INDEPENDENT NRC STAFF FINAL SAFETY ASSESSMENT

OF

DRY SPENT FUEL CASK STORAGE FACILITY

AT

PALISADES NUCLEAR POWER PLANT SITE

By

Office of Nuclear Reactor Regulation

Office of Nuclear Materials Safety & Safeguards

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EXECUTIVE SUMMARY

This final safety assessment (FSA) contains the results of an NRC independent and detailed examination of the location of the foundation for the independent spent fuel storage installation (ISFSI) at Palisades nuclear power plant. In response to concerns raised by an NRC staff and a member of the public, NRC and its consultants independently examined the ability of the foundation to withstand seismic and certain other natural hazards, such as the effect of lake water run-up caused by wave action, the effect of soil erosion caused by high winds, and the potential of soil slumping from the surrounding sand dunes. The staff also reviewed the licensee's analysis and considered the relevant public comments, including the appropriate reports generated by the U.S. Army Corps of Engineers and the International Joint Commission.

The premise for the generic approval of the VSC-24 and other spent fuel casks is that their rugged design permits them to withstand a broad spectrum of climatic conditions and other possible challenges, so that they should be suitable for use at most currently licensed nuclear power reactor sites to provide safe interim storage for the spent fuel assemblies without affecting the safe operation of the plant, or the public health and safety and the common defense and security, and without adversely affecting the environment. The nuclear utility wishing to use the VSC-24, or any other generically approved cask, must perform an evaluation to ascertain that the climatic and site conditions fit within the parameters established for the cask.

The staff concluded that the location of the storage pad at the Palisades site is acceptable to support the concrete storage cask against effects of the design-basis earthquake for the site, and against other such postulated natural hazards as high winds and floods. Furthermore, the VSC-24 dry casks and the concrete storage pad will not pose any unacceptable risk to public health and safety because of substantial horizontal and vertical separation of the ISFSI from the lake shore, the installed shoreline protection, surveillance and monitoring programs established by the licensee, and the ability to take remedial measures to remove sand deposited by wind or water. The NRC staff also reviewed the licensee's analyses and determined that the licensee's conclusions are similar to those of the NRC.

Independent NRC Staff Final Safety Assessment of Dry Spent Fuel Cask Storage Facility at Palisades Nuclear Power Plant Site

INTRODUCTION

In May 1993, the U.S. Nuclear Regulatory Commission (NRC) issued a Certificate of Compliance, pursuant to 10 CFR Part 72, to Sierra Nuclear Corporation (SNC), approving VSC-24 cask to be used, for a 20-year term, by holders of 10 CFR Part 50 licenses. Subsequently, the Consumers Power Company (CPCo) started to use VSC-24 cask for dry spent fuel storage at the Palisades plant. Various questions were raised by members of the public and an NRC Region III staff concerning the possible effects of earthquakes and erosion at Palisades on the VSC-24 storage pad and the safe storage of spent fuel in the VSC-24 casks. In response to these concerns, in a letter to the NRC dated March 22, 1994, CPCo committed to perform analysis to demonstrate that the VSC-24 will be capable of performing its intended safety function under the effects of earthquake or erosion of the sand dunes by water and wind. Concurrently, NRC began an independent assessment to more closely examine the behavior of the VSC-24 pad under both normal and hypothetical accident conditions. On April 5, 1994, an NRC audit team (including NRC consultants) visited Palisades to inspect the VSC-24 casks, the storage pad, and the adjacent sand dunes. The team also reviewed the above mentioned licensee's analysis and recommended additional soil borings adjacent to the storage pad. Results from the new soil boring data and the updated analysis were reported in a letter to NRC dated May 12, 1994. The staff's preliminary findings were described in a Draft Safety Assessment (DSA) issued on May 18, 1994. Additional site visits by NRC staff were conducted on May 23 and June 20, 1994, to further assess the sand dunes and lake front at Palisades in light of the potential impact from earthquake and erosion. On May 23, 1994, a public meeting with CPCo was held in Benton Harbor, Michigan, to discuss the licensee's examination of Palisades dry cask storage pad and preliminary results from NRC's independent assessment. This meeting also provided a forum for the public to raise comments and questions regarding the Palisades dry cask storage foundation.

This final safety assessment (FSA) contains the results of the NRC independent and detailed examination of the location of the foundation for the independent spent fuel storage installation (ISFSI) at Palisades plant. In response to concerns more fully described below, the staff and its consultants (see Attachment 1) independently examined the ability of the foundation to withstand seismic and certain other natural hazards, such as the effect of lake water run-up caused by wave action, the effect of soil erosion caused by high winds, and the potential of soil slumping from the surrounding sand dunes. The staff also reviewed the licensee's analysis and considered the relevant public comments (see Attachments 2 and 3, respectively), including the appropriate reports generated by the U.S. Army Corps of Engineers and the International Joint Commission.

BACKGROUND

Regulatory Background

The regulations of the U.S. NRC (at 10 CFR Part 72, Subparts K and L) implement the regulatory process specified in the *Nuclear Waste Policy Act of 1982* (42 U.S.C. 10100 *et seq.*) by which the NRC performs rulemaking activities to approve spent fuel storage casks. Casks are approved for use under a range of environmental conditions sufficiently broad to encompass most of the nuclear power plant sites in the United States, without the need for further site-specific approval by NRC. The NRC general license, issued under 10 CFR Part 72 (Subpart K), permits approved casks to be used by any utility licensed to operate a nuclear power plant under NRC regulations (10 CFR Part 50). The general license requires the utility to verify that the various conditions expected at the specific reactor site (including earthquakes and tornadoes) fit within the set of hazards reviewed by NRC in approving the cask design. The utility must also confirm that it will satisfy the conditions for use specified by the NRC approval of the cask, which are binding on any utility using the cask at any reactor site.

On April 7, 1993, the NRC issued a final rule approving the VSC-24 storage cask (58 FR 17948), based on its detailed Safety Evaluation Report on the cask. In issuing the rule, NRC added the VSC-24 to the four cask designs it had approved for use under the general license. The rulemaking action authorized the VSC-24 cask for use under 10 CFR Part 72 at any reactor site where site-specific hazards are bounded by the approved cask design. In May 1993, NRC issued a Certificate of Compliance, pursuant to 10 CFR Part 72, to SNC, approving VSC-24 to be used, for a 20-year term, by holders of 10 CFR Part 50 licenses. The VSC-24 cask was first used under the general license by CPCo at the Palisades plant in Michigan.

CPCo holds an NRC license under 10 CFR Part 50, which authorizes it to operate the Palisades plant to generate power and to possess the spent fuel resulting from that operation. Before issuing the reactor operating license, the NRC staff prepared a comprehensive environmental impact statement on the Palisades site, including detailed consideration of the possible public health and safety consequences over the life of nuclear reactor operation.

Questions were raised about the possible effects of earthquakes and erosion at the Palisades site on the safe storage of spent fuel in the VSC-24 dry casks. In July 1993, a member of the public (Dr. Mary Sinclair) expressed concern that the concrete pad beneath the free-standing casks was built on "shifting dunes." In addition, a member of the NRC Region III technical staff, Dr. Ross Landsman, raised questions about the stability of the soil and slopes below the ISFSI pad under earthquake induced vibratory ground motion. On the basis of these concerns, in March 1994, NRC began an independent assessment to more closely examine the behavior of the pad at Palisades under normal conditions, under the long-term effects of erosion, and under conditions of a postulated earthquake that might cause the sand below or around the pad to move. Concurrently, CPCo committed to analyze in greater detail the stability of the cask pad and foundation. The NRC staff also conducted an independent

evaluation of the licensee's analysis. The results of that independent evaluation are documented here in this Final Safety Assessment.

In light of the questions raised concerning earthquakes and erosion, it should be helpful to summarize, in lay terms, how NRC requirements generally address such considerations. In particular, it may be useful to deal with the concern which has been expressed that a different level of safety analysis is required depending on whether a spent fuel storage cask is approved on a site-specific basis or generically, through rulemaking. As will be seen, the expected level of analysis and the resulting level of safety are the same, although the language of the regulations may, because the regulations are less specific with regard to general licensing, allow readers to draw the incorrect inference that less analysis is intended for generically licensed casks.

Turning first to the NRC rulemaking on the VSC-24 -- the cask used at Palisades by CPCo --, NRC based its approval of the VSC-24 on a safety evaluation, including a review to assure it was safe to use the cask in the event of the occurrence of a range of site conditions that could diminish cask safety. Further, under the NRC general license, a licensee must verify that reactor site parameters are within the envelope of site conditions reviewed by NRC for the cask approval. If potential conditions exist at the reactor site, including potential erosion or earthquakes, that could unacceptably diminish the cask's safety by any credible means, then it is clear that the licensee's analysis must include an evaluation of the potential conditions to verify that impairment of cask safety is highly unlikely.

The NRC's regulations do not explicitly require a licensee using a cask under a general license to evaluate the cask storage pad and foundation under such site conditions as erosion or earthquakes. But as explained above, if conditions at the reactor site could unacceptably diminish cask safety by, for example, affecting the stability of the supporting foundation so as to put the cask in an unsafe condition, then the cask may not be used unless the foundation is appropriately modified or a suitable location at the reactor site is found. Implicitly, therefore, the pad and the underlying foundation materials must be analyzed under site conditions that include erosion and earthquakes, notwithstanding that the pad has no direct safety function and that the cask is designed to retain its integrity even assuming the occurrence of a range of site conditions.

In some cask systems, the cask pad and foundation are designed as an integral part of the storage system to prevent impairment of cask safety. To date, if a licensee plans to use such a design, it must obtain an NRC site-specific license, and NRC will review the pad as a part of the cask system. In such situations, unlike the VSC-24, the cask storage pad is cited as "important to safety," and the NRC general design criteria therefore apply to it. NRC would not require that these criteria be applied, however, to the cask pad for the VSC-24, because the pad has no safety function to prevent or mitigate an accident or its consequences (as noted above).

Explicit NRC requirements for a site-specific license require the licensee to evaluate the cask pad site for liquefaction (or other soil instability) and to show that soil conditions are adequate for the cask under site conditions that

include erosion or earthquakes. As noted, the general license implicitly requires the same type of analysis in order to ensure that erosion or earthquakes could not affect the foundation in such a way as to unacceptably diminish cask safety.

In sum, general licensing and site-specific licensing require analogous evaluation to verify that reactor site parameters, including soil conditions, are acceptable.

As a related matter, it should be noted that the licensee has the responsibility under the general license to evaluate the match between reactor site parameters and the range of site conditions (i.e., the envelope) reviewed by NRC for an approved cask. Typically, the licensee will have a substantial amount of information already assembled in the Final Safety Analysis Report (FSAR) for the nuclear reactor. In addition, the envelope for the approved cask is identified in the NRC Safety Evaluation Report and Certificate of Compliance and in the cask vendor's Safety Analysis Report for the cask. Of course, the licensee should consider whether the envelope evaluated by NRC adequately encompasses the actual location of the cask at the reactor site. In addition, the licensee should consider whether there are any site conditions associated with the actual cask location which could affect cask design and which were not evaluated in the NRC safety evaluation for the cask.

Engineering Background

The paramount objective of 10 CFR Part 72 is to protect the public health and safety by providing for the safe confinement of the spent fuel and preventing the degradation of the fuel cladding. The following discussion illustrates how the objective is achieved in dry cask spent fuel storage.

In the discussion on the general license (Subpart K to 10 CFR Part 72) for the dry storage of spent fuel, the staff states that casks must contain the radioactive material in order to protect the public health and safety and environment under both normal and postulated accident conditions. The VSC-24 dry cask storage system at Palisades is a vertical cask system composed of a steel multi-assembly sealed basket (MSB) and a ventilated concrete cask (VCC). The welded MSB provides confinement and criticality control for the storage and transfer of irradiated fuel. The VCC provides radiation shielding while allowing the MSB and fuel to be cooled by natural convection during storage. The MSB consists of a steel cylinder shell with a thick shield plug and steel cover plates welded at each end. An internal fuel basket is designed to hold 24 spent fuel assemblies. The VCC is a reinforced-concrete cask in the shape of a hollow right circular cylinder. The VCC has four inlets at the bottom of the cask and four outlets at the top to allow air to circulate through the VCC. The air flow path is formed by (1) the air inlet ducts, (2) the gap between the MSB exterior and the VCC interior, and (3) the air outlet ducts. The internal cavity of the VCC and the inlets and outlets are lined with steel. After the spent fuel assemblies are loaded into the MSB, the MSB is seal welded, dried, filled with helium, and then structurally welded. After the loaded MSB is inserted into the VCC, a shield ring is placed over the MSB/VCC gap and the weather cover is installed. Ceramic tiles are placed between the bottom of the MSB and the steel liner of the VCC to prevent

corrosion. Dry cask storage systems are massive devices designed and analyzed to provide shielding from direct exposure to radiation, contain the spent fuel in a safe storage condition, and prevent unacceptable releases to the environment. They are designed to perform these tasks relying only on passive heat removal and containment systems without moving parts and with minimal reliance on human intervention to safely fulfill their function for the term of storage. The designs consider margins of safety and defense in depth for both normal and accident conditions to protect the public health and safety, maintain the common defense and security, and protect the environment.

The existing fuel cladding is an additional barrier to the release of radioactive material from the spent fuel stored in dry casks. The cladding is, therefore, protected by the cask system during storage so that cladding degradation will not pose operational safety problems in the future when the spent fuel is removed from interim storage. The casks maintain an inert atmosphere, which further reduces the possibility of corrosion of the fuel cladding and reaction with the fuel. Staff at Lawrence Livermore National Laboratory (LLNL) extensively studied fuel cladding integrity under dry storage conditions and the effects of damage mechanisms that could lead to the failure of fuel cladding (LLNL 1987).

NRC has established temperature criteria for fuel cladding during storage that ensure compliance with the regulatory requirement. These criteria were based on consideration of material behavior to predict the state of cladding integrity under normal dry storage conditions.

Elevated temperatures can reduce the strength of concrete used in storage casks. Therefore, NRC established normal and accident temperature criteria for concrete systems to ensure that the concrete can perform its safety function.

Monitoring programs enable the licensee to determine when to take corrective action to maintain safe storage conditions. Surveillance programs ensure adequate cooling and detect blocked vents long before heat and blockages affect the safe operation of the storage system. If the vent is blocked, the licensee can clear the blockage before fuel cladding or concrete design temperature criteria are exceeded, and the cask can continue in service. Should these fuel cladding and concrete temperature criteria be exceeded under extreme accident conditions, safety function would not be lost, nor would the health and safety of the public be immediately threatened. However, the cask's ability to continue to perform its safety function for the remainder of the storage period could be affected. In that case, further testing and analyses may be required to ascertain that for continued use, the cask could be repaired, or the cask could be taken out of service and replaced with a new one.

Unlike interim storage prescribed in 10 CFR Part 72, the in-ground disposal of radioactive material, whether high-level or low-level waste (HLW or LLW), must take into account the geologic, hydrologic, and geochemical characteristics of the site or region to isolate the radioactive waste from the accessible environment. Site criteria for in-ground disposal of radioactive wastes enable an applicant to choose an appropriate site, one with a combination of

favorable conditions that will be a natural barrier to retard or attenuate the migration of any leaked radioactive material over a long period to control releases within acceptable limits. The disposal period for LLW is on the order of 500 years and for HLW, greater than 10,000 years. On the other hand, for interim spent fuel storage, site characteristics are investigated and assessed under Part 72, not to determine their suitability as a barrier to release of radioactive material, but rather to determine the frequency and the severity of external natural and artificial events that could affect the safety of an ISFSI. Unlikely but credible severe events are considered to determine the safety of the storage cask design.

