV-2274	NITROGEN FOR C-150	2.46E-03	1.00E+00
J1RV2274	RV-2274	NITROGEN SYSTEM RV	2.46E-03	1.00E+00
J1TKT24	T-24	P-9B DIESEL DAY TANK	6.23E-01	1.14E+00
J1TKT3	T-3	CCW SURGE TANK	2.43E-02	1.00E+00
J1TKT40	T-40	P-41 DIESEL DAY TANK	3.29E-01	1.07E+00
J1TREX13	EX-13	STATION POWER XFRMR 13	3.33E-02	1.03E+00
J3BSEB21	42-2139	MO-2139	1.87E-08	1.00E+00
J3BSEB21	52-2111	V-15A BREAKER	1.87E-08	1.00E+00
J3BSEB21	52-2113	HPSI VALVE MO-3081	1.87E-08	1.00E+00
J3BSEB21	EB-21	MCC 21	1.87E-08	1.00E+00
J3HEAB590	E-60A	SHUTDOWN COOLING HX	7.82E-03	1.01E+00
J3HEAB590	E-60B	SHUTDOWN COOLING HX	7.82E-03	1.01E+00
J3HSEY50	EY-50	BYPASS REGULATOR HS	9.73E-05	1.00E+00
J3MORB590	MO-3007	HPSI TO LOOP 1A	2.90E-02	1.31E+00
J3MORB590	MO-3009	HPSI TO LOOP 1B	2.90E-02	1.31E+00
J3MORB590	MO-3010	LPSI TO RX COOLANT LOOP 1B	2.90E-02	1.31E+00
J3MORB590	MO-3011	HPSI TO LOOP 2A	2.90E-02	1.31E+00
J3MORB590	MO-3012	LPSI TO RX COOLANT LOOP 2A	2.90E-02	1.31E+00
J3MORB590	MO-3013	HPSI TO LOOP 2B	2.90E-02	1.31E+00
J3MORB590	MO-3014	LPSI TO RX COOLANT LOOP 2B	2.90E-02	1.31E+00
J3MORB590	MO-3082	HPSI HOT LEG INJECTION	2.90E-02	1.31E+00
J3MORB590	MO-3083	HPSI HOT LEG INJECTION	2.90E-02	1.31E+00
J3RVAB570	RV-3057A	CV-3057 CLOSING AIR	3.56E-02	1.34E+00

TABLE B

Importance Measures for Seismic Components in the Cutset Solution				
Event	Equipment ID	Equipment Description	FusVes	AchW
J3RVAB570	RV-3057B	CV-3057 OPENING AIR	3.56E-02	1.34E+00
J3TKAB570	T-9A	HIGH PRESSURE CONTROL AIR	2.56E-02	2.05E+00
J3TKAB570	T-9B	HIGH PRESSURE CONTROL AIR	2.56E-02	2.05E+00
J3TKT10	T-10	DG FUEL OIL STORAGE TANK	5.64E-05	1.00E+00
J47HERB590	VHX-1	CONTAINMENT AIR COOLER 1	1.13E-02	1.60E+00
J47HERB590	VHX-2	CONTAINMENT AIR COOLER 2	1.13E-02	1.60E+00
J47HERB590	VHX-3	CONTAINMENT AIR COOLER 3	1.13E-02	1.60E+00
J47HERB590	VHX-4	CONTAINMENT AIR COOLER 4	1.13E-02	1.60E+00
J4PMAB590	P-52B	COMP. COOLING PUMP	2.54E-04	1.00E+00
J5PMP8A	P-8A	MOTOR DRIVEN AUX FEED PUMP	2.55E-03	1.33E+00
J5PMP8C	P-8C	MOTOR DRIVEN AUX FEED PUMP	4.92E-02	3.22E+00
J5PMSWS	P-7A	SERVICE WATER PUMP	1.53E-02	4.15E+00
J5PMSWS	P-7B	SERVICE WATER PUMP	1.53E-02	4.15E+00
J5PMSWS	P-7C	SERVICE WATER PUMP	1.53E-02	4.15E+00
J5RE127D1	127D-1	DG 1-1 UNDERVOLTAGE	1.43E-03	1.68E+00
J5RE127D2	127D-2	DG 1-2 UNDERVOLTAGE	2.53E-02	3.07E+00
J5TKT58	T-58	SAFETY INJ REFUEL WTR TK	1.98E-06	1.00E+00

ENCLOSURE 3

TABLE C,
SEISMIC BASIC EVENTS
REPRESENTING MORE THAN ONE COMPONENT

ENCLOSURE 3

TABLE C Seismic Basic Events Representing More Than One Component			
Seismic Event	Equipment ID	Equipment Description	HCLPF
J1AVAB590	CV-0944A	SPENT FUEL POOL CLG ISOL.	0.1
J1AVAB590	CV-1045	P-69A/B SUCTION	0.1
J1AVAB602	CV-1037	CLEAN WASTE RECEIVER TK RECIRC	0.1
J1AVAB602	CV-1101	T-67 INLET VENT HEADER	0.1
J1AVAB602	CV-3001	CS HEADER ISOLATION	0.1
J1AVAB607	CV-0501	MAIN STEAM ISOLATION E-50B	0.1
J1AVAB607	CV-0510	MAIN STEAM ISOLATION E-50A	0.1
J1BYTB590	ED-36A	P-9B BATTERY BANK 1	0.1
J1BYTB590	ED-36C	P-9B BATTERY BANK 2	0.1
J1BYTB590	ED-38A	P-41 BATTERY BANK 1	0.1
J1BYTB590	ED-38C	P-41 BATTERY BANK 2	0.1
J1CMAB590	C-50A	WASTE GAS COMPRESSOR	0.1
J1CMAB590	C-50B	WASTE GAS COMPRESSOR	0.1
J1EC137	EC-137	P-41 ACTUATING PANEL	0.1
J1EC137	PS-5350	P-41 DISCHARGE PS	0.1
J1HCAB590	HIC-0826	CCW HX SW OUTLET	0.1
J1HCAB590	HIC-0881	CCW HX SW OUTLET	0.1
J1HCAB590	HIC-0882	CCW HX SW OUTLET	0.1
J1TKAB590	T-13A	DG 1-1 JACKET WATER SURGE TANK	0.1
J1TKAB590	T-13B	DG 1-2 JACKET WATER SURGE TANK	0.1
J1TKSITS	T-82A	SAFETY INJECTION TANK	0.1
J1TKSITS	T-82B	SAFETY INJECTION TANK	0.1
J1TKSITS	T-82C	SAFETY INJECTION TANK	0.1
J1TKSITS	T-82D	SAFETY INJECTION TANK	0.1
J3AVAB602	CV-1064	T-64A/B/C/D VENT VALVE	0.3
J3AVAB602	CV-1102	T-67 INLET VENT HEADER	0.3
J3AVAB602	CV-1211	1A CONTAINMENT ISOLATION	0.3
J3AVAB602	CV-1814	V-46 DISCHARGE	0.3
J3BSEB21	42-2139	MO-2139	0.3

TABLE C Seismic Basic Events Representing More Than One Component			
Seismic Event	Equipment ID	Equipment Description	HCLPF
J3BSEB21	52-2111	V-15A BREAKER	0.3
J3BSEB21	52-2113	HPSI VALVE MO-3081	0.3
J3BSEB21	EB-21	MCC 21	0.3
J3BSEB22	42-2239	MO-3198	0.3
J3BSEB22	52-2213	HPSI VALVE MO-3082	0.3
J3BSEB22	EB-22	MCC 22	0.3
J3BSEB23	52-2313	HPSI VALVE MO-3083	0.3
J3BSEB23	EB-23	MCC 23	0.3
J3BSEB26	52-2625	MO-1043A BREAKER	0.3
J3BSEB26	EB-26	MCC 26	0.3
J3HESG	E-50A	'A' STEAM GENERATOR	0.3
J3HESG	E-50B	'B' STEAM GENERATOR	0.3
J3KVAB586	SV-1103	CONTAINMENT SUMP DRAIN	0.3
J3KVAB586	SV-1104	CONTAINMENT SUMP DRAIN	0.3
J3KVAB602	SV-1037	P-70 DISCHARGE ISOLATION	0.3
J3KVAB602	SV-1064	CLEAN WASTE RECEIVER TANK	0.3
J3KVAB602	SV-1065	CLEAN WASTE RECEIVER TANK	0.3
J3KVAB602	SV-1101	T-67 INLET	0.3
J3KVAB602	SV-1102	T-67 INLET	0.3
J3KVAB602	SV-1813	V-46 DISCHARGE	0.3
J3KVAB602	SV-1814	V-46 DISCHARGE	0.3
J3KVRB607	SV-2113	E-56 TO CHARGING LINE LOOP 1A	0.3
J3KVRB607	SV-2115	E-56 TO CHARGING LINE LOOP 1B	0.3
J3MORB590	MO-3007	HPSI TO LOOP 1A	0.3
J3MORB590	MO-3009	HPSI TO LOOP 1B	0.3
J3MORB590	MO-3010	LPSI TO RXTR COOLANT LOOP 1B	0.3
J3MORB590	MO-3011	HPSI TO LOOP 2A	0.3
J3MORB590	MO-3012	LPSI TO RXTR COOLANT LOOP 2A	0.3
J3MORB590	MO-3013	HPSI TO LOOP 2B	0.3
J3MORB590	MO-3014	LPSI TO RXTR COOLANT LOOP 2B	0.3
J3MORB590	MO-3082	HPSI HOT LEG INJECTION	0.3
J3MORB590	MO-3083	HPSI HOT LEG INJECTION	0.3
J3MORBSITS	MO-3041	SAFETY INJ TANK T-82A OUTLET ISOL	0.3

TABLE C Seismic Basic Events Representing More Than One Component			
Seismic Event	Equipment ID	Equipment Description	HCLPF
J3MORBSITS	MO-3045	SAFETY INJ TANK T-82B OUTLET ISOL	0.3
J3MORBSITS	MO-3049	SAFETY INJ TANK T-82C OUTLET ISOL	0.3
J3MORBSITS	MO-3052	SAFETY INJ TANK T-82D OUTLET ISOL	0.3
J3PMRCP	P-50A	PRIMARY COOLANT PUMP A	0.3
J3PMRCP	P-50B	PRIMARY COOLANT PUMP B	0.3
J3PMRCP	P-50C	PRIMARY COOLANT PUMP C	0.3
J3PMRCP	P-50D	PRIMARY COOLANT PUMP D	0.3
J3RVAB570	RV-3057A	CV-3057 CLOSING AIR	0.3
J3RVAB570	RV-3057B	CV-3057 OPENING AIR	0.3
J3TKAB570	T-9A	HIGH PRESSURE CONTROL AIR	0.3
J3TKAB570	T-9B	HIGH PRESSURE CONTROL AIR	0.3
J3HEAB570	E-60A	SHUTDOWN COOLING HX	0.32
J3HEAB570	E-60B	SHUTDOWN COOLING HX	0.32
J33PMAB570	P-67A	LPSI PUMP	0.33
J33PMAB570	P-67B	LPSI PUMP	0.33
J3HEAB590	E-54A	CCW HEAT EXCHANGER	0.33
J3HEAB590	E-54B	CCW HEAT EXCHANGER	0.33
J47HERB590	VHX-1	CONTAINMENT AIR COOLER 1	0.47
J47HERB590	VHX-2	CONTAINMENT AIR COOLER 2	0.47
J47HERB590	VHX-3	CONTAINMENT AIR COOLER 3	0.47
J47HERB590	VHX-4	CONTAINMENT AIR COOLER 4	0.47
J5PMSWS	P-7A	SERVICE WATER PUMP	0.52
J5PMSWS	P-7B	SERVICE WATER PUMP	0.52
J5PMSWS	P-7C	SERVICE WATER PUMP	0.52
J5HEE53AB	E-53A	SPENT FUEL POOL HX A	0.58
J5HEE53AB	E-53B	SPENT FUEL POOL HX B	0.58

ENCLOSURE 4

Stevenson & Associates Calculation Number C-003, Revision 1

Time History for IPEEE for Palisades Site

Client: Consumers Power Company Calculation No. C-003

Title: Time History for IPEEE for Palisades Site

Project: IPEEE - Seismic / A46

Method: Using SPECTRA for Response Spectrum to Time History Conversions.

Acceptance Criteria: SRP 3.7.1 and Engineering Practice

Remarks: Three statistically independent time histories are generated in Rev. 1.

REVISIONS

No.	Description	By	Date	Chk.	Date	App.	Date
0	Initial Issue	MSLi	11/11/93	TMT	11/14/93	[illegible]	11/15/93
1	Three Time Histories	MSLi	1/6/94	TMT	1/6/94	WD	1/17/94

S&A Stevenson and Associates

CALCULATION COVER SHEET

FIGURE 1.3

CONTRACT NO. 92C2750

S&A	JOB NO. 92C2750 SUBJECT: Consumer Power Co. IPEEE/A46	SHEET #1 OF 12
STEVENSON & ASSOCIATES a structural-mechanical consulting engineering firm	Time History for IPEEE C-003	Revision 1 By MSL 1/6/94 Chk. TMT 1/6/94

Table of Contents

1. Introduction

This calculation documents the generation of ground motion time histories for use in the Consumers Power Company IPEEE activity. The input motions will be used for the Reactor Building and Auxiliary Building at the Palisades Site.

The Revised Lawrence Livermore National Laboratory (LLNL) site specific ground response spectra (GRS) define in Ref. 1:

> **P. Sobel, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," Draft Report for Comment NUREG-1488, October 1993.**

is selected as the IPEEE free field horizontal input motion. Per the FSAR of Palisades Plant [4], the vertical ground motion is taken as two-thirds of the corresponding horizontal ground motion. Artificial time histories consistent with these RS are generated using S&A's SPECTRA program [7]. These time histories have been checked to be statistically independent.

2. IPEEE Input Motion Spectrum Shape

According to the requirement of [2], Section 3.1.1.2 Hazard Selection, the PRA should be performed using the higher of the mean (arithmetic) hazard estimates from Lawrence Livermore National Laboratory (LLNL) [1] and the Electric Power Research Institute (EPRI) [3] Since there is no seismic hazard curve provided for Palisades site in the EPRI report [3], the revised LLNL curves will be used in the subsequent analysis as the IPEEE free field input motion.

As explained in Appendix D, Question and Answer 7.47 of [2], the input ground response spectrum shape should be based on the median spectral shape for a 10,000-year return period. The spectral shapes of the revised LLNL hazard curve are listed in Appendix B of Ref. 1 and converted to acceleration units G as shown in the following table. The shape is 5 percent damped. To complete the spectral shape, the median PGA value of 0.06809G is interpolated from Ref. 1, the 50% data of Appendix A.