EVALUATION

Seismology

The design basis of safety features for each nuclear power plant must account for the effects of earthquake ground motion. The safe-shutdown earthquake (SSE) defines the maximum ground motion for which certain structures, systems, and components necessary for safe shutdown are designed to remain functional. Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," requires that the design bases for earthquakes be determined by evaluating the geologic and seismic history of the site and surrounding region. The largest earthquakes possible in the site region must be assessed. An evaluation is required to determine whether any active faults in the site region could generate earthquakes large enough to be significant to the earthquake design bases. Those earthquakes that cannot be correlated with particular geological structures must be assumed capable of occurring throughout regions containing similar geologic structures (tectonic provinces).

The Palisades plant is in the Michigan basin of the Central Stable Region (CSR) tectonic province, an area of low seismicity. There are no known capable faults in the site area. The largest historic seismologic events in the CSR tectonic province were

- the 1929 earthquake at Attica, New York, estimated magnitude 5.5, 660 kilometers (410 miles) from the Palisades site

- the 1937 earthquake at Anna, Ohio, estimated magnitude 5.0-5.3, 270 kilometers (170 miles) from the Palisades site

The magnitude is a method of stating the size of earthquakes. It is calculated from measurements on seismograms and is independent of the location of the observation. It is a measure of the energy released by the earthquake in the form of elastic waves. The Attica earthquake was associated with the Clarendon-Lindon tectonic structure. Therefore, the 1937 event at Anna, Ohio, was the largest earthquake in the CSR tectonic province that was not associated with a tectonic structure.

The closest known earthquake to the site was the magnitude 3.6 event of October 1, 1899, which had a reported location about 30 kilometers (18 miles)

from the site. No other earthquakes were reported to have occurred within 50 kilometers (31 miles) of the site. Figure 1 is a map of all known earthquakes within 320 kilometers (200 miles) of the Palisades site.

Geology

The Palisades plant site area is underlain by dune sand, glacial deposits consisting of dense till, and lake deposits. Bedrock (Mississippian Shale) is at an elevation of about 140 meters (450 feet).

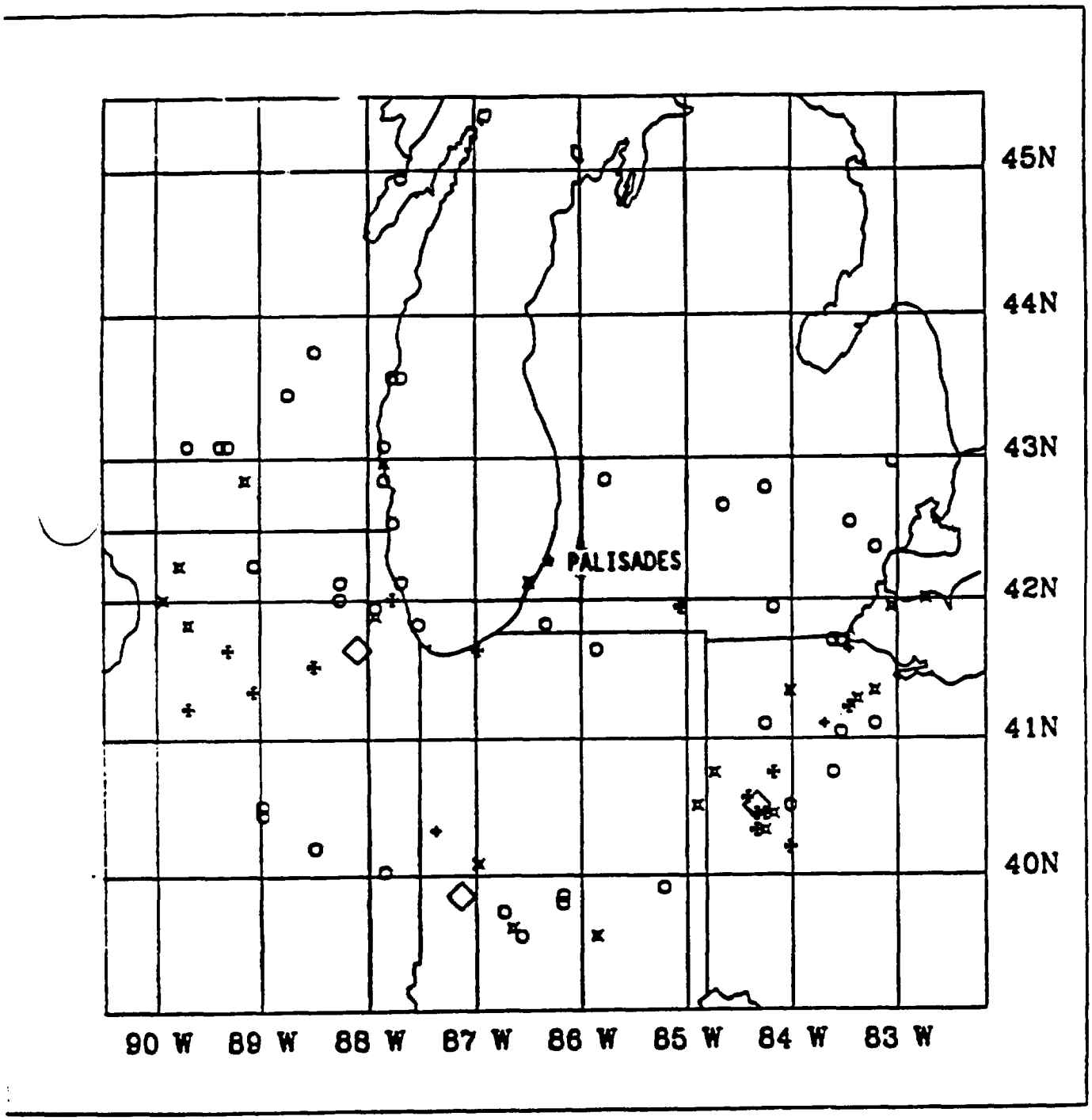
Vibratory Ground Motion

The Palisades plant was designed, constructed, and licensed before Appendix A to 10 CFR Part 100 and the Standard Review Plan (SRP) (NUREG-0800) took effect. During the licensing process for the Palisades plant, the Seismology Division of the U.S. Coast and Geodetic Survey (USC&GS), as advisors to the Atomic Energy Commission (AEC), reviewed the seismology for the plant. USC&GS concluded that, considering the site geology as sand dunes, the area will be subjected to an earthquake of Modified Mercalli intensity (MMI) VI with a peak ground acceleration of 0.1g during the lifetime of the plant, and that an earthquake of MMI VII with a peak ground acceleration of 0.2g (considering the effect of the sand dunes) might occur and should be considered the maximum potential earthquake. MMI VII corresponds approximately to the vibratory ground motion in the epicentral area of a magnitude 5.25 earthquake (LLNL 1981). In the Safety Evaluation Report for the Palisades construction permit review, the AEC staff noted that the site is in a region of low seismic activity and concurred with the USC&GS that the greatest ground motion acceleration during the life of the plant may be 0.1g, and that an acceleration of 0.2g should be the maximum potential earthquake. The staff confirmed this position in the Safety Evaluation Report for the operating license review.

Beginning in the late 1970s, the NRC, the successor to the AEC, conducted the Systematic Evaluation Program (SEP) to reevaluate the seismic design of 11 older nuclear power plants, including Palisades. Within the framework of the SEP study, the staff calculated a site-specific seismic design response spectrum for each plant site. Figure 2 is a plot of the Palisades site-specific SEP spectrum (NRC 1983a), the Palisades SSE Housner spectrum with a zero period acceleration of 0.2g, and the Regulatory Guide 1.60 spectrum with a zero period acceleration of 0.25g. The SEP spectrum is lower than the Palisades SSE spectrum in the frequency range of significance (above 1 Hertz) for nuclear power plant structures, systems, and components, and is significantly lower than the Regulatory Guide 1.60 spectrum, which is the design basis for the dry fuel storage casks at Palisades.

The NRC is funding LLNL to estimate the probabilistic seismic hazard for nuclear power plant sites east of the Rocky Mountains. In making certain safety decisions, the NRC considers the median probability of exceeding the SSE. The most recent LLNL probabilistic seismic estimates for Palisades indicate that the probability of exceeding the Housner design response spectrum with the high frequency anchor of 0.2g is on the order of once every 20,000 years. A more conservative estimate of the probability of exceeding the SSE is gained by comparing it to the mean probability of exceedance. Figure 3 shows certain results for the Palisades site from the most recent

First date: Aug 20, 1804 Last date: Dec 20, 1990



MAGNITUDES:

? ○ 1 □ 2 + 3 × 4 ✚ 5 ◇ * PALISADES PLANT SITE

Figure 1 Seismicity within 320 kilometers (200 miles) of Palisades

Response Spectra

RG 1.60/0.25g, Housner/0.2g, SEP Site Specific

5% damping

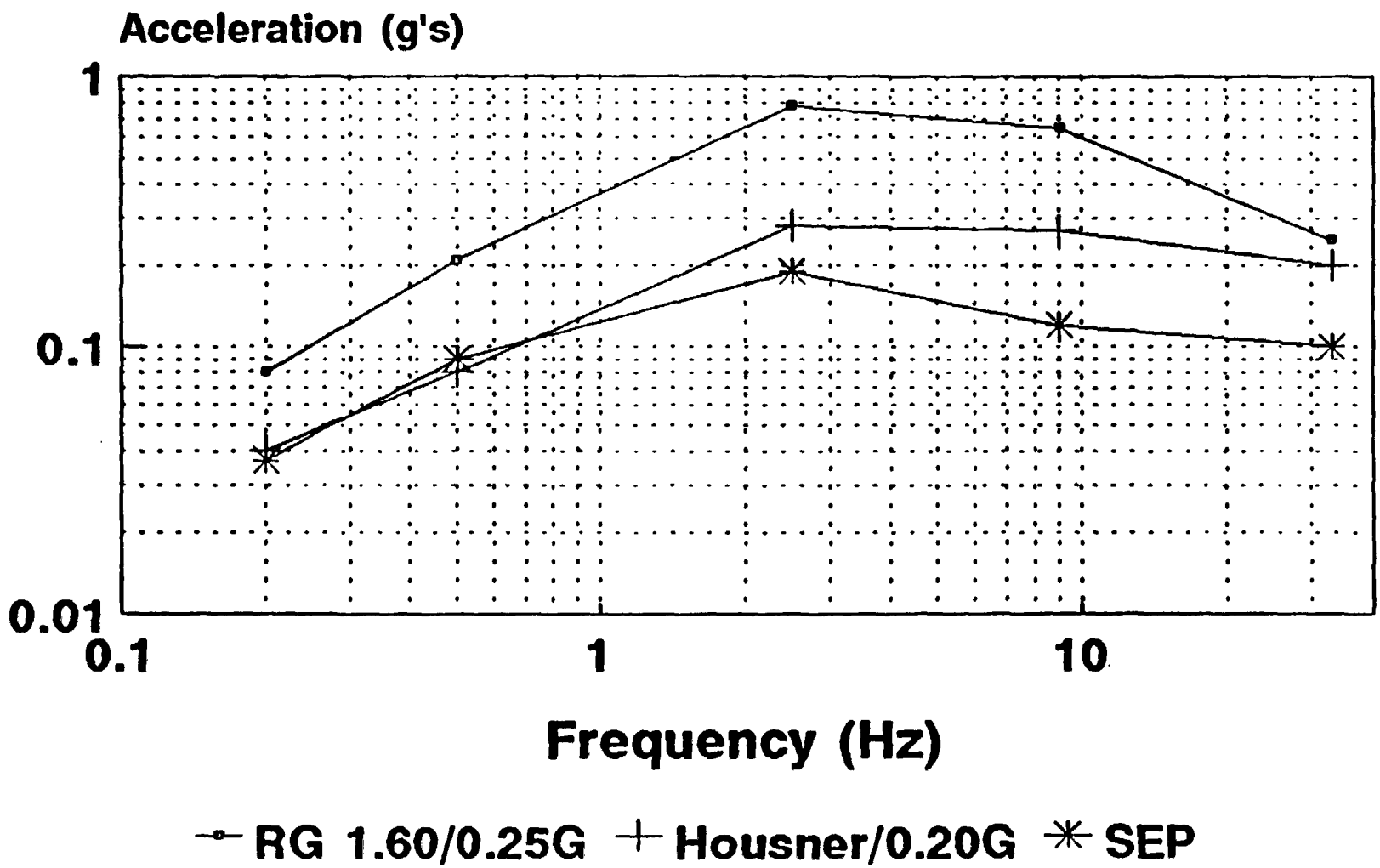
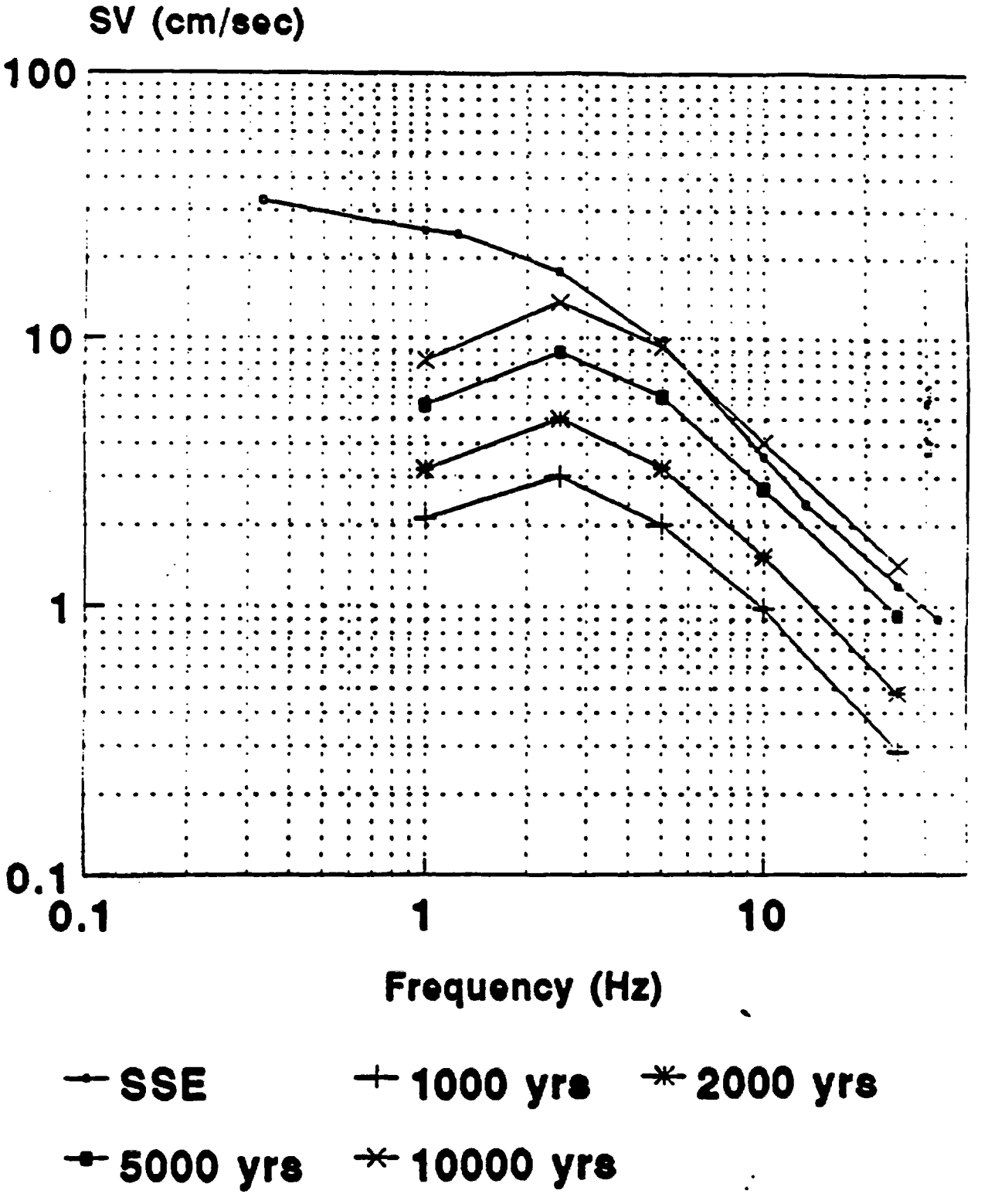


Figure 2 Cask design spectrum, Palisades SSE, and SEP spectrum

PALISADES

Horizontal Spectra - 5% Damping

SSE (0.20g - Housner) vs 1993 LLNL Mean Estimates



Soil - S2

Figure 3 Palisades SSE and Uniform Hazard Spectra

studies. This indicates that the SSE for the Palisades plant, which is a Housner response spectrum with a high-frequency anchor of 0.2g, is a conservative ground motion for the site since its return period ranges from 7500 years to greater than 10,000 years. In comparison, the SSEs for most of the eastern and central United States nuclear power plants are in the range of 1000 to 10,000 years.