Freq. (Hz)	1	2.5	5	10	25	ZPA
Accel. (G)	0.02357	0.07302	0.10280	0.11721	0.11241	0.06809

Table 1 - Revised LLNL 10,000-year Response Spectra 50% Probability for the Palisades Site

Per the FSAR of Palisades Plant [4], the vertical ground motion is taken as two-thirds of the corresponding horizontal ground motion. The spectral shape is shown as the solid curve in Figure 1 and 2 with a ZPA level of 0.06809G for horizontal motion and in Figure 3 with a ZPA level of 0.04539G for vertical motion. The ZPA is assumed to start at 40 Hz. The target RS are saved in file RLLNLH.RS and RLLNLV.RS for horizontal and vertical motions, respectively..

S&A	JOB NO. 92C2750	SHEET #3
	SUBJECT: Consumer Power Co. IPEEE/A46	OF 12
STEVENSON & ASSOCIATES a structural-mechanical consulting engineering firm	Time History for IPEEE C-003	Revision 1 By NSL 1/6/94 Chk. TMT 1/6/94

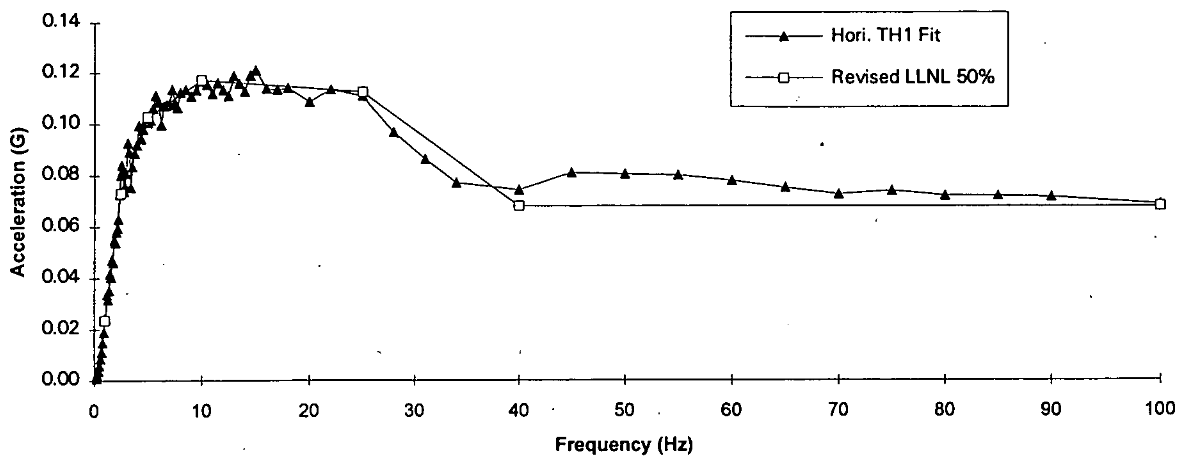


Figure 1 - IPEEE Response Spectrum and the First Time History Fit for the Palisades Site Based on 10,000-year Revised LLNL 50% Probability for Horizontal Motion

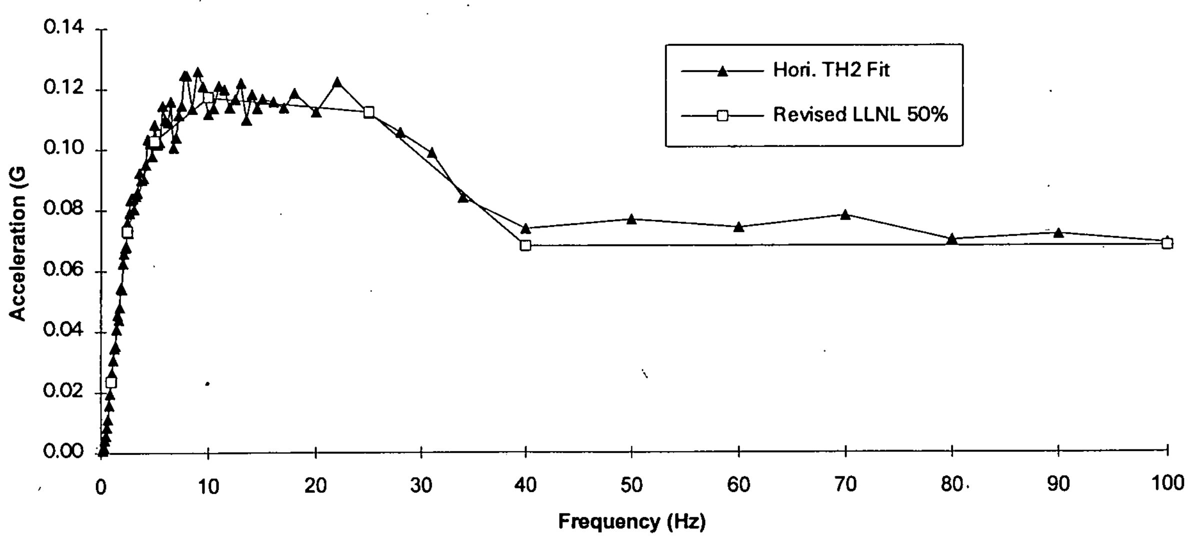


Figure 2 - IPEEE Response Spectrum and the Second Time History Fit for the Palisades Site Based on 10,000-year Revised LLNL 50% Probability for Horizontal Motion

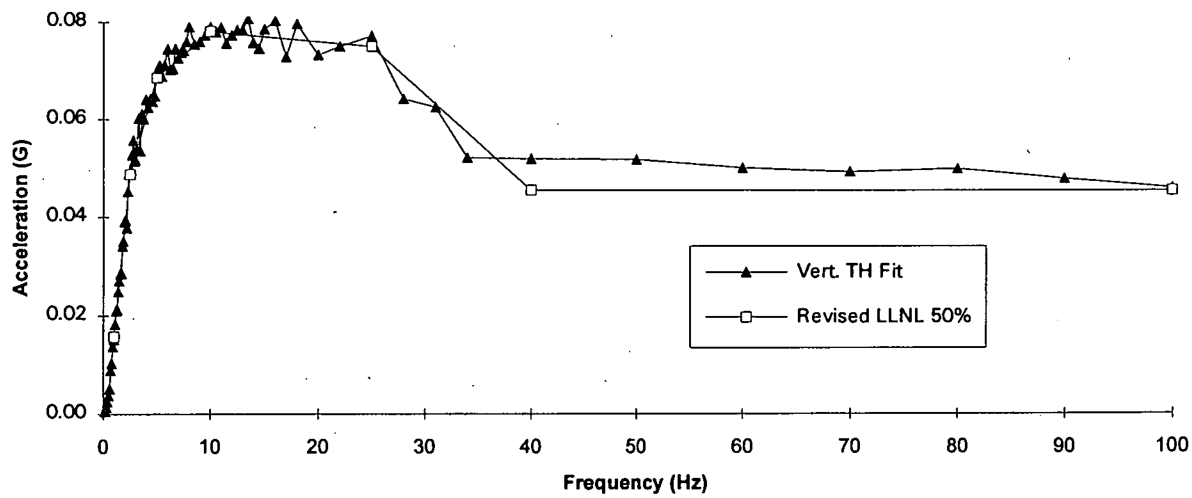


Figure 3 - IPEEE Response Spectrum and the Time History Fit for the Palisades Site Based on 10,000-year Revised LLNL 50% Probability for Vertical Motion

3. Conversion to Time History

Synthetic time histories are generated from the ground response spectrum using S&A's computer program SPECTRA [7]. The SPECTRA program is capable of converting between different forms of excitation, namely, time histories, response spectra, and power spectra.