Even though Palisades was licensed before Appendix A was issued, the NRC staff reevaluated the SSE ground motion for the Palisades site in accordance with the current siting regulation (10 CFR, Part 100, Appendix A) and the SRP. The staff considered the largest historic earthquakes reasonably associated with tectonic structures, the largest earthquakes in neighboring tectonic provinces on the boundary at the closest approach to the site, the largest historic earthquake in the CSR tectonic province whose epicenter cannot reasonably be correlated with tectonic structures as if this earthquake occurs near the site, and the geologic characteristics of the site.

The plant's original design ground motion conservatively represents the ground motion obtained from this reevaluation. The probability of exceeding this ground motion is low in comparison to other nuclear power plant sites in the United States. The SEP site-specific spectrum (NRC 1983a) is below the SSE spectrum at frequencies of significance to plant structures, systems, and components.

Stability of Subsurface Materials

The staff did an independent safety evaluation of the foundation soils below the ISFSI pad to determine how the stability of the surrounding slopes would be affected by liquefaction caused by SSE ground motion with a peak acceleration of 0.2g. Attachment 1 is a detailed technical evaluation of this subject prepared by the NRC consultant, Brookhaven National Laboratory (BNL). This independent evaluation is based on soil data submitted in a report by the licensee (Consumers Power Company) on May 12, 1994 (CPCo 1994b), and on information from several telephone conferences between the staff, its consultant, the licensee, and the licensee's consultant. The licensee's consultant performed analyses using a pseudostatic method, whereas the staff's consultant performed a dynamic finite element analysis and considered the time variation of the SSE ground motion. As stated in Attachment 1, the findings of this independent evaluation are essentially consistent with the licensee's analysis and calculations for the liquefaction of soils at the ISFSI site, the settlement of the ISFSI pad, and the stability of the slopes beside the pad (CPCo 1994b).

Liquefaction Analysis

In 1965 and 1966, the licensee made numerous borings at the Palisades plant area and found four distinct layers of soil: (1) dune sand, (2) dense to very dense gray silty sand, or sandy silt, (3) stiff gray clay, and (4) stiff to hard gray glacial till (CPCo 1992). Below the glacial till, bedrock begins at about elevation (El) 137.16 meters (450 feet) above mean sea level (MSL). Before the site was graded, the sand dunes rose steeply from El 177.39 meters (582 feet) MSL at the lake shore to about El 237.74 meters (780 feet) MSL at

the site of the reactor building. All heavy structures in the main plant area were constructed after excavating the loose dune sand above El 179.53 meters (589 feet) MSL. The reactor building foundation mat base is located at El 175.26 meters (575 feet) MSL.

The ISFSI pad is 59.44 meters (195 feet) long and 9.14 meters (30 feet) wide, and is placed with the longer dimension oriented east to west. The pad is about 0.61 meter (2 feet) thick at its center and about 0.91 meter (3 feet) thick near its edges. It was constructed in a valley between two sand dunes with its top at El 190.5 meters (625 feet) MSL, just north of the main plant, about 137.16 meters (450 feet) from the lake shore. In April 1994, the subsurface conditions below the pad were determined by two borings (B94-1 and B94-2) very close to the edge of the pad (CPCo 1994b). Both of these borings were taken down to about El 167.64 meters (550 feet), that is, to a depth of 22.25 meters (73 feet) from the ground elevation of about 189.89 meters (623 feet), and the condition (relative densities and strengths) of the subsurface soils was determined by standard penetration tests (SPTs). These borings revealed that the groundwater depth at the pad site was at about El 179.83 meters (590 feet) MSL, which is consistent with data from previous borings at the site and with the lake elevation.

The recent borings indicate that the soils at the ISFSI site consist of uniform dune sands to a depth of about 15.24 meters (50 feet) below the pad elevation and become finer at greater depths. As indicated by the SPT blow counts, the density of the dune sands below the pad vary. The dune sands from the ground surface down to the groundwater table (GWT) are loose to moderately dense with SPT blow counts (N), in the range of 10 to 20 blows per foot. At greater depths near and below the GWT, the sands generally increase to between moderately dense and dense with N values ranging from 20 to 40 blows per foot. However, some low blow count materials (about 13 to 16 blows per foot) are below the GWT. Since such sands are susceptible to liquefaction under earthquake loading, the licensee analyzed their potential for liquefaction at the design SSE ground motion with a peak ground acceleration of 0.2g. It is important to understand that these loose sands do not appear to be continuous throughout the site because borings made between the lake and the pad site did not reveal a soft saturated zone (see Attachment 1). The finer soils below the dune sands are very dense, with SPT blow counts exceeding 100 blows per foot, and are, therefore, of no concern for the stability of the foundation and slopes. Attachment 1 details the site conditions.

Liquefaction Potential

Attachment 1 describes the procedures and results of an evaluation of the liquefaction potential of the saturated loose sands below the ISFSI pad. This evaluation includes information from the new borings, B94-1 and B94-2. Data from boring B94-1 indicate that the safety factor against liquefaction of the sand below the water table is greater than unity (1.0) for the design earthquake ground motion applicable to the Palisades site. However, data from boring B94-2 indicate a safety factor near or below unity against liquefaction of the saturated sands at depths of 12.5 meters (41 feet) and 16.15 meters (53 feet), if subjected to the SSE ground motion. Therefore, it is conservatively assumed that the SSE ground motion would cause the softer soils at depths of

12.50 meters (41 feet) and 16.15 meters (53 feet) to liquefy. Although average values of SPT blow counts indicate that part of the area under the pad will not liquefy, it is assumed for the purpose of assessing the sensitivity of adjacent slopes to liquefaction of these soils, that the entire zone of soft soils below the pad would liquefy.

Stability of Slopes

Upon determining soil stability (as discussed in the preceding paragraph), the staff consultant assumed a potentially liquefiable zone of about 4.57 meters (15 feet) thick for the stability analyses using circular failure surfaces to postulate that a significant portion of the failure surface passes through the weakened zone. However, for the stability analysis based on a wedge-shaped sliding mass of the slope (Lambe & Whitman 1969), it is sufficient to assume only a thin weakened zone to obtain conservative estimates of safety factors against slope failures.

Stability of North-South (NS) Slopes

The staff consultant investigated the stability of the NS slopes by doing a quasi-static analysis of the slope subjected to a peak horizontal acceleration of 0.2g due to the design SSE. The failure surface is assumed circular and passes through the weakened liquefied zone and then through the toe of the slope, or beyond the toe. The driving force acting on the soil is the horizontal inertial force proportional to the seismic acceleration and the mass of the soil within the failure surface. The weight of the soil at the bottom of the slope, which must be moved to allow the failure of the slope, acts as a counterbalance to the driving force. The consultant found safety factors well below unity, indicating that the slope would move if the liquefied soil lost all original shear strength and extended uniformly to great distances from the toe of the slope. The consultant also did a detailed finite element dynamic analysis of this case and considered the time variation of ground motion (see Attachment 1, Appendix III).

Laboratory tests of sand samples and field observations of the behavior of slopes subjected to major earthquakes (such as the lower San Fernando dam section) have shown that liquefied sandy soils do not lose their strength completely, exhibiting residual shear strength. Such residual shear strength of sandy soils resists the sliding movement of the earth slopes and helps to maintain their stability. Stability analysis of the NS slope indicated that, to achieve safety factors averaging above unity under the action of the SSE ground motion, the liquefied soil must have a residual strength of about 38.3 kilopascals (kPa) (800 pounds per square foot (psf)). However, since the ISFSI pad is in a valley with two slopes on either side of the pad which are oriented north to south, there is a counterbalancing effect of the opposite slope on the stability of the deep failure surface. No credit was taken for such a beneficial effect of the opposite slope in the stability analyses of the NS slope. If this additional counterweight were included in the stability analysis, the residual soil strength required to prevent motion would be significantly reduced. Data from earlier slope failures caused by earthquakes, and laboratory soil strength tests, indicate the average value of residual shear strength available for clean sands (as those occurring at

Palisades), with SPT blow counts varying from 13 to 16 blows per foot, ranges from about 23.9 kPa (500 psf) to 38.3 kPa (800 psf). It is also not likely that the loose sand zone extends uniformly for a great distance under the pad, as was assumed in the preceding analyses. These factors would not likely allow the north-to-south deep failure surfaces to cause the slope to move (see Attachment 1).

Stability of East-West (EW) Slope

A gentle slope with a grade of about 1 in 10 is oriented from east to west from the pad to the lake. The staff consultant analyzed the stability of this slope, making assumptions similar to those for the NS orientation. The consultant indicated that residual strengths on the order of 19.2 kPa (400 psf) to 28.7 kPa (600 psf) are required to achieve a safety factor of unity. A simplified analysis considering a wedge-type failure mode confirmed this. The residual strengths available for these soils indicate that gross slope movement is not likely in the EW orientation. The logs of borings west of the pad indicate that the soft zone does not extend to the edge of lake water. This precludes a deep slope failure into the lake in the EW orientation.

Stability of the Slope in the Southwest (SW) Direction

The staff consultant investigated the possibility for failure of the slope which lies in a southwesterly direction from the pad, rises from an elevation of about 179.83 meters (590 feet) MSL to about 188.98 meters (620 feet) MSL at the top of the crest, and then gradually rises to the pad elevation (see Figure 4 in Attachment 1). This analysis indicates the slope could fail in a shallow manner when subjected to the SSE ground motion, allowing the toe of the slope to move about 3.05 meters (10 feet) toward the bottom, while the crest could move about 3.05 meters (10 feet) toward the pad, which is about 22.86 meters (75 feet) away from the crest of the slope. This small amount of slope movement would not impair the integrity of the pad.

The consultant also investigated a deep-seated failure of this SW slope that could result from liquefaction of the soil. The SW slope is even more stable than the slope in the EW direction. This is because the amount of soil mass that is acted upon by the earthquake to drive the SW slope is less than that in the uniform slope case considered in the EW direction. Also, there is a somewhat longer failure surface involved in the SW slope, which provides an increased amount of shear resistance as described in Attachment 1.

<u>Settlement Effects</u>

The settlements induced by liquefaction of the loose sands over a wide area are estimated to be on the order of 7.62 centimeters to 10.16 centimeters (3 to 4 inches). The effect of the differential settlement from liquefaction of the sandy zone on the 59.44-meter-long (195-foot-long) pad would be negligible.

The independent evaluation of the stability of the subsurface soils under the ISFSI pad and the slopes adjacent to the pad indicate that SSE ground motion at Palisades would not decrease the integrity of the pad.

Other Natural Hazards

The NRC staff studied the effects of erosion from natural phenomena on the stability and adequacy of the ISFSI. This evaluation included a review of previous NRC staff and licensee analyses of maximum water levels in Lake Michigan, an assessment of regional and local erosion along Lake Michigan, an assessment of the effects of wave action at the site, an evaluation of the effects of surface water runoff, and an evaluation of the potential for wind erosion to affect the ISFSI.

Lake Michigan Water Levels

The level of Lake Michigan fluctuates and is generally dependent on long-term excesses or deficiencies of precipitation and runoff. Information submitted by the licensee (CPCo 1992) indicates that the maximum monthly mean lake level occurred in 1886 at El 178 meters (583.7 feet) above MSL and that the lowest monthly mean level occurred in 1964 at El 176 meters (576.9 feet) MSL. Short-term variations and oscillations (seiches) occur occasionally and are caused by meteorological factors. The greatest short-term fluctuation on Lake Michigan of about 2.44 meters (8 feet) occurred at Montrose Harbor (Chicago) in 1954. Short-term variations along the eastern shore of Lake Michigan are rare and have not historically caused problems at the site.

During the Systematic Evaluation Program (SEP), the NRC staff estimated the probable maximum lake level at the plant site to be 181 meters (593.5 feet) MSL (NRC 1983a). Estimates developed by the licensee's consultant indicate that the maximum lake level would be approximately 179.5 meters (588.4 feet) MSL (CPCo 1994c). Plant systems, including the service water pump motors, are designed to withstand a level of 181.4 meters (594.7 feet) MSL.

The ISFSI facility is located approximately 137.16 meters (450 feet) from the shoreline of Lake Michigan. The pad is at El 190.2 meters (624 feet) MSL, approximately 9.15 meters (30 feet) above the probable maximum lake level. After reviewing the difference of maximum lake levels and the elevation of the facility, the staff concludes that water levels in Lake Michigan will have no effect on the ISFSI.

Regional and Local Erosion

The U.S. Army Corps of Engineers (COE) has been assigned the responsibility for conducting detailed site studies at various locations along the shoreline of Lake Michigan to determine the potential for damages caused by fluctuating water levels. These studies appeared in various reports; the report most applicable to the Palisades plant was completed for Berrien County (IJC 1993a). This report contains information about erosion and recession damages in this area. Although no report was prepared for Van Buren County, where Palisades is located, the northern boundary of Berrien County is only 8 miles south of the plant site, and the report is considered to be generally applicable to this area of the Lake Michigan shoreline.

The report indicates that shoreline erosion and bluff recession are very common along Lake Michigan and that some erosion can be expected to occur in

areas near the Palisades plant site. The report discusses various high-risk areas of erosion and indicates that the beach and dunes erode frequently and rapidly in many of these areas. The report also quantifies erosion rates in protected and unprotected areas and discusses areas that would be susceptible to erosion and damage.

In comparing the difference between the shoreline erosion at the plant and the erosion to the south of the plant (such as the Palisades Park and Covert Township Park areas to the south), the report discusses the effects of differing shoreline configurations and differing types of shoreline protection. These factors are important, because to the south, the dunes and bluffs are relatively steep and are located near the shoreline. These areas are also susceptible to sliding and slumping and are generally not protected by rock revetments. At the plant site, a heavily armored revetment is present; immediately north of the plant, the dunes are separated from the shoreline by a wide beach. Thus, the erosion rate at the plant is much lower than the rates for other areas near the site.

The licensee described the dune-beach system common to the area (CPCo 1994c). This submittal describes types of beach and dune configurations which are susceptible to erosion, based on their location relative to the active beach zone. To the south of the plant, houses and other structures have been damaged because they were constructed on or near dunes or shorelines which are located in the active beach zone. At the plant site, the ISFSI facility is located well away from the active beach zone. As discussed by the licensee, under severe conditions with a maximum water level of 181 meters (593.5 feet) MSL, the lake will be more than 30.5 meters (100 feet) away from the ISFSI. Therefore, the staff concludes that the ISFSI facility, by virtue of its location and elevation, will not be susceptible to the same storms which caused erosion in unprotected or exposed areas near the site. The staff further considers that erosion will continue to occur in various areas to the north and south along the shoreline of Lake Michigan; however, the published historic erosion rates for many of these exposed and unprotected areas may be misleading for certain shorelines and do not reflect conditions which are present in the immediate plant site area.

Wave Action

As discussed above, some wave damage and erosion can be expected to occur along the shoreline of Lake Michigan during major storms. Damage from severe wave action along the Lake Michigan shoreline is common, and some damage to the unprotected shoreline may occur in the immediate plant vicinity. However, since the probable maximum lake level is estimated to be approximately 181 meters (593.5 feet) MSL, no significant damage can be expected to occur at locations that are not adjacent to the shoreline or that are above this level. The staff concludes that most damage will be confined to the unprotected shoreline in the site area. The shoreline area immediately fronting the plant is protected by a rock revetment which is constructed of large rocks weighing from 1.8 to 5.4 metric tons (2 to 6 tons). The revetment varies in elevation and reaches a maximum elevation of 178.73 meters (586 ft) MSL at the northern end. The staff reviewed detailed drawings and analyses of the stability of the revetment submitted by the licensee (CPCo 1994c). Based on that review,

which included a review of water levels, breaking wave heights, and revetment configuration, the staff concludes that the revetment will substantially protect the shoreline from large storms. Wave action could erode the beach and, to a minor degree, the dunes in the area; however, damage to the dunes would be confined to areas in which the dunes are directly impacted by wave action. On the basis of staff experience and review of information in detailed site studies (IJC 1993a), it appears that the damage would be minimal and confined to an area northwest of the ISFSI facility below the maximum water level.

The ISFSI is located approximately 137.16 meters (450 feet) from the shoreline and about 9.15 meters (30 feet) above the estimated probable maximum lake level. The large dune to the north and west of the pad area will provide significant protection for the pad area and has a maximum elevation of about 214.42 meters (703 feet) MSL and a base elevation (at the pad area) of about 190.63 meters (625 feet) MSL. At the western end of this dune, the elevation at the base of the steeper west-facing dune slope is about 186.1 meters (610 feet) MSL.

On the basis of a review of wave action using U.S. Army Corps of Engineers design procedures (COE 1976; COE 1977), the staff concludes that any waves breaking at the shoreline would be largely dissipated in that area, and any wave runup directly toward the ISFSI would reach an elevation well below the pad elevation of 190.2 meters (624 feet) MSL. With a minimum elevation of 186.1 meters (610 feet) MSL, the dune that protects the ISFSI will not be damaged. Therefore, the staff concludes that the ISFSI facility will not be affected by wave action, even under the most critical and conservative conditions of lake level and wave run-up.

Surface Water Runoff

The licensee evaluated effects of surface water runoff in the ISFSI area (CPCo 1994b). The licensee indicates that erosion due to rainfall and runoff is of no concern because of the relatively coarse particles composing the dunes. In general, the staff disagrees with this conclusion. On the basis of the information submitted and a site visit to the ISFSI area, the staff concludes that some erosion of the face of the dunes and subsequent sedimentation of the eroded sand in the pad area could occur during heavy rainfalls. The total accumulation of sediment during any specific storm event is difficult to quantify, but it is possible that several feet of sediment could deposit in a localized area where a small gully could form. However, the staff agrees with the licensee that sedimentation in this area will not pose a problem. This conclusion is based on the monitoring program at the ISFSI and the frequency of inspections that will be conducted to detect the presence of deposited sediments. Specifically, the licensee indicates in Appendix A3 to EA-FC-864-46 (CPCo 1994b) that:

> Inspection of the concrete storage casks is done on a daily basis...to ensure the air vents of the cask are open, free of snow, sand, insects, etc...and also to monitor the temperature of the inlet and outlet air...these inspections monitor any accumulation of blowing sand on the storage pad. This

accumulation of sand would be considered a blockage of the air vent and would be reported for removal.

...The inspectors try to be alert to any other potential problems, and would report them to engineering. Situations of large buildups of sand near the pad, undermined or cracked foundations etc., would be clearly evident and would be immediately reported.

A second system of warning plant personnel of sand intruding into the storage pad area is the microwave system of the Intrusion Detection System. The microwave system monitors any movement at the perimeter of the cask storage area. It has been found in the past that any encroachment of sand in this monitored area will set off alarms and is required to be cleared away to keep this system operating properly.

Considering the licensee's ability to monitor, detect, and remove any accumulations of eroded sediments, the staff concludes that the ISFSI is adequately protected against erosion from the effects of rainfall and surface water runoff.

Wind Erosion

Wind erosion of sand dunes along Lake Michigan is very common. Less frequently, areas of wind erosion, termed "blowouts," occur and can cause the displacement of large quantities of sand. These problems have been addressed by the licensee (CPCo 1994a). The licensee consulted with local experts, who indicated that dune erosion in the site area is largely dependent on the establishment of a healthy vegetation cover on the dunes. The licensee stated that the dunes in the site area support dense vegetation, and stability is expected to be achieved in areas around the pad and any areas disturbed during construction of the ISFSI. Quantities of windblown sand should be relatively minor and sand should be deposited inland of the ISFSI.

The licensee also compared topographic maps and aerial surveys of the site area using data from 1965 and 1992 (CPCo 1994a). These maps indicate that the dunes in the area have changed very little during that 26-year period and that the dunes continue to maintain the same general slopes and elevations in 1992 as in 1965.

On the basis of its review of information submitted by the licensee, the staff concludes that some windblown sand is likely to be deposited in the ISFSI area. However, it should be emphasized that the safety function of the casks will not be lost, even if they are completely covered by windblown sand. As discussed above regarding surface water erosion, daily inspections should be sufficient to ensure that the deposited sands will be removed in a timely manner. Therefore, the staff concludes that wind erosion will not be a major problem and will not impair the safety of the ISFSI.

Effects of Erosion from Natural Phenomena

After reviewing the potential for erosion to occur from various natural phenomena, the staff concludes that the ISFSI is adequately protected. This conclusion is based on the following:

- The probable maximum level of Lake Michigan during major storm events is approximately 9.15 meters (30 feet) below the elevation of the ISFSI.

- Significant erosion will not occur at the ISFSI due to the presence of a heavily armored revetment fronting the site and the presence of a wide beach separating the dunes and the ISFSI from the shoreline. Shoreline erosion and dune recession, as reported in local and regional publications, is likely to continue at unprotected locations along the shoreline of Lake Michigan.

- Wave action during major storm events will not reach the ISFSI and will have no effect on its stability.

- Even though surface water runoff may erode some sand from the surrounding dunes, these sediments will not affect the functioning of the ISFSI because the sediments will be promptly detected during routine daily inspections and will be easily removed.

- Even though wind erosion may cause some windblown sands to be deposited in the pad area, the material will be detected during routine daily inspections and will be easily removed.

CONCLUSION

The premise for the generic approval of the VSC-24 and other spent fuel casks is that their rugged design permits them to withstand a broad spectrum of climatic conditions and other possible challenges, so that they should be suitable for use at most currently licensed nuclear power reactor sites to provide safe interim storage for the spent fuel assemblies without affecting the safe operation of the plant, or the public health and safety and the common defense and security, and without adversely affecting the environment. The nuclear utility wishing to use the VSC-24, or any other generically approved cask, must perform an evaluation to ascertain that the climatic and site conditions fit within the parameters established for the cask.

In the present case, concerns first introduced by an interested member of the public, Dr. Mary Sinclair, and additional issues raised by an NRC staff member, Dr. Ross Landsman, led the licensee, Consumers Power Company, to conduct additional studies of the stability of the reinforced concrete pad that supports the VSC-24 casks to evaluate both possible erosion and possible liquefaction of the sand underlying the pad in the event of an earthquake.

The NRC performed analyses independently from the utility's studies to ensure that the licensee had considered potential adverse circumstances that could affect the safe storage of the spent fuel as well as the impact of the ISFSI

on the safe operation of the plant. The staff concluded that the location of the storage pad at the Palisades site is acceptable to support the concrete storage cask against the effects of the design-basis earthquake for the site, and against other such postulated natural hazards as high winds and floods. Furthermore, the VSC-24 dry casks and the concrete storage pad will not pose any unacceptable risk to public health and safety because of substantial horizontal and vertical separation of the ISFSI from the lake shore, the installed shoreline protection, surveillance and monitoring programs established by the licensee, and the ability to take remedial measures to remove sand deposited by wind or water. The NRC staff also reviewed the licensee's analyses and determined that the licensee's conclusions are similar to those of the NRC.

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ATTACHMENT 1

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DETERMINISTIC EVALUATION OF LIQUEFACTION POTENTIAL OF FOUNDATION SOILS AND ITS EFFECTS AT THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) PAD AT PALISADES NUCLEAR POWER PLANT

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Task No. 1, JCN J-2042

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INTRODUCTION

This report presents the results of an evaluation of the impact of seismic effects on the potential liquefaction of the soils beneath the reinforced concrete spent fuel storage pad located at the Palisades Nuclear Power Plant as well as the impact of this liquefaction on the stability of the surrounding slopes. The data used for this evaluation was obtained at the site visit conducted on April 5th, 1994, together with additional information provided by the Licensee since that visit.

An additional site visit was conducted on May 23rd, 1994 to finalize conclusions reached and attend a public meeting on the issues with the NRC Staff. It should be noted that in addition to the independent calculations performed for this study, a review was made by Brookhaven National Laboratory (BNL) of the calculations performed by the Licensee and their Engineering Consultant, Sargent and Lundy. This review was limited to the liquefaction settlement and slope stability issues discussed in this report. Their conclusions and evaluations are essentially consistent with those presented herein.

I. DESIGN SEISMIC MOTIONS

The SSE ground motion for which this evaluation was performed is defined by the Housner spectrum scaled to a peak ground acceleration (PGA) of 0.2 g. A plot of the response spectrum associated with the Housner spectrum is shown in Fig. 1. For this study, an artificial time history was developed appropriate for this spectrum and is shown in Fig. 2. The time history generated to envelope the Housner spectrum has a duration of about 10 seconds and is considered to be associated with an earthquake of Richter magnitude 5.25 (M5.25) to M5.5. The comparison of the spectra from the artificially generated ground motion with the corresponding target spectra is shown in Fig. 3. As may be noted, the spectrum from the artificially generated ground motion exceeds the target Housner spectrum and therefore leads to conservative estimates of the computed site responses. In other words, the seismic motions are bounded by the current study.

II. SITE CONFIGURATION

The spent fuel pad is located just north of the plant site on the surface of an excavated zone within the sand dunes. The pad is at an elevation of about 625 feet (ft) mean sea level (MSL). The pad is approximately 30 ft wide and 195 ft long and is oriented with the long dimension in the East-West direction. The pad is about 2 ft thick at its center and is thickened near its edges to about 3 ft. The pad is about 400 to 500 ft east of the water's edge at Lake Michigan. The Plant grade is approximately at elevation 590 MSL while the Lake elevation is about 585 MSL.