First, the RS is imported to the SPECTRA program and stored in the SPECTRA database PASSI. Then SPECTRA is used to convert the response spectrum into time history. The conversion process is straightforward. The parameters used in the conversion are summarized as follows:

Duration	10.24 sec
Rise time	2 sec
Steady state duration	6 sec
Δt	0.01 sec
Random seed	324516 for the First Horizontal Time History
	452762 for the Second Horizontal Time History
	332165 for the Vertical Time History

The tri-linear envelope option in SPECTRA is used to generate synthetic time histories (TH), which include a rise time to a constant maximum steady state, and a final decay time equals (Duration - Rise Time - Steady State). The total and steady state duration of all time histories are with the range specified on page 3.7.1-5 of the SRP [5].

The SPECTRA program always synthesizes time histories that envelope the target RS conforming to the SRP 3.7.1 [5]. However, for the IPEEE, following the spirit of Seismic Margin Analysis (SMA) or PRA, the conservatism of enveloping is unnecessary. As a result, the TH is scaled down by trial and error to fit the required RS while maintaining the peak of the time history constant. It follows the following procedure

1. The time history is exported to file *U.TH
2. Run the SCALE1.EXE program with a scale factor. The SCALE1 program scales a TH by multiplying the TH by a scale factor but keeps the peak value unchanged. The scales TH file is *S.TH.
3. Import the scaled TH back into SPECTRA and convert to RS.
4. Plot the resulting RS against the target. The results verify itself.

The scale factors are 0.93 for the first horizontal time history, 0.95 for the second horizontal time history and 0.94 for the vertical time history. The time histories obtained by this process are shown in the Figure 4 to 6 for the first and second horizontal and vertical time histories, respectively. The fit of these RS to their target RS have been illustrated in Figure 1 to 3. Table 1 lists the original and scaled file names of time history.

File Name	First Horizontal Time History	Second Horizontal Time History	Vertical Time History
original	RLLNLH1U.TH	RLLNLH2U.TH	RLLNLVU.TH
Scaled	RLLNLH1S.TH	RLLNLH2S.TH	RLLNLVS.TH

Table 1 - List of File Name of original and Scaled Time Histories

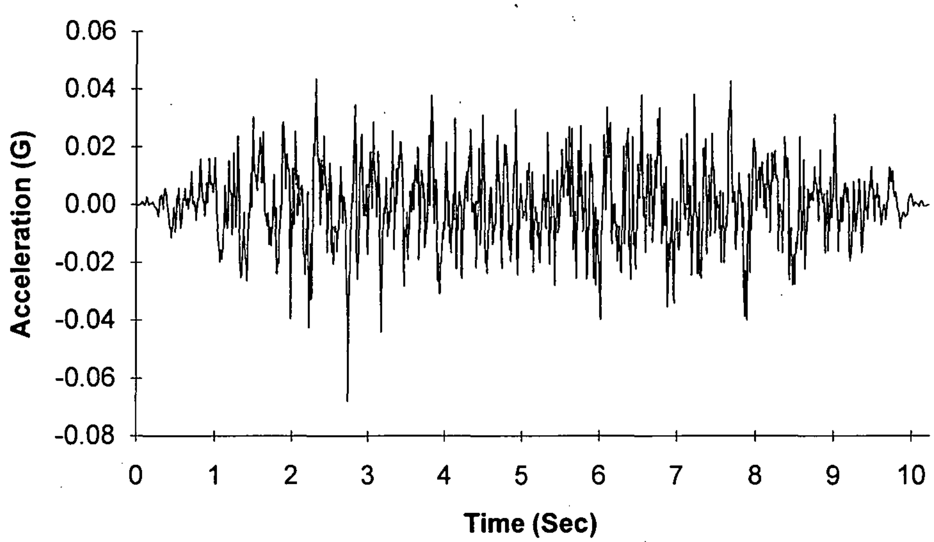


Figure 4 - IPEEE First Horizontal Ground Acceleration Time History for the Palisades Site Based on 10000-year Revised LLNL 50% Response Spectrum Shape

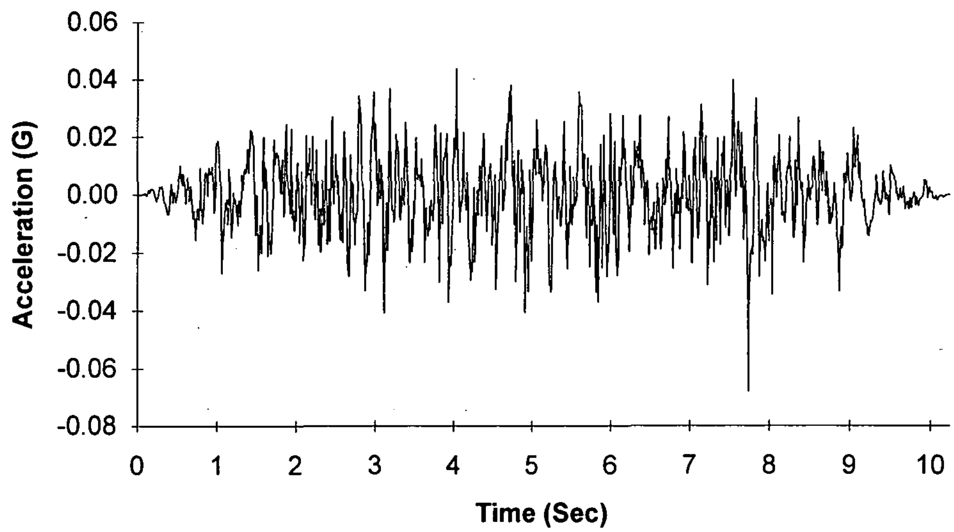


Figure 5 - IPEEE Second Horizontal Ground Acceleration Time History for the Palisades Site Based on 10000-year Revised LLNL 50% Response Spectrum Shape

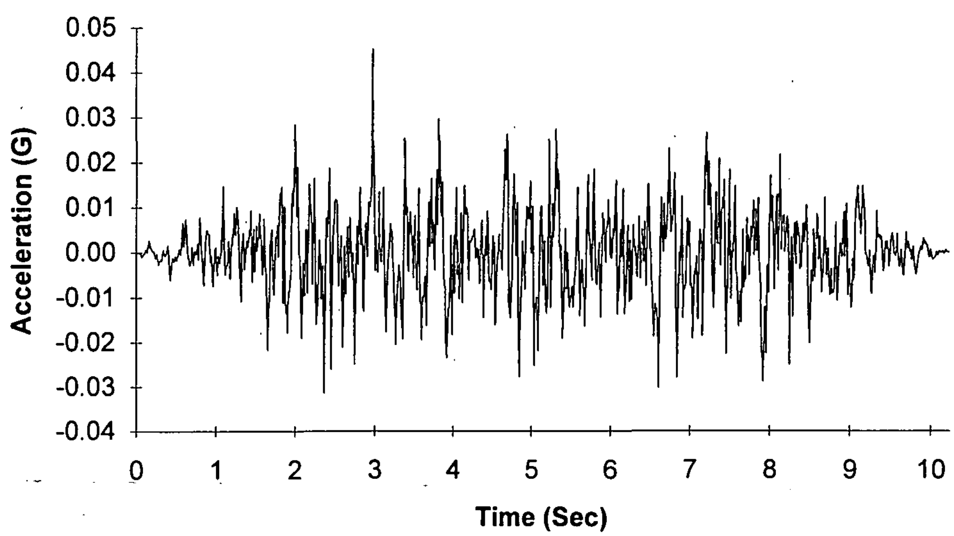


Figure 6 - IPEEE Vertical Ground Acceleration Time History for the Palisades Site Based on 10000-year Revised LLNL 50% Response Spectrum Shape

Page of 3.7.1-5 of the SRP [5] requires that time histories be statistically independent. Per ASCE standard [6], time histories whose normalized cross correlation function has a peak value below 0.3 will be considered statistically independent. Table 2 shows the maximum of the normalized cross correlation function for any two original time histories is below 0.3, which is acceptable. Utility program CORRL.EXE is used to compute correlation function for pairs of time histories. Source code of program CORRL.C and the input and output files, CORRLRL.LIS and CORRLRL.CHK, are stored in diskette. Further information about the program CORRL.EXE can be found in Ref. 8.

Time History	RLLNLH1S.TH	RLLNLH2S.TH	RLLNLVS.TH
RLLNLH1S.TH	1.000	0.157	0.135
RLLNLH2S.TH		1.000	0.161
RLLNLVS.TH			1.000

Table 12 - Maximum of Normalized Cross Correlation Function of Time Histories

Reference

1. P. Sobel, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," Draft Report for Comment NUREG-1488, October 1993.

2. U. S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", Final Report, NUREG-1407, June 1991.

3. Risk Engineering, "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," EPRI NP-6395-D, April 1989.

4. Consumers Power Company, "Final Safety Analysis Report Update of Palisades Plant", Rev. 15, April 1993.

5. U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Section 3.7.1, August 1989.

6. ASCE Standard, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety Related Nuclear Structures", ASCE 4-86, September 1986.

7. Stevenson & Associates, "SPECTRA - Users Manual," Version 1.1, 1992.

8. Stevenson & Associates, "Generation of Artificial Time Histories", Project No. 91C2690, Calculation C-001, Rev. 1, May 1992.

Appendix A -- List of Program SCALE1.BAS

```
DEFINT I-N
DIM th(4096)
PRINT "This program scales a TH (EDASP format) while keeping the ZPA independently to another value"
INPUT "Enter the input TH file name"; ti$
INPUT "Enter the output TH file name"; to$
OPEN ti$ FOR INPUT AS #1
INPUT #1, n, dt
FOR i = 1 TO n: INPUT #1, th(i): NEXT i
CLOSE 1
'Find the maximum of the TH
xmas = 0
k = 0
FOR i = 1 TO n
  IF ABS(th(i)) > xmas THEN
    xmas = ABS(th(i))
    k = i
  END IF
NEXT i
INPUT "Enter the scale factor"; scal
scal = ABS(scal)
PRINT "Enter the correct ZPA value ("; xmas; ")"; : INPUT zpa
IF zpa = 0 THEN zpa = xmas ELSE zpa = ABS(zpa)
FOR i = 1 TO n
  IF i = k THEN
    th(i) = SGN(th(i)) * zpa
  ELSE
    th(i) = th(i) * scal
  END IF
NEXT i
OPEN to$ FOR OUTPUT AS #1
PRINT #1, n; dt
FOR i = 1 TO n: PRINT #1, th(i): NEXT i
CLOSE 1
END
```

Job #92C2750
C-003 (Rev.1)
sheet 10/12

DESCRIPTION OF ANALYSIS: Response Spectrum to Time History Conversion.

COMPUTER CODE: SPECTRA VERSION: 1.1

RELEASE DATE: Nov. 92 AUTHOR/VENDOR: S&A

COMPUTER TYPE/SYSTEM: IBM Compatible

PROGRAM STATUS: ☐ Project Specific ☒ General Use/QA Approved

VERIFICATION/VALIDATION DOCUMENTATION: ☐ Attached ☒ On File

RUN NUMBER:

	ORIGINATOR	DATE	CHECKER	DATE
INPUT REPRODUCED ON LISTING	MSLi	1/6/94	TMT	1/6/94
MODEL VALID AND ASSUMPTIONS DOCUMENTED	N/A	—	—	—
PROGRAM APPROPRIATE AND ADEQUATE	MSLi	1/6/94	TMT	1/6/94
MODEL BEHAVES REASONABLE	MSLi	1/6/94	TMT	1/6/94
RESULTS PROPERLY INTERPRETED	MSLi	1/6/94	TMT	1/6/94

REMARKS:

S&A Stevenson and Associates	COMPUTER PROGRAM COVER SHEET FIGURE 2.8	CONTRACT NO. 92C2750

Job #92C2750
C-003 (Rev.1)
Sheet 11/12

DESCRIPTION OF ANALYSIS: Scale a TH by multiply the TH by a scale factor but keeps the Peak Value unchanged

COMPUTER CODE: SCALE1.EXE VERSION: 1.0

RELEASE DATE: Sept. 92 AUTHOR/VENDOR: S & A

COMPUTER TYPE/SYSTEM: IBM Compatible

PROGRAM STATUS: ☒ Project Specific ☐ General Use/QA Approved

VERIFICATION/VALIDATION DOCUMENTATION: ☒ Attached (see remarks) ☐ On File

RUN NUMBER:

	ORIGINATOR	DATE	CHECKER	DATE
INPUT REPRODUCED ON LISTING	MSLi	1/6/94	TMT	1/6/94
MODEL VALID AND ASSUMPTIONS DOCUMENTED	N/A	—	—	—
PROGRAM APPROPRIATE AND ADEQUATE	MSLi	1/6/94	TMT	1/6/94
MODEL BEHAVES REASONABLE	MSLi	1/6/94	TMT	1/6/94
RESULTS PROPERLY INTERPRETED	MSLi	1/6/94	TMT	1/6/94

REMARKS: Program was verified by line-by-line check.

S&A Stevenson and Associates	COMPUTER PROGRAM COVER SHEET FIGURE 2.8	CONTRACT NO. 92C2750

Job #92C2750
C-003 (Rev. 1)
sheet 12/12

DESCRIPTION OF ANALYSIS: Compute Correlation Function for pairs of Time Histories.