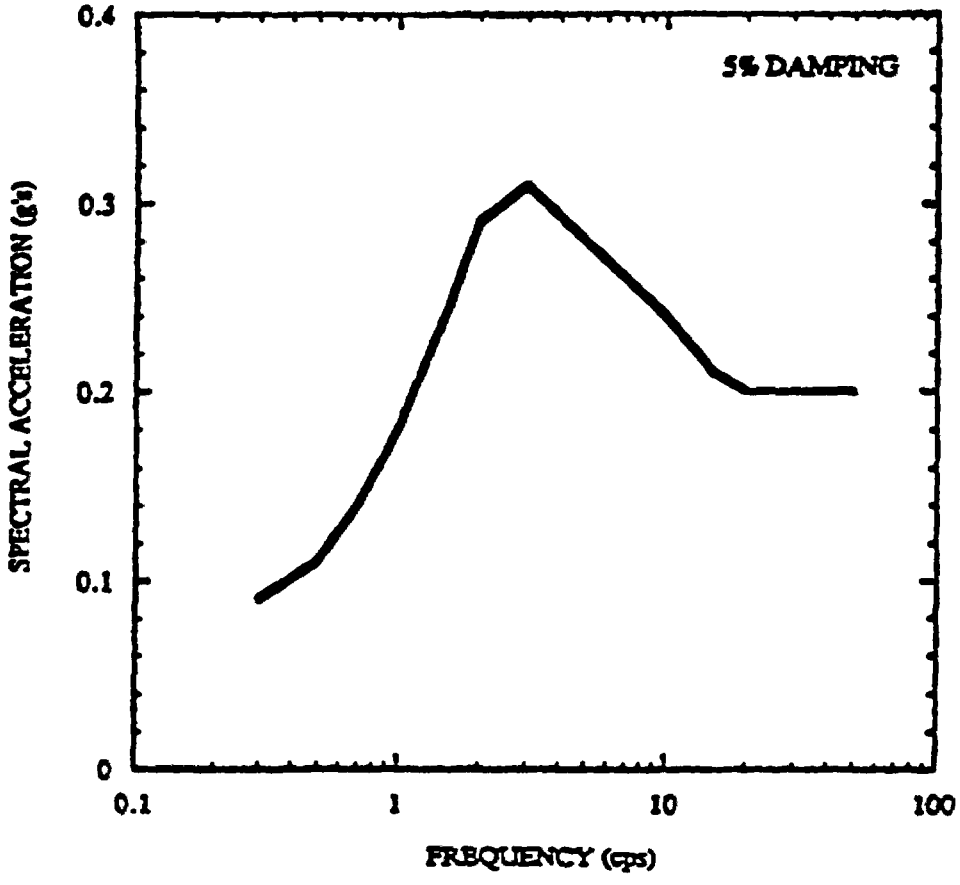

FIGURE 1: HOUSNER TARGET RESPONSE SPECTRUM
PGA = 0.2g

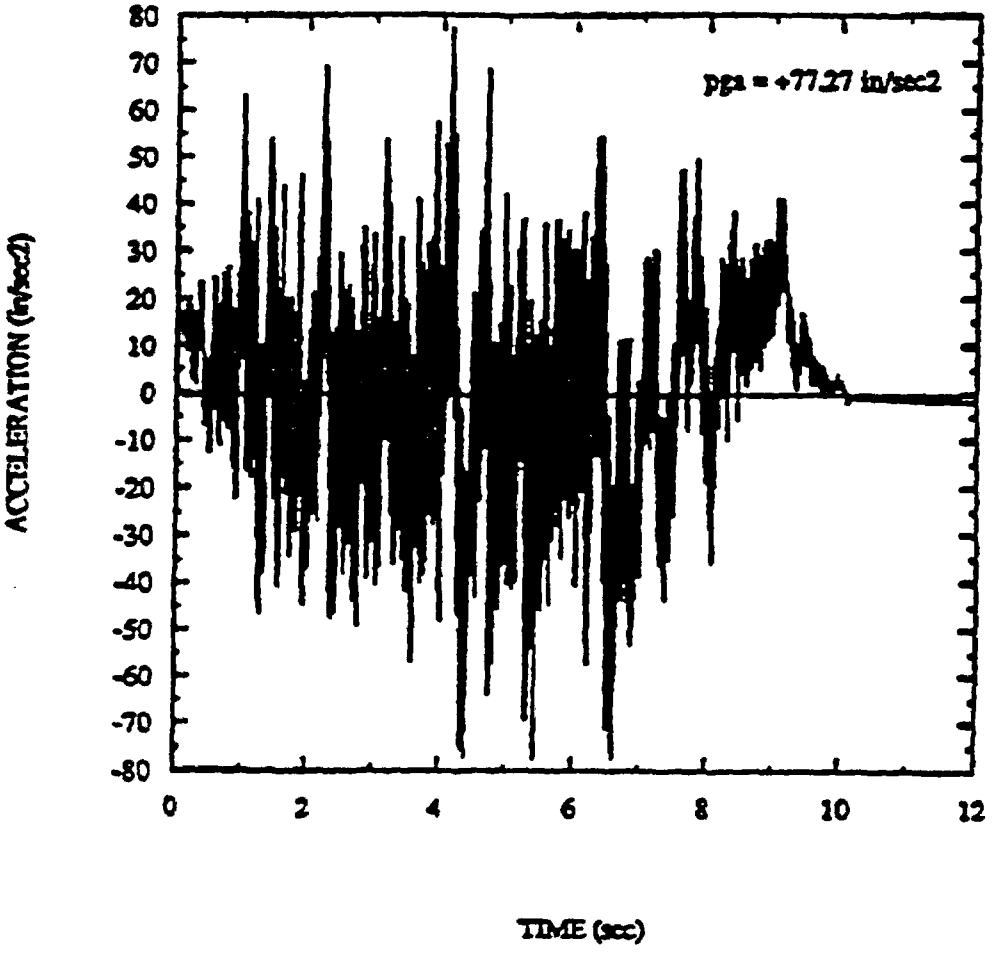

FIGURE 2: HOUSNER TIME HISTORY (magnitude 5.5)
using 4,096 TIME HISTORY RECORDS

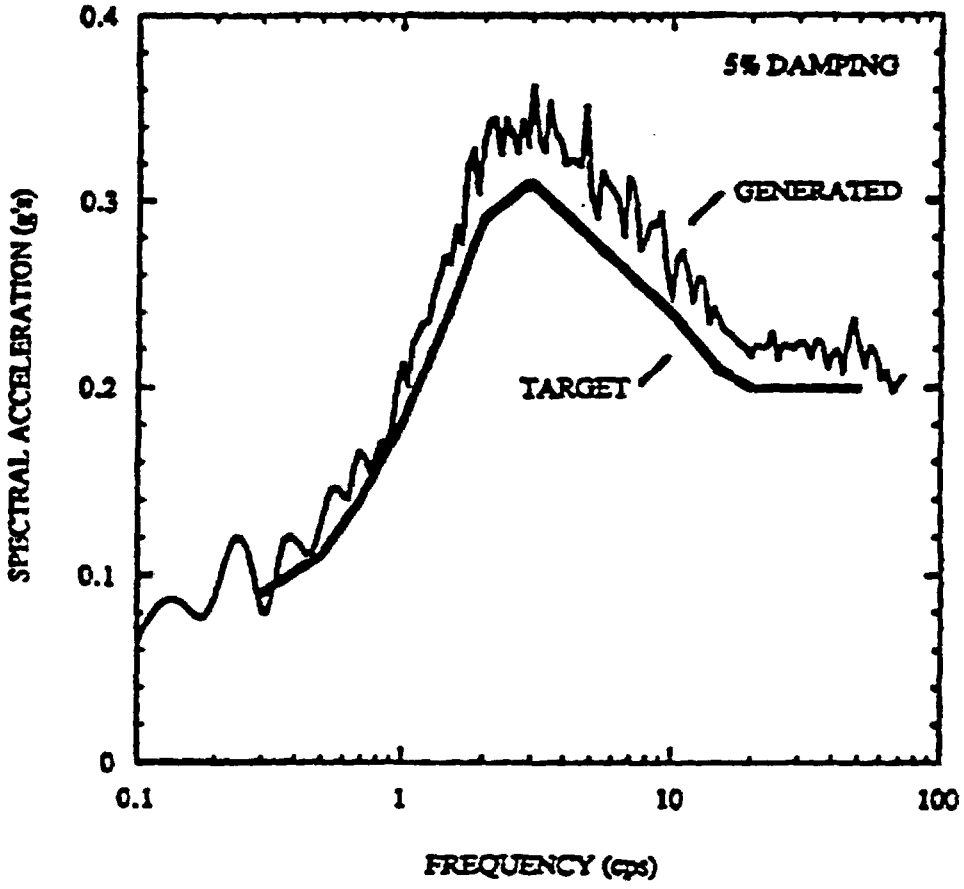

FIGURE 3: HOUSNER TARGET & GENERATED
RESPONSE SPECTRA
PGA = 0.2g

The logs of several borings were provided by the Licensee, with the closest one to the pad being boring D15. At the time of the pad construction, several shallow borings were taken at the pad location but these did not extend to the depth of the ground water table (GWT). They were thus unusable for this liquefaction evaluation. It was suggested at the first site visit that several deeper site specific borings be taken to assist in the development of these liquefaction evaluations. The logs of two borings were then provided (Borings B94-1 and B94-2) which extended to below the ground water table. The descriptions and sample blow counts from these borings were used for this evaluation. Blow counts are measures of the density of the tested soil in units of blows per foot. The higher blow counts means denser soil. The ground water depth at the pad site was found from these borings to be about 34 feet below the ground surface, or at about elevation 590 MSL, which is consistent with the data from the other older borings as well as the known lake elevation.

The borings indicate that the soils at the site consist of relatively uniform dune sands to a depth of about 50 ft below the pad elevation (about elevation 565 to 570 MSL), after which the soils become finer at greater depths. As indicated by the Standard Penetration Test (SPT) blow counts, the dune sands are in a variable density condition. Near the ground surface and above the ground water table, the sands are in a loose to medium dense condition, with SPT blow counts in the range of 10 to 20 blows per foot. At the deeper depths near and below the ground water table, the sands generally become denser and are in a medium dense to dense condition, with sample blow counts obtained in the range of 20 to 40 blows per foot. Some low blow count material (about 13 to 16 blows per foot) was encountered below the ground water table and these lower values were used in the liquefaction and slope stability evaluation. The finer soils below the dune sands become compact to very compact with blow counts exceeding 100 blows per foot and are of no concern for either a liquefaction or stability assessment.

The results of these two site specific borings conform to the general picture of the subsurface conditions throughout the plant site. Based on the number of other boring logs reviewed, the dune sands are assessed to be generally in a medium dense to dense condition below the ground water table but with the potential for encountering some zones of looser material. The newer borings, together with the older boring D15, indicate the existence of a thin strata of loose saturated sands about 5 to 10 feet thick, although this loose condition does not appear to be continuous throughout the site. In particular, the older borings taken between the lake's edge and the pad site (borings B1, B2 and B3 as well as borings D5, D6, D10 and D11) do not indicate that the saturated soft zone is continuous throughout the area.

III. LIQUEFACTION POTENTIAL

Prior to investigating the impact of the seismic motions on slope stability, an estimate of liquefaction potential was made using the profiles obtained from the two new borings, B94-1 and B94-2. The procedure used is shown in detail in Appendix I and is based on the simplified empirical method of evaluation as described, for example, in Refs. 1 and 2. The approach is based on a study of empirical evidence of liquefaction that has occurred at a number of sites located primarily in California, Japan and China. The procedure estimates strength of the saturated soil as defined by its SPT blow count and incorporates factors to account for such items as earthquake magnitude and

duration, soil sample depth and fines content. This strength is then compared with the seismic demand expected from the design basis earthquake. The ratio of seismic capacity to seismic demand is then defined as the safety factor. When the safety factor is greater than unity, the soil will not liquefy, while if the safety factor falls below unity, the potential for liquefaction exists.

Using the SPT sample blow counts for the new borings taken adjacent to the pad, a liquefaction assessment was made with the numerical results listed in Table 1.1 of Appendix I. A range of magnitudes was assumed to indicate the sensitivity of the computations to the earthquake magnitude or duration. The ground water table was located at a depth of 34 feet and only those samples below the ground water table considered. Below a depth of about 55 feet, the high sample blow counts lead to high safety factors and are not listed in the Table. In Boring B94-1, the only safety factors which fall below unity occur for a large magnitude 7.5 event in the depth range from about 37 feet to 53 feet. For the design basis earthquake of Richter magnitude M5.25, the safety factors are all greater than unity.

For Boring B94-2, safety factors for the M7.5 event fall below unity in the depth range from 41 feet to 55 feet. For the M5.25 event, safety factors at a depth of 41 feet and 53 feet are also near or below unity. It is therefore considered reasonable to consider the potential for the softer portion of this zone to liquefy, although it is clear that based on average values of SPT blow counts the entire area under the pad will not liquefy. To assess the sensitivity of the stability of adjacent slopes to liquefaction of these soils, it was assumed that this zone does uniformly liquefy. For the stability analyses using circular failure surfaces, a potential liquefiable zone 15 feet thick was assumed, to allow for a significant portion of the surface to pass through the weakened zone. For the simpler wedge type analyses, a thin weakened zone is all that is required to lead to conservative estimates of slope safety factor.

IV. SETTLEMENT EFFECTS

If the soils in the weakened zone liquefy over a relatively wide area, the settlements induced from the post-event dissipation of the excess pore pressures are estimated to be of the order of three to four inches. The analysis procedure is again based on a empirical procedure using the SPT sample blow counts (Ref. 3). A small amount of additional settlement is also possible from consolidation due to shaking of the dry sands above the ground water table, although it is not felt that this is significant for the M5.25 event.

Using this upper bound value of induced surface settlement, the impact on the concrete pad itself was evaluated in the following manner. First, the 195 feet long pad is divided at its center by a construction joint, with the two halves of the pad only connected together by short dowels. It was therefore assumed that half of the pad is subjected to a differential settlement of 3 to 4 inches at its center and held with no movement at both ends, an extreme deflection assumption. Assuming the pad to crack at its center with no additional bending, the total angle of rotation that could develop was computed to be about 0.01 radians. Following the guidance provided in Section C.3.4 of ACI 349-80 (Ref. 4), the rotational capacity of any hinge is limited to a value of 0.07 radians. Even at this angle of assumed pad rotation, the casks will not tip over. This upper limit is based upon test data on beams and is indicated to be conservative for the deeper sections. Thus, using a most conservative

estimate of behavior of the ground surface as well as the pad, a significant amount of margin is available before any concern for separation and failure of the pad need be voiced.

V. STABILITY OF NORTH-SOUTH SLOPES

The stability of the NS slopes was made using a quasi-static conservative analysis of the slope subjected to a peak horizontal acceleration of 0.2 g. The analysis assumes that this acceleration is applied to the slope without any variation in time. Thus, the slope is subjected to static forces caused by its own dead weight acting vertically as well as a horizontal inertial force proportional to the seismic acceleration and the mass of the soil within the assumed failure zone. In the stability analysis of the slopes both shallow and deep surfaces were analyzed. The shallow surface is associated with the surficial soils of the slopes while the deep failure surface penetrates into the weak soil zones beneath the pad. The failure surfaces are either cylindrical (hence circular in the two-dimensional analysis of the slope) or noncircular (e.g., wedge-shaped).

The deeper failure surfaces are selected to be circular, passing through the weakened liquefied zone and breaking out downstream of the toe. The weight of soil at the bottom of the slope which must be moved to allow the failure, acts as a counterbalance to the driving forces. The details of the analyses are provided in Appendix II. The REAME computer code (Ref. 6) was used and is a state-of-the-art computer program to evaluate stability. The evaluation of the slope also included shallow failure surfaces which do not reach the weakened zone.

The first analyses performed assumed that the seismic loads are applied to the slope with no liquefaction of the weakened soil occurring. The purpose of the analysis is to define the potential movement caused by expected shallow slope failures. With the SSE peak ground acceleration applied, the computer safety factor falls slightly below unity as expected since most natural slopes have safety factors near unity. Some sensitivity analyses were also performed which are provided in Appendix II. To bring the safety factor back to unity, the slope would have to flatten somewhat from its current configuration. The potential movement of the toe of the slope to reachieve equilibrium is estimated to be about 5 feet.

For the case of a complete loss of strength of the liquefiable soils below the base of the slope, and assuming that this zero strength material extends uniformly to great distances from the toe of the slope, the REAME analysis leads to safety factors falling well below unity, implying that some slope movement would occur. For this case, a more detailed finite element dynamic analysis method was applied, based on a relatively complex analysis using the full variation of ground motion. The analysis is based upon the POROSLAM computer program (Ref. 5) which has been used for other stability evaluations. The purpose of the analysis was to eliminate to as large an extent as possible the conservatism inherent in an already conservative problem definition. The details of the analysis are provided in Appendix III.

The results indicate that to achieve safety factors having an average value of unity following the motion, residual soil strengths of the weakened soils must be relied upon to reach values of about 800 psf. In addition, for the deep

failures to occur, the failure surfaces must break out well down stream of the toe of the slope, into the existing slope at the opposite side of the pad. If this additional counterweight were included in the stability analysis, the residual soil strength required to maintain slope integrity would be significantly decreased.

The average value of residual shear strength available for such clean sands with SPT blow counts varying from 13 to 16 blows per foot range from about 500 to 800 psf (Refs. 7 and 8). Such strength estimates are based upon the results of laboratory testing of sand samples as well as from the known behavior of slopes that have been subjected to major seismic events (such as the Lower San Fernando Dam Section). These strengths indicate that the soil does not lose its strength completely as was postulated in the analyses above, but maintains a minimum value even at large strains. Considering that the soft zone is not anticipated to be uniformly located under the site, potential slope movements due to deep failure surfaces are not expected to be significant. Using reasonable estimates of residual soil strength, as well as the counterweight effects of the soils above the ground water table, computed safety factors for potential deep failure surfaces exceed a value of 1.1 under SSE loading conditions.

VI. STABILITY OF EAST-WEST SLOPE

From the site visit, it was noted that in the EW direction a gentle slope of about 1 on 10 exists from the pad area to the lake. Again, to investigate the impact of potential liquefaction conditions on slope movements in this direction, a simplified quasi-static REAME analysis was performed and is summarized in Appendix IV. In these evaluations it was again conservatively assumed that the liquefiable zone extends from under the pad directly to the water's edge. Even though this slope is flatter than those in the NS direction, the lack of any counterbalancing soils at the water's edge leads to a potentially more unstable situation if the weakened soil zone does in fact extend to this distance.

The results indicate that residual strengths of this weakened zone required to achieve safety factors of unity are of the order of 400 to 600 psf. A simplified analysis assuming a wedge-shaped failure surface was also performed confirming these values. Based on the residual strengths available for these soils mentioned above, it is not expected that the potential for gross slope movements is significant in the EW direction. In addition, a review of the logs of borings west of the pad, between the pad and the lake, provided by the licensee indicates that the soft zone does not extend to the water's edge. Thus, it is felt that the potential for an extensive deep failure into the lake is remote. Using the available boring data to restrict the size of the liquefied zone, slope stability factors for the EW slope exceed values of 1.3.

An alternate failure path was considered which evaluates a potential slope failure in a more southwesterly direction from the pad, with the failure extending towards the area at which some temporary structures are located, and which has an elevation of 590 before reaching the lake. The distance to the water's edge from the pad is somewhat greater than the uniform slope considered above. In addition, the amount of soil mass acted upon by the earthquake to drive the slope is less than that for the uniform slope case considered above, since the slope had to be excavated to achieve the lower elevation of 590 MSL. A deep seated wedge analysis was performed, again assuming that liquefaction of the soft zone below the water table was developed. Due to the reduced driving

mass of soil, as well as the increased resistance from the somewhat longer failure surface, the stability of this slope is greater when considering the deep failure due to liquefaction of the soft zone.

In addition to this deep seated failure, an alternate slope condition was evaluated to assess the potential for shallow failure of the steeper section of the above slope in the southwesterly direction. A cross-section of this slope condition is shown in Fig. 4. The slope rises from an elevation of 590 MSL at an angle of about 30 degrees to about elevation 620 MSL at the top of the crest, then gradually rising to the pad elevation. Performing a quasi-static failure analysis of this slope indicates that this slope may flatten to an angle of about 23 degrees when subjected to the 0.2 g design basis earthquake. The toe of the slope will move about 10 feet towards the temporary structures at the bottom and the crest of the slope will move about 10 feet towards the pad which is 75 feet away from the crest of the slope. The impact of this failure condition on the pad is not considered significant.

VII. SUMMARY

The effects of an SSE on the pad and its adjacent slopes has been evaluated for a variety of assumed conditions which can be summarized as follows:

A. It is anticipated that some slope movements of the NS slopes can occur when subjected to the design basis earthquake. These movements are associated with shallow surface type failures. This can lead to relatively small toe movements of the order of 5 feet.

B. If the soft pocket of soil below the pad immediately below the ground water table fully liquefies during the design basis event, the maximum surface settlements that can be estimated are of the order of 3 to 4 inches. If these settlements are placed on the pad in a conservative configuration, it is expected that the pad at worst will crack but is not expected to fail or separate.