COMPUTER CODE: CORRL.EXE VERSION: 1.0

RELEASE DATE: Nov. 91 AUTHOR/VENDOR: S&A

COMPUTER TYPE/SYSTEM: IBM Compatible

PROGRAM STATUS: ☒ Project Specific ☐ General Use/QA Approved

VERIFICATION/VALIDATION DOCUMENTATION: ☐ Attached ☒ On File (see remarks)

RUN NUMBER:

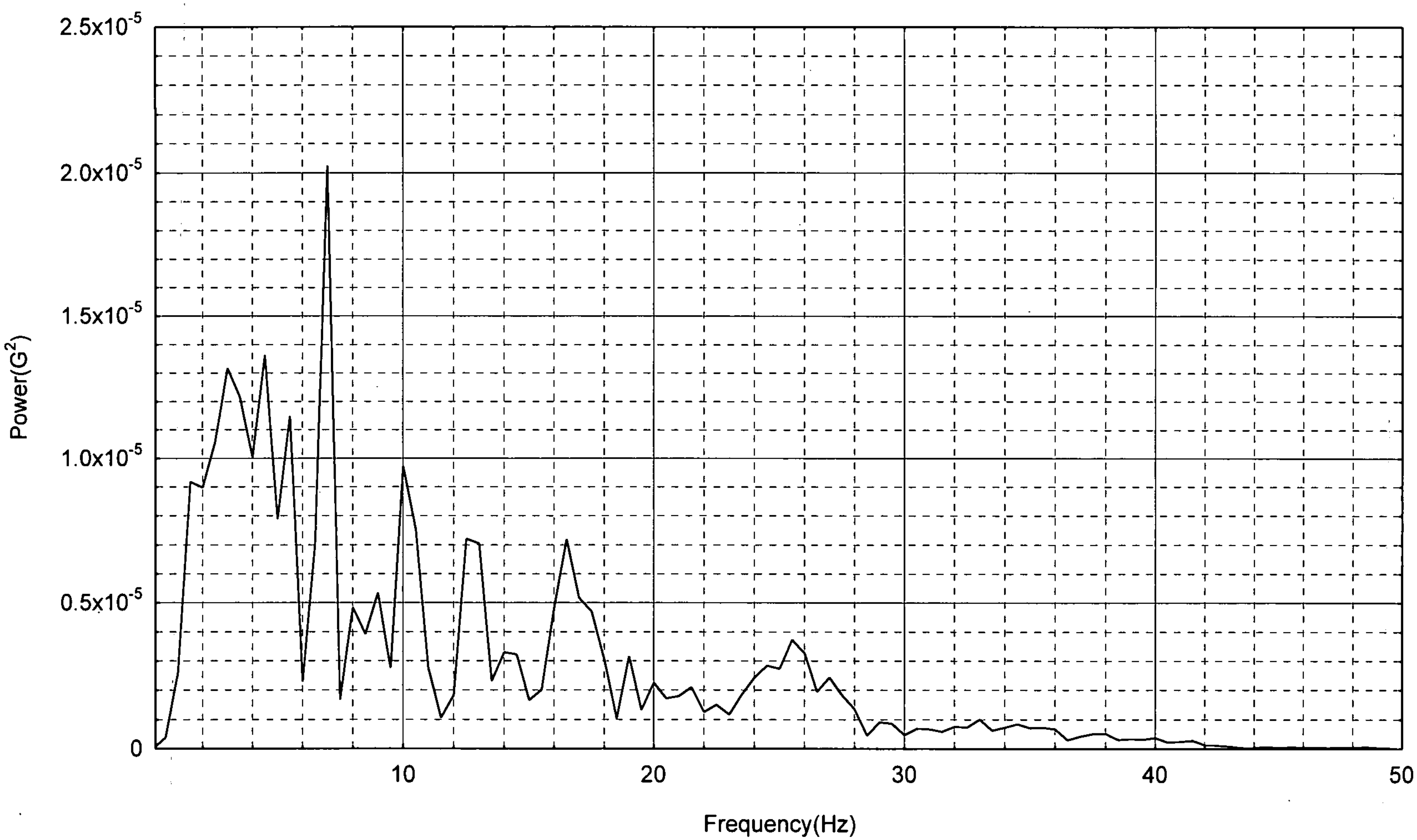
	ORIGINATOR	DATE	CHECKER	DATE
INPUT REPRODUCED ON LISTING	MSLi	1/6/94	TMT	1/6/94
MODEL VALID AND ASSUMPTIONS DOCUMENTED	N/A	—	—	—
PROGRAM APPROPRIATE AND ADEQUATE	MSLi	1/6/94	TMT	1/6/94
MODEL BEHAVES REASONABLE	MSLi	1/6/94	TMT	1/6/94
RESULTS PROPERLY INTERPRETED	MSLi	1/6/94	TMT	1/6/94

REMARKS: Program was verified in Project 91C2690, C-001. Rev. 1. (All conditions are the same as those specified in Project 91C2690)

S&A Stevenson and Associates	COMPUTER PROGRAM COVER SHEET FIGURE 2.8	CONTRACT NO. 92C2750

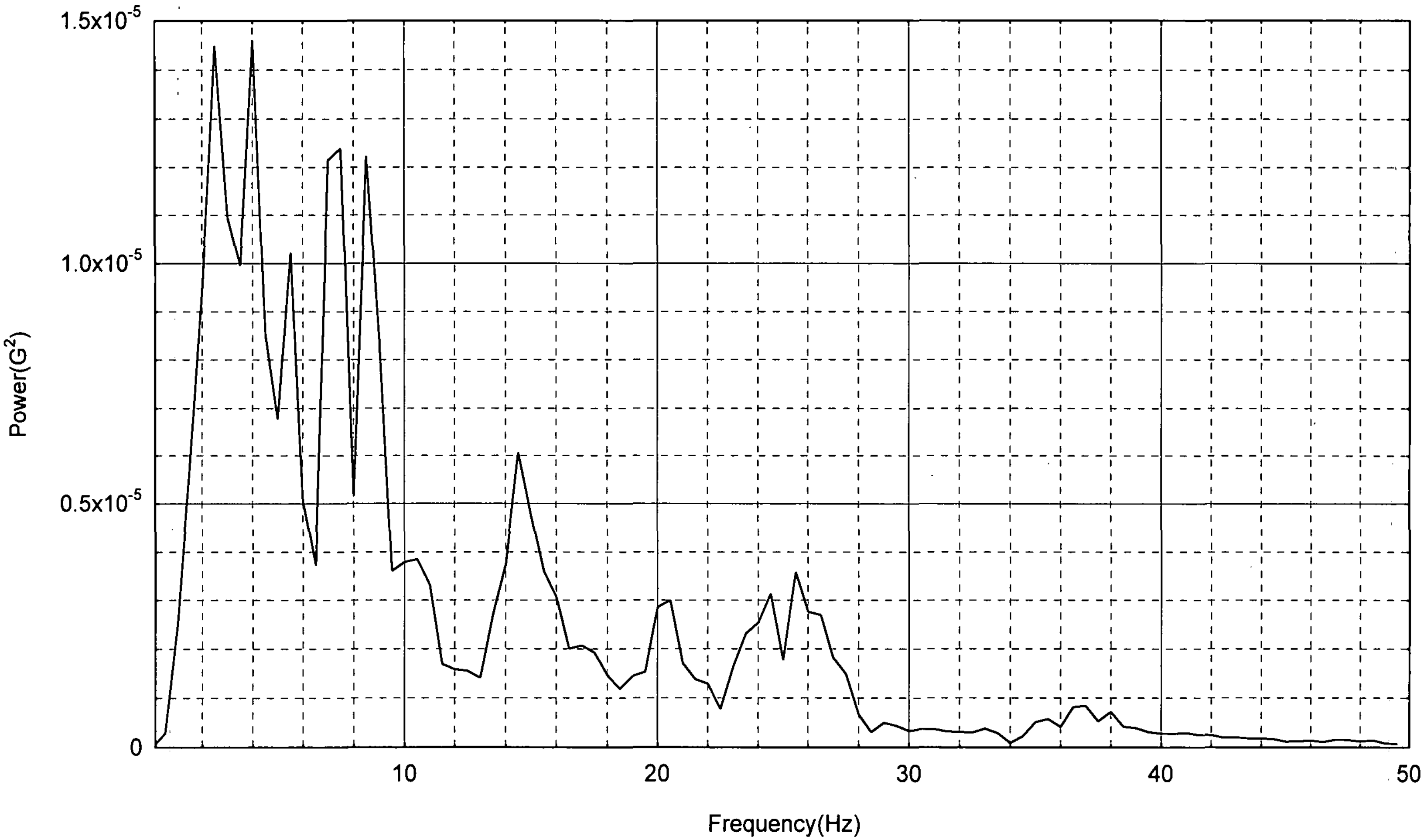
Consumers Power Company
Palisades Nuclear Plant
Power Spectrum of IPEEE Time History

BUILDING : OUTDOOR
ELEVATION : GROUND
DIRECTION : HORI1
MOTION : RLLNL50%S



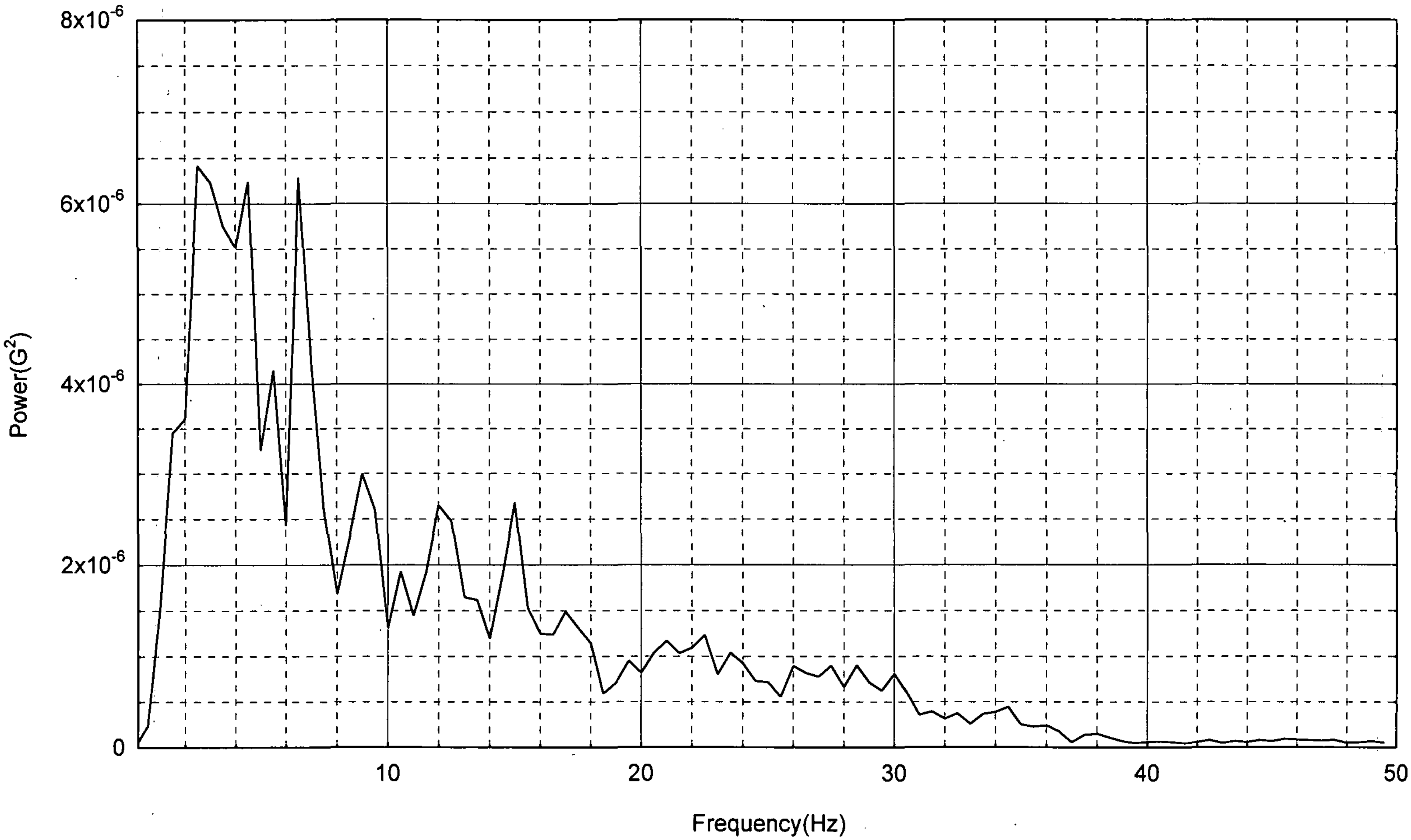
Consumers Power Company
Palisades Nuclear Plant
Power Spectrum of IPEEE Time History

BUILDING : OUTDOOR
ELEVATION : GROUND
DIRECTION : HORI2
MOTION : RLLNL50%S



Consumers Power Company
Palisades Nuclear Plant
Power Spectrum of IPEEE Time History

BUILDING : OUTDOOR
ELEVATION : GROUND
DIRECTION : VERT
MOTION : RLLNL50%S



ENCLOSURE 5

Supporting Documentation for Selected Components

Palisades IPEEE

Evaluation No.: EA-POC0007899-OSVS-T2

PALISADES NUCLEAR PLANT OUTLIER SEISMIC VERIFICATION SHEET (OSVS)		GIP Rev 2, Corrected 2/14/92 Sheet 1 of 1
ID : T-2 (Rev. 0)	Class : 21. Tanks and Heat Exchangers	
Description : CONDENSATE STORAGE TANK		
Building : TB		Floor El. : 590.00
Room, Row/Col : OUTSIDE		Base El. : 590.00

1. OUTLIER ISSUE DEFINITION - Tanks and Heat Exchangers

a. Identify all the screening guidelines which are not met. (Check more than one if several guidelines could not be satisfied.)

Shell Buckling	
Anchor Bolts and Embedment	
Anchorage Connections	
Flexibility of Attached Piping	
Other	X

b. Describe all the reasons for the outlier (i.e., if all the listed outlier issues were resolved, then the signatories would consider this item of equipment to be verified for seismic adequacy).

A ring-type foundation is used to support a large, flat-bottom, vertical tank.

2. PROPOSED METHOD OF OUTLIER RESOLUTION (Optional)

a. Defined proposed method(s) for resolving outlier.

b. Provide information needed to implement proposed method(s) for resolving outlier (e.g., estimate of fundamental frequency).