C. If an extensive soft liquefiable zone exists under the pad area and if this soil loses all its strength, the safety factors of the NS slopes will fall to below unity. For the soil type under and around the pad area, residual strengths of approximately 800 psf are believed to be available which results in a safety factor of unity. If the counter balancing effects from soils above the GWT are included in the stability assessment, the resulting safety factor is estimated to be greater than 1.1.

D. The slope in the EW direction from the pad to the water's edge does not benefit from any counterbalancing effect from adjacent slopes but is significantly flatter than the NS slopes. To provide stability following an event, residual strengths of the order of 400 psf to 600 psf are required. Again, the residual strength of these type soils is expected to be significantly higher than this value. Considering the restricted zone of potentially liquefiable soil and using reasonable estimates of residual soil strength, computed safety factors for deep slope failure exceed values of 1.3.

E. The stability of the slope in the SW direction from the pad was also considered since this slope drops at a sharper rate to an adjacent area at elevation 590 MSL. This steeper slope reduces the effective driving force when considering a deep failure zone associated with liquefaction. However, the

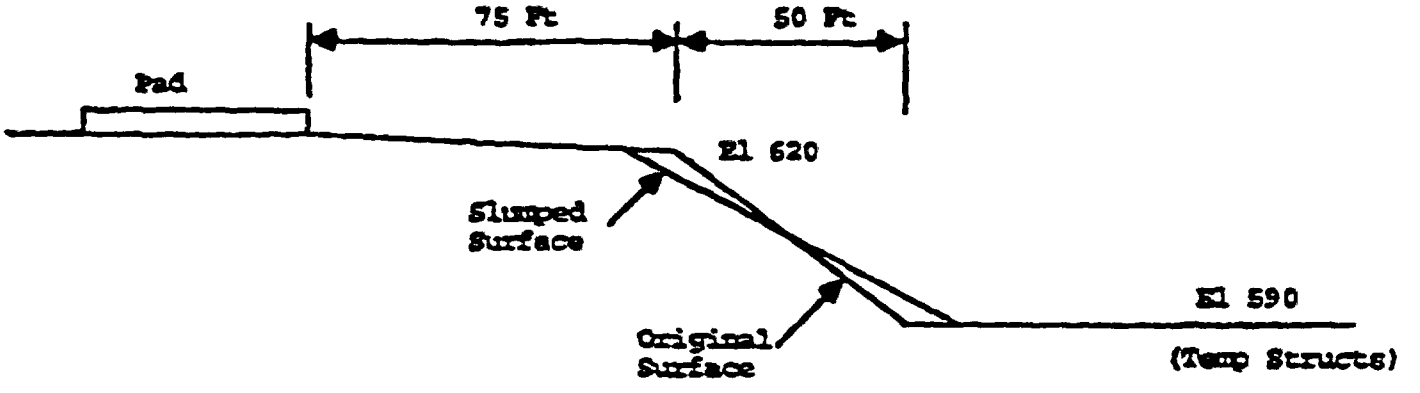

FIGURE 4

SHALLOW FAILURE CONDITION
IN SW SLOPE

steeper slope increases the potential for a shallow failure when subjected to the design basis earthquake. An analysis of this shallow failure indicates that the toe and crest of the slope will move at most about 10 feet. Since the crest is about 75 feet from the edge of the pad, this potential movement is not considered detrimental to the pad.

In summary, if a design basis earthquake occurs at Palisades, some slumping of the sand dunes adjacent to the storage pad could occur. However, the maximum movement would be 5 to 10 feet at the toe of the dunes. Thus, one can conclude that this potential slope slumping has essentially no impact on the storage pad.

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APPENDIX I TO ATTACHMENT 1

LIQUEFACTION ASSESSMENT OF SOILS UNDER CONCRETE PAD

An empirical liquefaction assessment has been made of the soils beneath the concrete storage pad, using the information available from the recent borings taken immediately adjacent to the pad area (Borings B94-1 and B94-2, see Table 1.1). These boring logs indicate that SPT samples were taken from the ground surface through the softer dune sands (SP) which extend to a depth of about 50 to 55 feet below the surface and into the stiffer silty sands (SM-SP) and silts (ML) below. An approximate profile is shown in Fig. 1.1 from the descriptions provided on the logs. In addition, the descriptions are compared with those from an older boring D15 taken nearby. These results indicate that the descriptions are fairly consistent. The ground water table (GWT) found in the newer borings is indicated to lie about 34 feet below the ground surface.

The values of the SPT blow counts obtained from the logs of these three borings are shown in Fig. 1.2 and are tabulated at their proper depth. From these blow counts, the dune sands are generally in a loose to medium dense condition near the surface, increasing in density with depth. The silty sands are compact while the silts below are very compact (blow counts exceeding 100 blows/foot).

At several locations within the dune sands below the GWT, loose pockets of sands were found, as indicated by the low values of their blow counts. Plots of the blows per foot obtained from the dune samples are shown in Fig. 1.3 which indicate the variability with depth. Fig. 1.4 indicates approximate values of shear wave velocity and shear modulus estimated from the sample blow counts.

TABLE 1.1
PALISADES NPP
BORINGS B941 & 2

SAMPLE NO	DEPTH (FT)	SIGMA V TOTAL (TSF)	PORE PRESSURE (TSF)	SIGMA V PRIME (TSF)	N (BPF)	CN	N1 (BPF)	% FINES	DELTA N1	N1 EFFECT	TAU/SIGMA CHART	K SIGMA	STRENGTH RATIOS M7.5	M6	M5.25	R SUB D	T/S INDUCED	SAFETY FACTORS M7.5	M6	M5.25
BORING B94-1:																				
15	33	1.980	.000	1.980	38	.743	28.76	5	0	28.76	.309	0.836	0.258	0.341	0.387	.902	.117	2.20	2.91	3.30
16	35	2.106	.031	2.075	25	.728	18.20	5	0	18.20	.199	0.823	0.164	0.216	0.246	.889	.117	1.40	1.84	2.10
17	37	2.238	.094	2.144	20	.717	14.35	5	0	14.35	.152	0.814	0.124	0.163	0.185	.875	.119	1.04	1.37	1.56
18	41	2.502	.218	2.284	17	.697	11.84	5	0	11.84	.125	0.796	0.100	0.131	0.149	.844	.120	0.83	1.09	1.24
19	43	2.634	.281	2.353	22	.687	15.10	5	0	15.10	.161	0.788	0.127	0.167	0.190	.827	.120	1.05	1.39	1.58
20	47	2.898	.406	2.492	17	.667	11.34	5	0	11.34	.120	0.771	0.093	0.122	0.139	.792	.120	0.77	1.02	1.16
21	49	3.030	.468	2.562	22	.658	14.47	5	0	14.47	.153	0.764	0.117	0.155	0.176	.773	.119	0.99	1.30	1.48
22	51	3.162	.530	2.632	22	.649	14.27	5	0	14.27	.151	0.756	0.114	0.151	0.171	.754	.118	0.97	1.28	1.45
23	53	3.294	.593	2.701	44	.640	28.14	5	0	28.14	.356	0.749	0.267	0.352	0.400	.735	.117	2.29	3.02	3.43
BORING B94-2:																				
15	35	2.106	.031	2.075	22	.728	16.02	5	0	16.02	.173	0.823	0.142	0.187	0.213	.889	.117	1.21	1.60	1.82
16	37	2.238	.094	2.144	26	.717	17.93	5	0	17.93	.197	0.814	0.160	0.212	0.240	.875	.119	1.35	1.78	2.03
17	39	2.370	.156	2.214	29	.707	20.50	5	0	20.50	.229	0.805	0.184	0.243	0.276	.860	.120	1.54	2.03	2.31
18	41	2.502	.218	2.284	13	.697	9.06	5	0	9.06	.099	0.796	0.078	0.104	0.118	.844	.120	0.65	0.86	0.98
19	45	2.766	.343	2.423	19	.677	12.86	5	0	12.86	.136	0.779	0.106	0.140	0.159	.810	.120	0.88	1.16	1.32
20	47	2.898	.406	2.492	21	.667	14.01	5	0	14.01	.148	0.771	0.114	0.151	0.172	.792	.120	0.96	1.26	1.43
21	49	3.030	.468	2.562	24	.658	15.79	5	0	15.79	.170	0.764	0.130	0.171	0.194	.773	.119	1.09	1.44	1.64
22	51	3.162	.530	2.632	26	.649	16.86	5	0	16.86	.183	0.756	0.139	0.183	0.208	.754	.118	1.18	1.55	1.77
23	53	3.294	.593	2.701	16	.640	10.23	5	0	10.23	.110	0.749	0.082	0.108	0.123	.735	.117	0.70	0.93	1.06
24	55	3.426	.655	2.771	43	.631	27.13	5	0	27.13	.333	0.742	0.247	0.326	0.371	.716	.115	2.15	2.84	3.22
25	59	3.690	.780	2.910	30	.614	18.42	5	0	18.42	.203	0.728	0.148	0.195	0.222	.678	.112	1.32	1.75	1.99
26	61	3.822	.842	2.980	35	.606	21.20	15	4.5	25.70	.297	0.721	0.214	0.283	0.321	.659	.110	1.95	2.57	2.92