3. COMMENTS

4. CERTIFICATION:

The information on this OSVS is, to the best of our knowledge and belief, correct and accurate, and resolution of the outlier issues listed above will satisfy the requirements for this item of equipment to be verified for seismic adequacy:

Approved by: WJ [signature] Date: 4/24/95

[signature] 4/26/95

Owner's Review: Dale C Engle Date: 4/28/95

PALISADES NUCLEAR PLANT SCREENING EVALUATION WORK SHEET (SEWS)		GIP Rev 2, Corrected, 2/14/92 Status: No Sheet 1 of 4
ID : T-2 (Rev. 0)	Class : 21 - Tanks and Heat Exchangers	
Description : CONDENSATE STORAGE TANK		
Building : TB	Floor El. : 590.00	Room, Row/Col : OUTSIDE
Manufacturer, Model, Etc. :		

BASIS : External analysis

1. The buckling capacity of the shell of a large, flat-bottom, vertical tank is equal to or greater than the demand.	Unk
2. The capacity of the anchor bolts and their embedments is equal to or greater than the demand.	Unk
3. The capacity of connections between the anchor bolts and the tank shell is equal to or greater than the demand.	Unk
4. Attached piping has adequate flexibility to accommodate the motion of a large, flat-bottom, vertical tank.	Yes
5. A ring-type foundation is not used to support a large, flat-bottom, vertical tank.	No

IS EQUIPMENT SEISMICALLY ADEQUATE? **No**

COMMENTS

SRT: Djordjevic/Anagnostis Date: 7/7/93

REF 1: CPCo Drwg. # C-18-1-2, Rev. 1.
REF 2: CPCo Drwg. # C-18, Sh. 41, Rev. 3.
REF 3: CPCo Drwg. # C-37, Rev. 4.
REF 4: CPCo Drwg. # C-38, Rev. 5.

Photo: 5-14 shows typical anchor bolt.
Photo: 5-18 shows typical anchor bolt for T-81

Anchorage: 12 2" anchors in a 14 anchor pattern - 2 anchors not installed due to valve pit. See markup of REF 1 for bolt chairs. REFs 3 &4 indicate that the tank is on a ring foundation. See document for ultrasonic testing of bolt embedment.

Interaction hazard: Tank T-81 is about 3' away. Same height, somewhat smaller diameter, anchored with only 6 3/4" anchors.

Outlier due to a ring-type foundation used to support a large, flat bottom, vertical tank.

Preliminary analysis performed (S&A Calculation # C009, Rev. 0 or CPCo engineering analysis # EA-POC0007899-T2).
Major Assumptions:
1. Bolt tension allowable = 0.24 x Bolt capacity = 25.63 kips.
Minor Assumptions:
1. Height of fluid at the maximum level = 0.93 x tank shell height (actucal maximum level = 0.84 x tank shell height).

Revised analysis should be performed based on the new available information:

Evaluation No.: EA-POC0007899-SEWS-T2

PALISADES NUCLEAR PLANT SCREENING EVALUATION WORK SHEET (SEWS)		GIP Rev 2, Corrected, 2/14/92 Status: No Sheet 2 of 4
ID : T-2 (Rev. 0)	Class : 21 - Tanks and Heat Exchangers	
Description : CONDENSATE STORAGE TANK		
Building : TB	Floor El. : 590.00	Room, Row/Col : OUTSIDE
Manufacturer, Model, Etc. :		

1. Maximum water level.
2. Bolt embedment.
3. New soil horizontal and vertical stiffnesses.

Evaluated by: W [signature] Date: 4/24/95

[signature] 4/26/95

Owner's Review: Dale E Engle Date: 4/28/95

Attachment: Pictures
Attachment: T-2: Ultrasonic Testing of Anchor Bolts

PALISADES NUCLEAR PLANT SCREENING EVALUATION WORK SHEET (SEWS)		GIP Rev 2, Corrected, 2/14/92 Status: No Sheet 3 of 4
ID : T-2 (Rev. 0)	Class : 21 - Tanks and Heat Exchangers	
Description : CONDENSATE STORAGE TANK		
Building : TB	Floor El. : 590.00	Room, Row/Col : OUTSIDE
Manufacturer, Model, Etc. :		

PICTURES

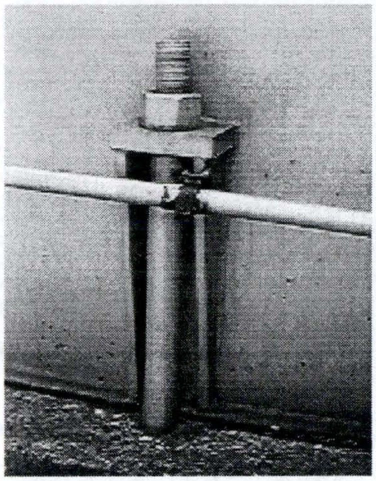


Figure 1: T-2 Bolt and Bolt Chair Detail

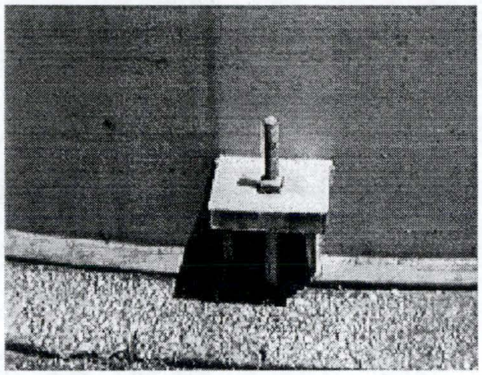


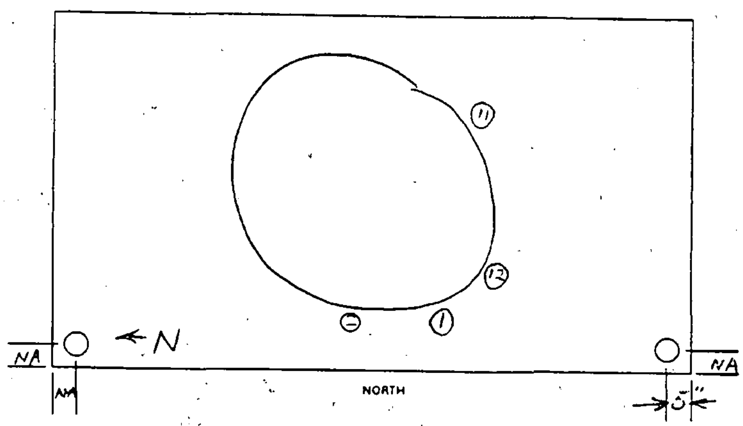
Figure 2: T-81 Bolt and Bolt Chair Detail

Evaluation No.: EA-POC0007899-SEWS-T2

PALISADES NUCLEAR PLANT SCREENING EVALUATION WORK SHEET (SEWS)		GIP Rev 2, Corrected, 2/14/92 Status: No Sheet 4 of 4
ID : T-2 (Rev.. 0)	Class : 21 - Tanks and Heat Exchangers	
Description : CONDENSATE STORAGE TANK		
Building : TB	Floor El. : 590.00	Room, Row/Col : OUTSIDE
Manufacturer, Model, Etc. :		

T-2: Ultrasonic Testing of Anchor Bolts

Ultrasonic testing of Anchor Bolts for the SQUG Program



Equip ID T-2 Number of Anchor bolts 12

A. Bolt Location	1	2	3	4	5	6	7	8	9	10	11	12
B. Bolt dia.	2"	2"	2"	2"	2"	2"	2"	2"	2"	2"	2"	2"
C. Total Bolt Length	41"	41"	40½"	41"	40½"	40½"	41"	41"	41"	41"	41"	41"
D. Bolt dist. above floor	24"	24"	24"	24"	24"	24"	24"	24"	24"	24"	24"	24"
E. embeddment length(c-d=e)	17"	17"	16½"	17"	16½"	16½"	17"	17	17	17	17	17"

Data taken by Dale Engle Date 1/17/95

Ultrasonic testing performed by Brian Lenius Date 1/17/95

Ultrasonic tester serial number 000855

T-2 - Condensate Storage tank