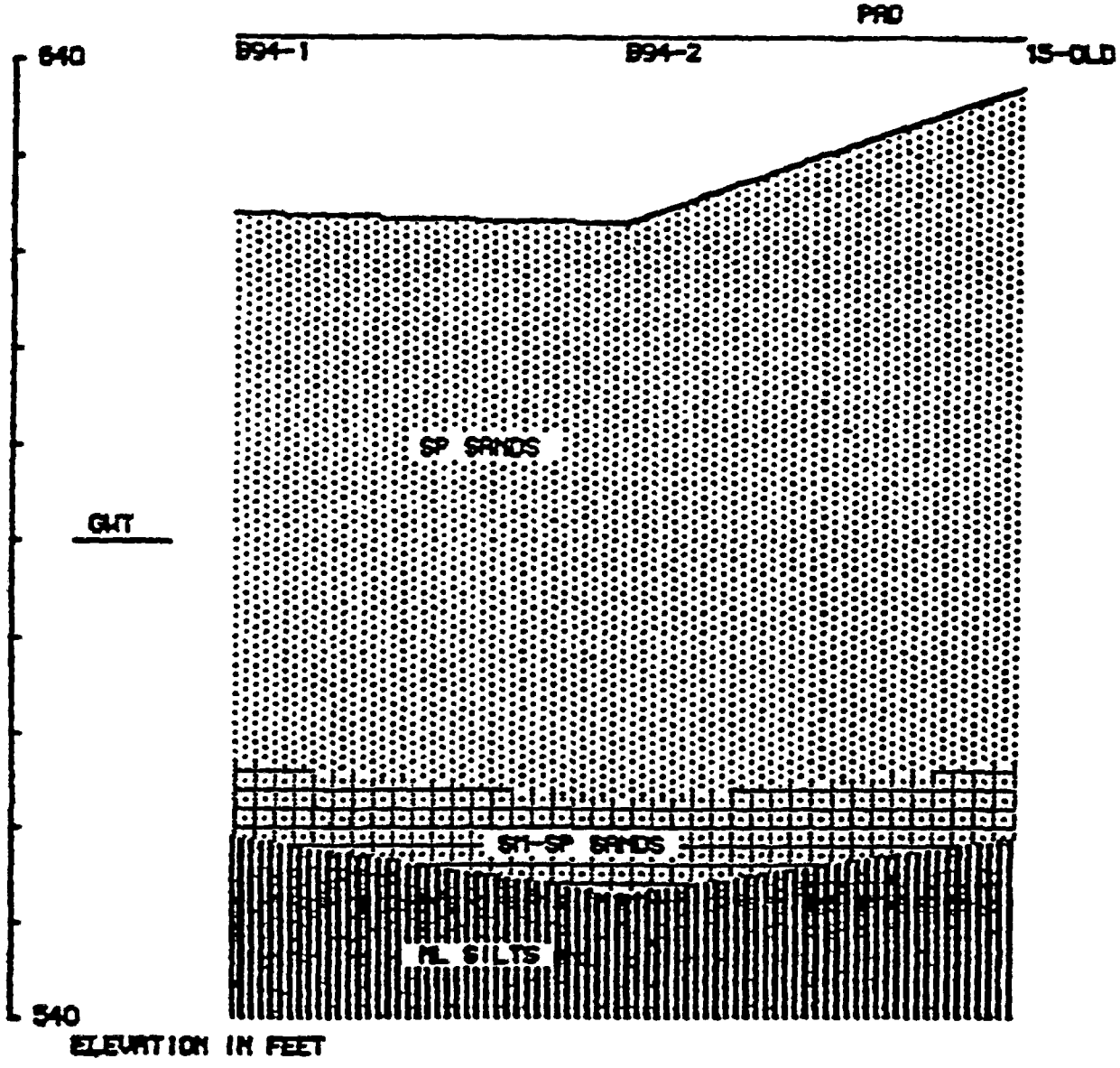

PRO
640
B94-1
B94-2
15-OLD
SP SANDS
GWT
SM-SP SANDS
ML SILTS
540
ELEVATION IN FEET

FIG. 1.1

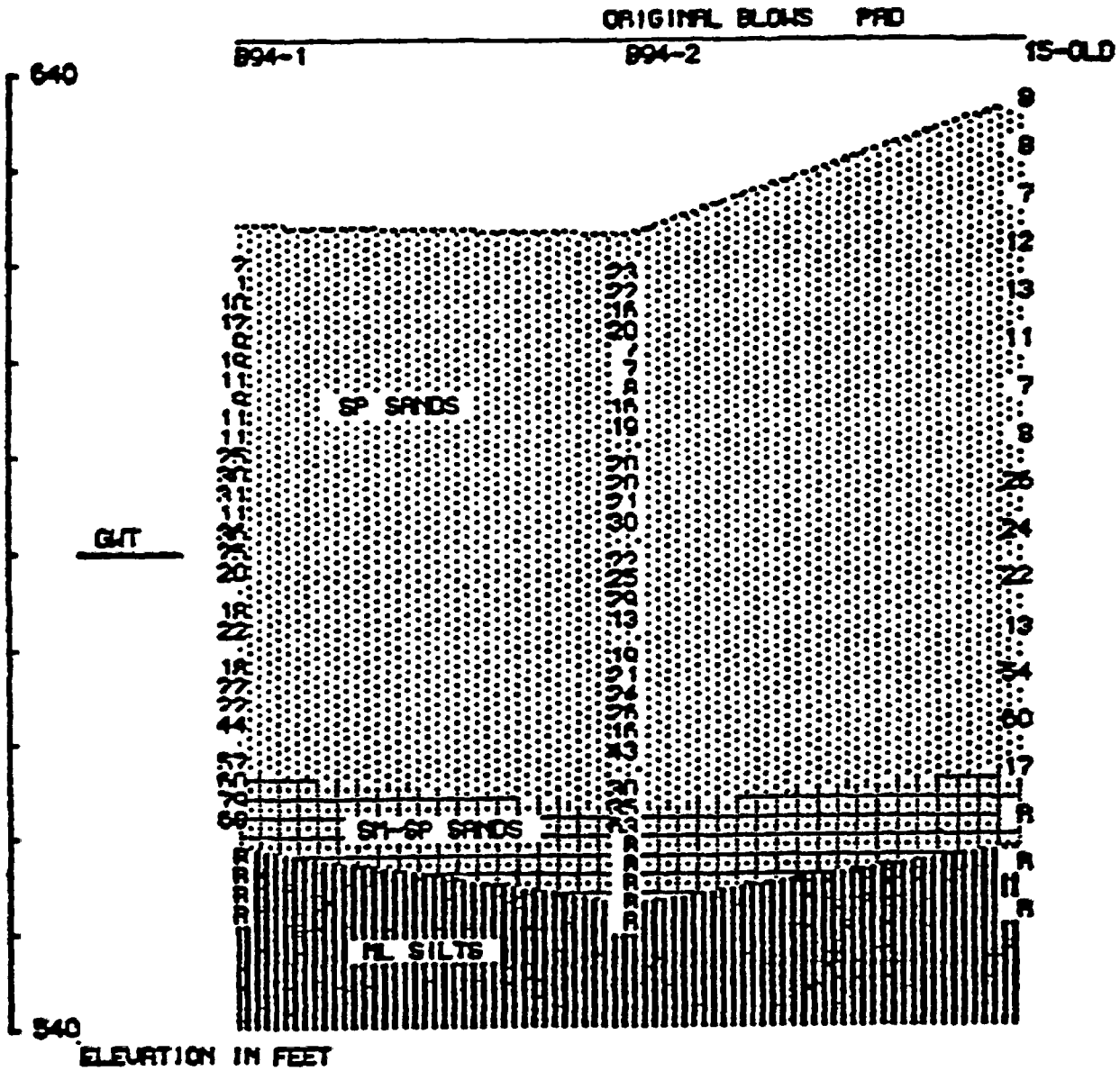
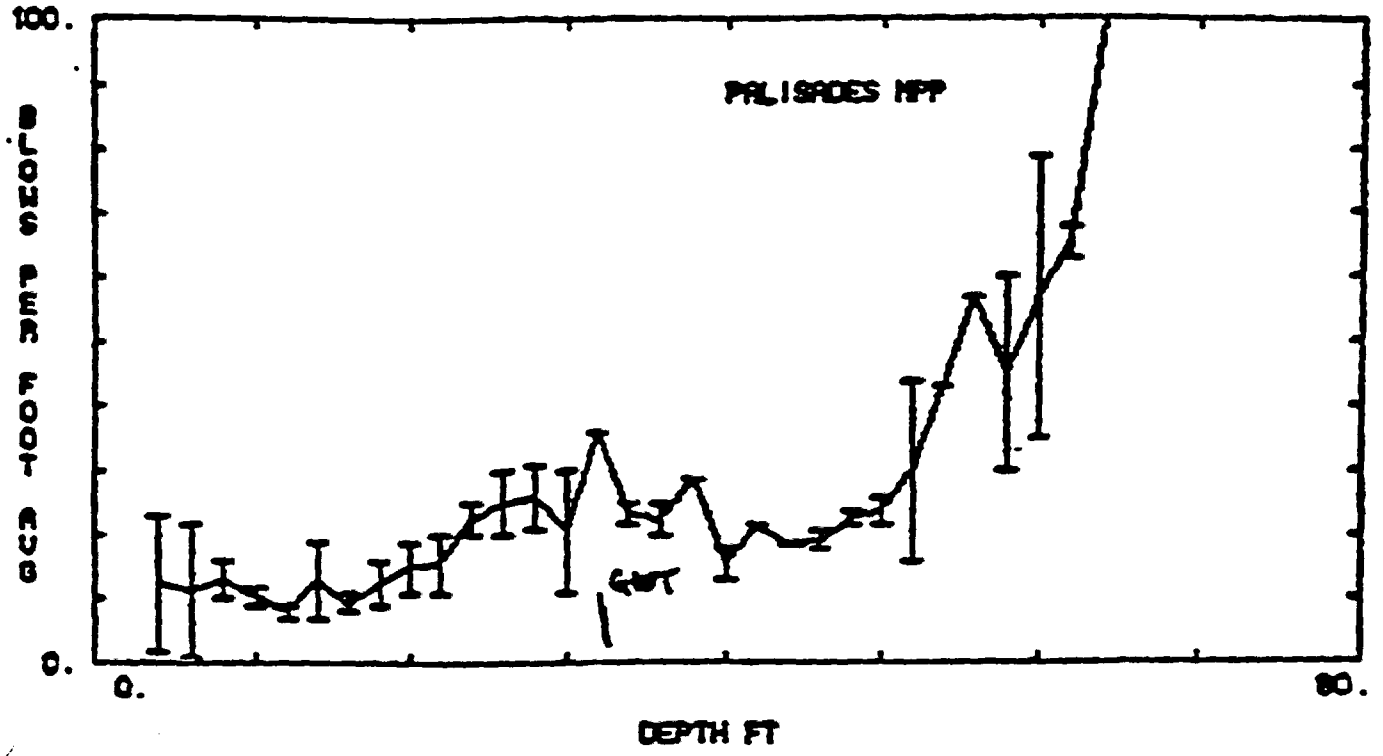

ORIGINAL BLOWS PRO
B94-1
B94-2
15-OLD
640
540
GWT
SP SANDS
SM-SP SANDS
ML SILTS
ELEVATION IN FEET

FIG. 1.2


100.
0.
BLOWS PER FOOT-BOR
PALISADES NPP
GWT
0.
80.
DEPTH FT

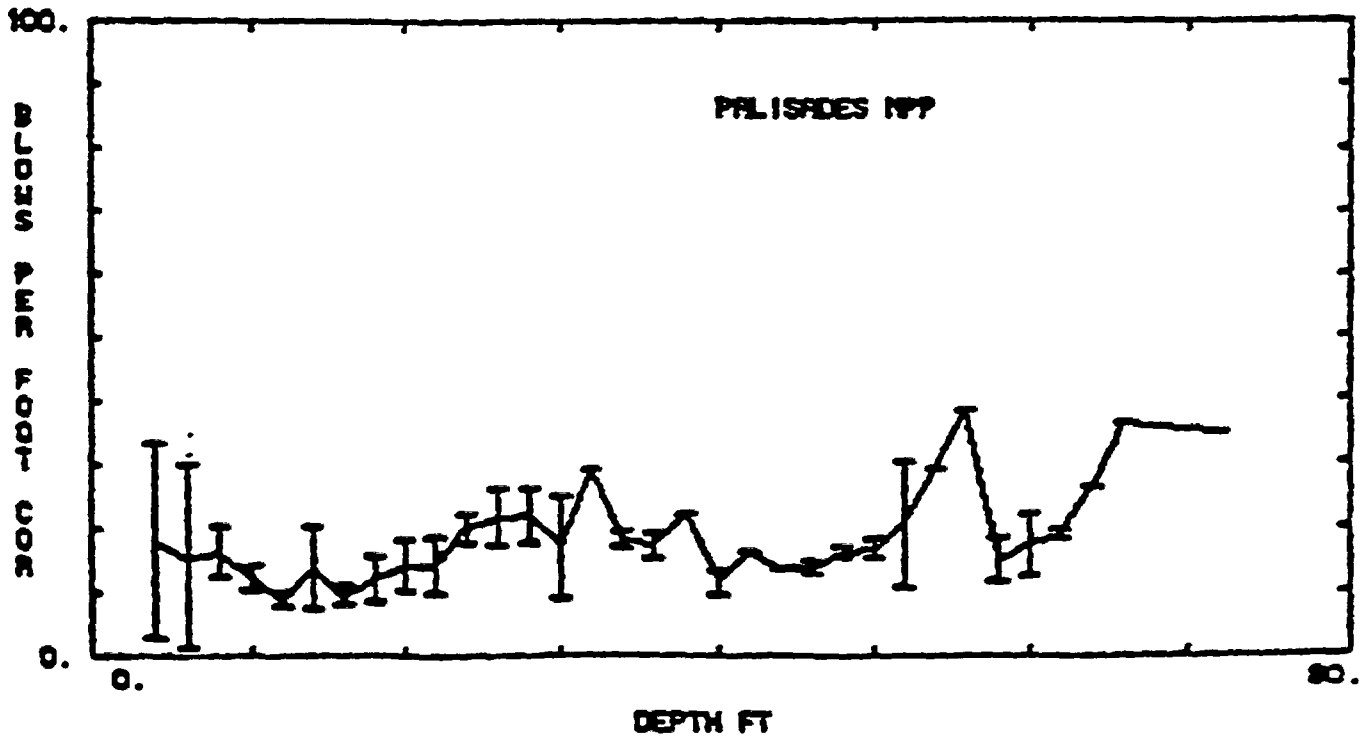
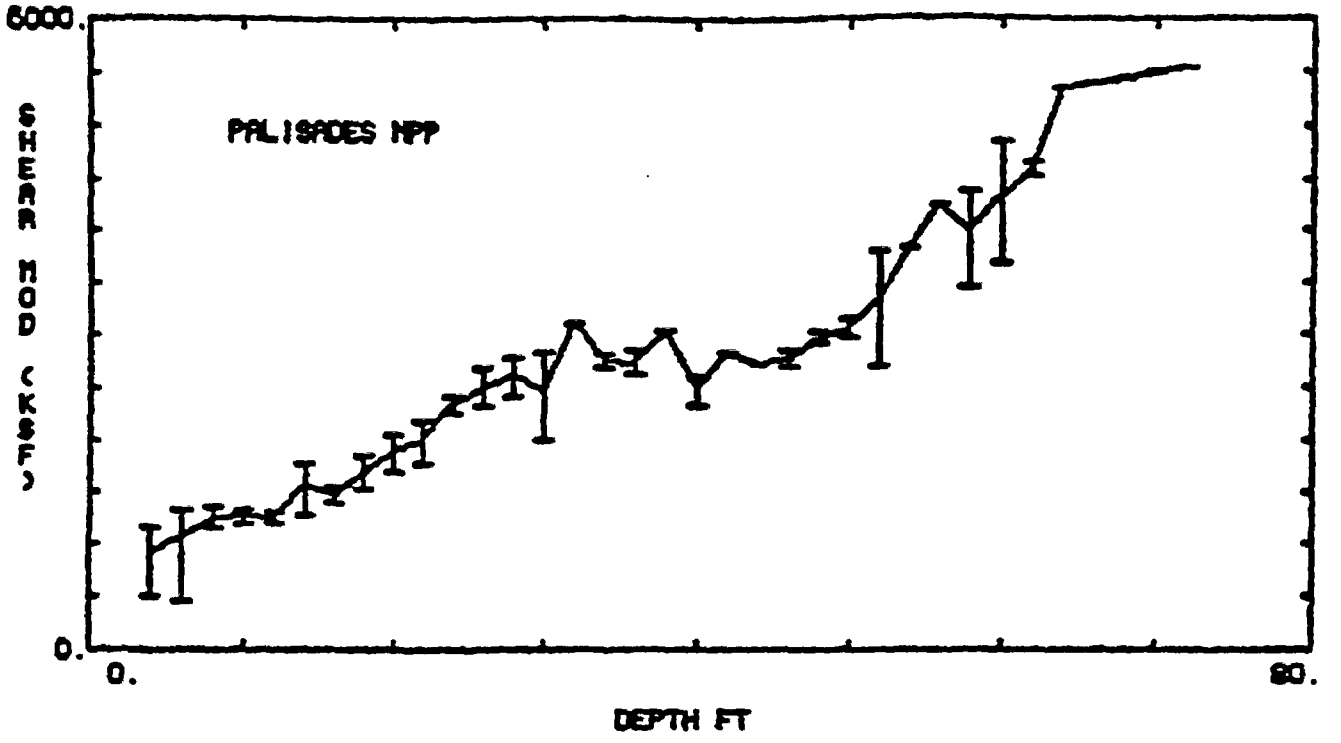

100.
0.
BLOWS PER FOOT-COR
PALISADES NPP
0.
80.
DEPTH FT

FIG. 1.3


6000.
PALISADES NPP
SHEAR MOD (KSF)
0.
0.
80.
DEPTH FT

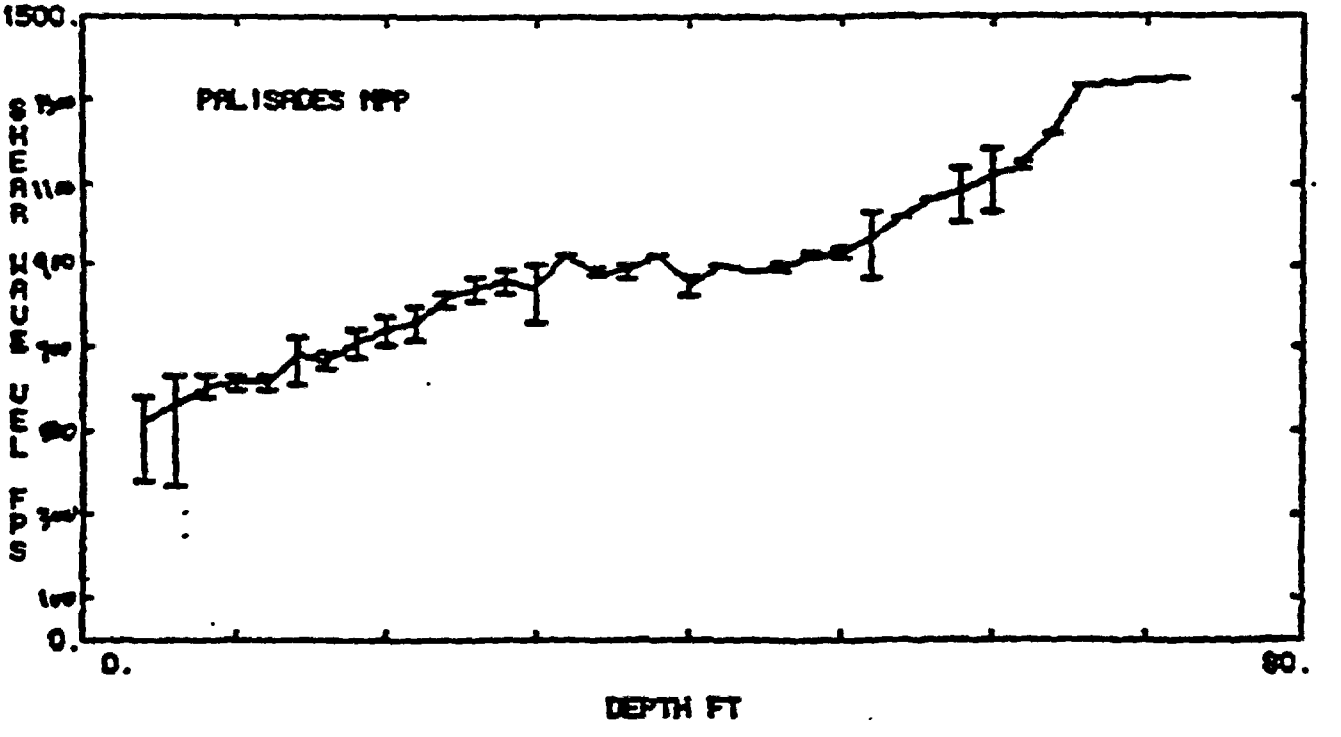

1500.
PALISADES NPP
SHEAR WAVE VEL FPS
0.
0.
80.
DEPTH FT

FIG. 1.4

APPENDIX II TO ATTACHMENT 1

STABILITY OF NORTH-SOUTH SLOPES: REAME ANALYSIS

The REAME (Rotational Equilibrium Analysis of Multilayered Embankments) computer code is used in determining conservative assessments of the slope sliding potential at the Independent Spent Fuel Storage Installation.

The program can evaluate factors of safety against sliding based on a "cylindrical failure surface" and the Bishop method. It can handle different soil zones, seepage or excess pore pressure. The analysis is quasi-static (uses a seismic coefficient that reflects the PGA at the site) and can only handle horizontal seismic loads.

For the ISFSI site both the north and south slopes were studied and the factors of safety against slope sliding were calculated. Specifically, the following figures (Fig. II.1 & 2) show the cross section of the two slopes and the applicable properties assigned to the different soils. It should be noted that for the analysis of the north slope, Elev. 0 was arbitrarily set at the bottom of the slope in the vicinity of the pad. Similarly, for the analysis of the south slope, Elev. 0 was arbitrarily set at the bottom of the slope at the top of the sheet pile wall.

NORTH SLOPE

From the REAME analysis of the north slope the safety factors obtained are shown in the following figures. Specifically,

a. For static conditions with seismic coefficient (S.C.) = 0 and soil properties for those of Fig. II.1 - the location of the potential failure surface and the accompanying minimum safety factor are shown in Fig. II.3.

b. For seismic conditions with S.C. = 0.1 (0.1 g) - the results are shown in Fig. II.4. The 0.1 g case is examined to get a feel for the sensitivity of FS to PGA.

c. For seismic conditions with S.C. = 0.2 (0.2 g) - the results for this case are shown in Fig. II.5.

d. For seismic conditions (S.C. = 0.2 g) for the north slope that has been flattened (specifically the toe moved to the right (in Fig. II.1) by 5 feet and the top of the slope to the left by 5 feet.) - the results which show a safety factor better than 1.0 are presented in Fig. II.6.

e. For seismic conditions (S.C. = 0.2 g) and liquefied critical zone (cohesion = 0, and friction angle, Ø, = 0^0) - the results shown in Fig. II.7 indicate F.S. < 1.0.

f. For seismic conditions (S.C. = 0.2 g) with the liquefied zone assumed to have a residual strength in the form of cohesion C = 1000 psf - Fig. II.8 still indicates that F.S. < 1.0; however, a more rigorous and sophisticated analysis (Appendix III) showed that for the same surface a C = 800 psf is needed for F.S. > 1.0.

SOUTH SLOPE

For the south slope the failure conditions for S.C. = 0.2 have been calculated and shown below (Fig. II.9). The safety factor is calculated to be S.F. = 0.9 which is similar to the condition found for the north slope (Fig. II.5). A flattening of the slope (similar to that of the north slope shown in Fig. II.6) would bring the safety factor back to unity.

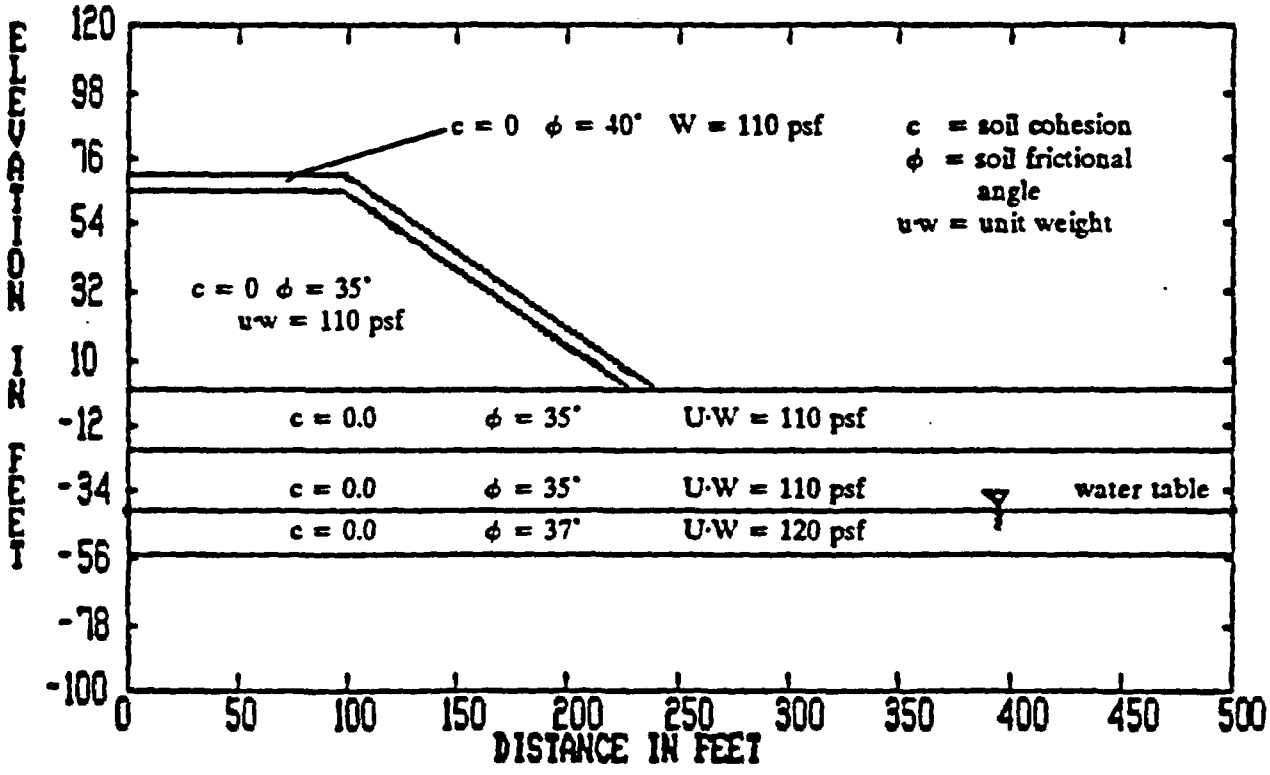


FIG. II.1: CROSS SECTION OF THE NORTH SLOPE

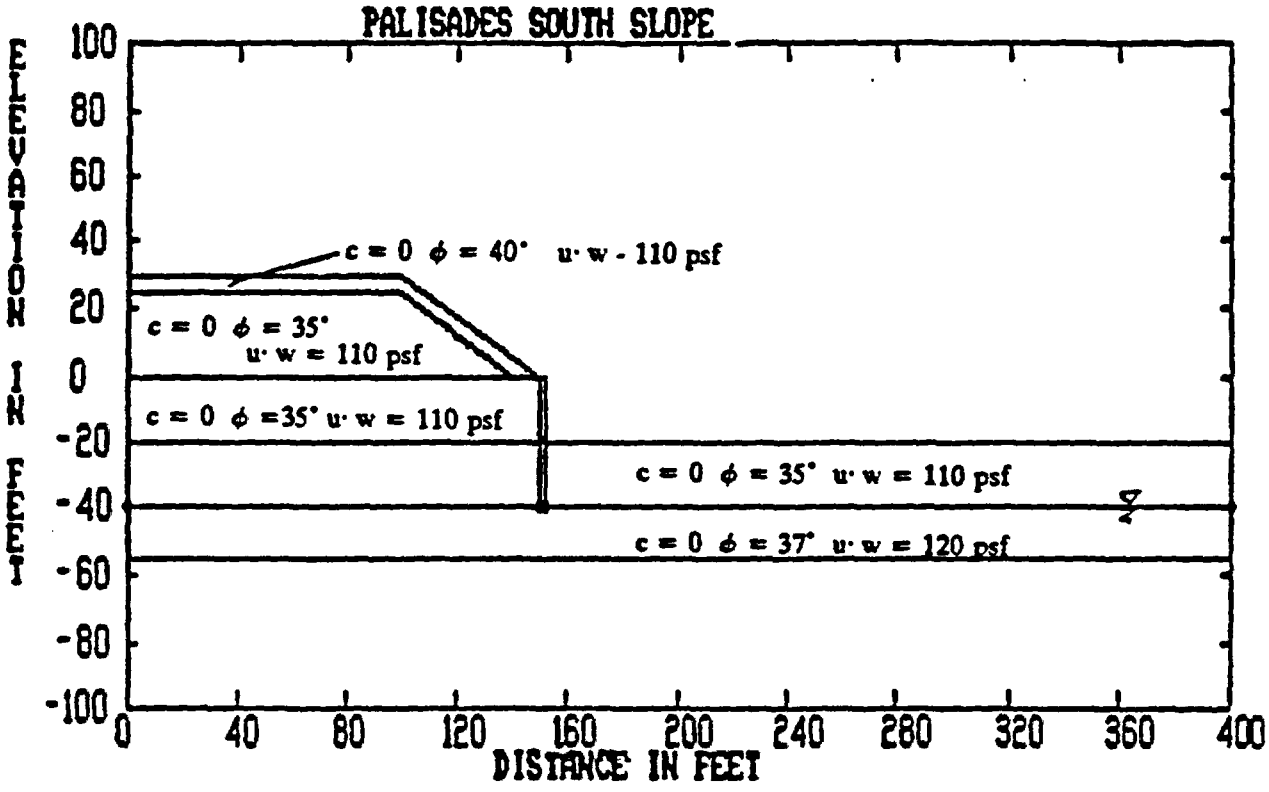


FIG. II.2 CROSS SECTION OF THE SOUTH SLOPE

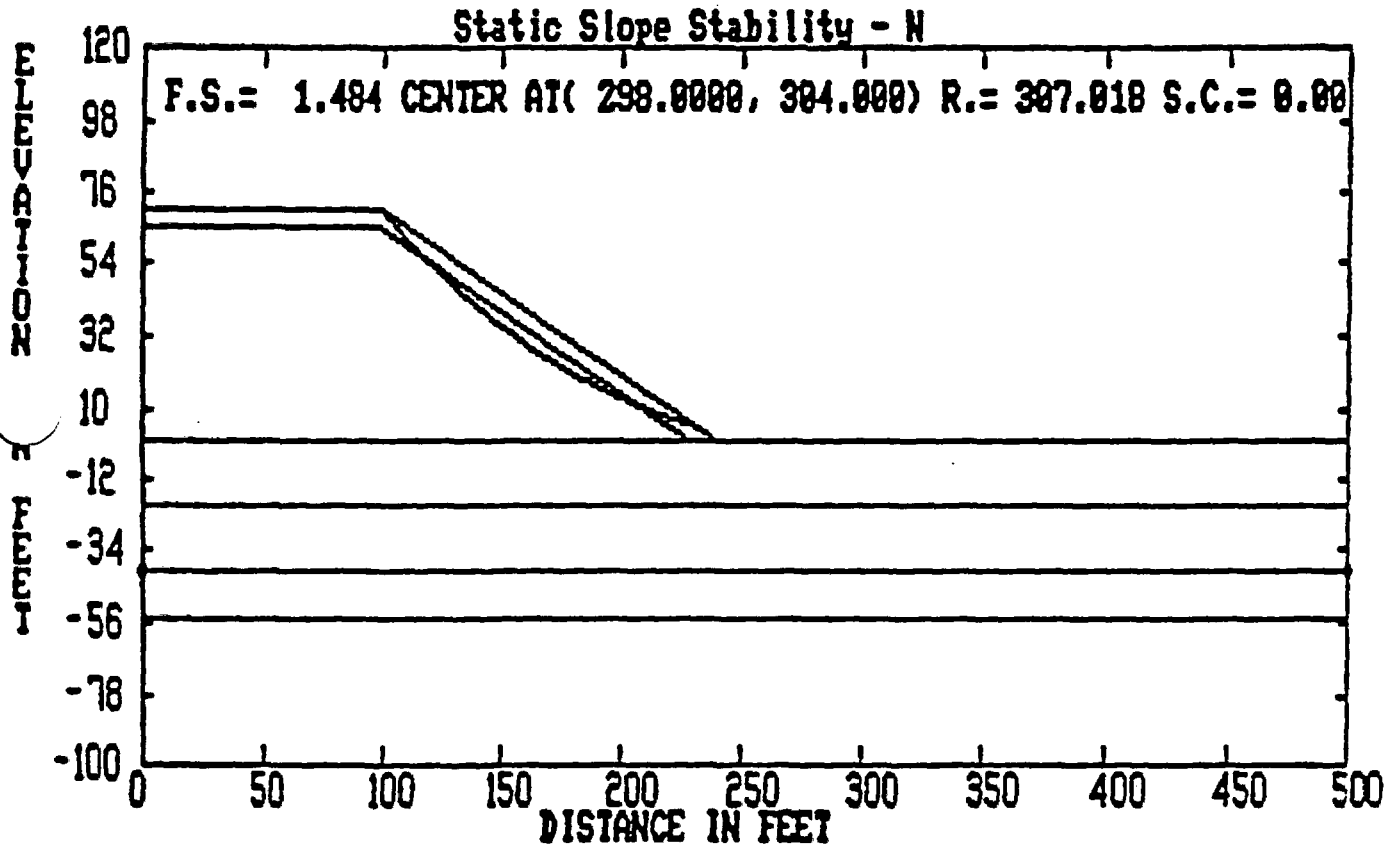


Fig. II.3: NORTH SLOPE

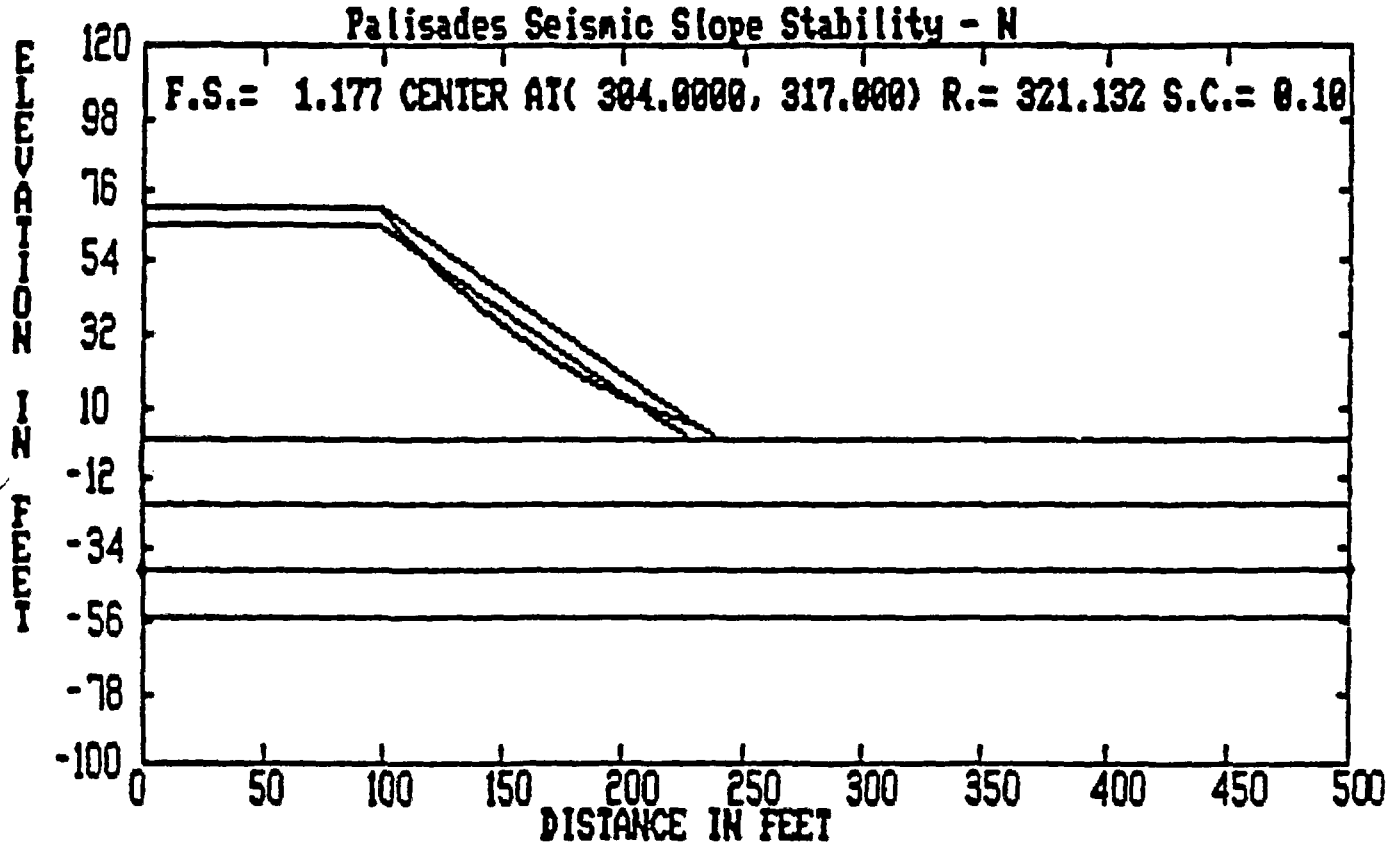


FIG. II.4: NORTH SLOPE (PGA = 0.1 g)

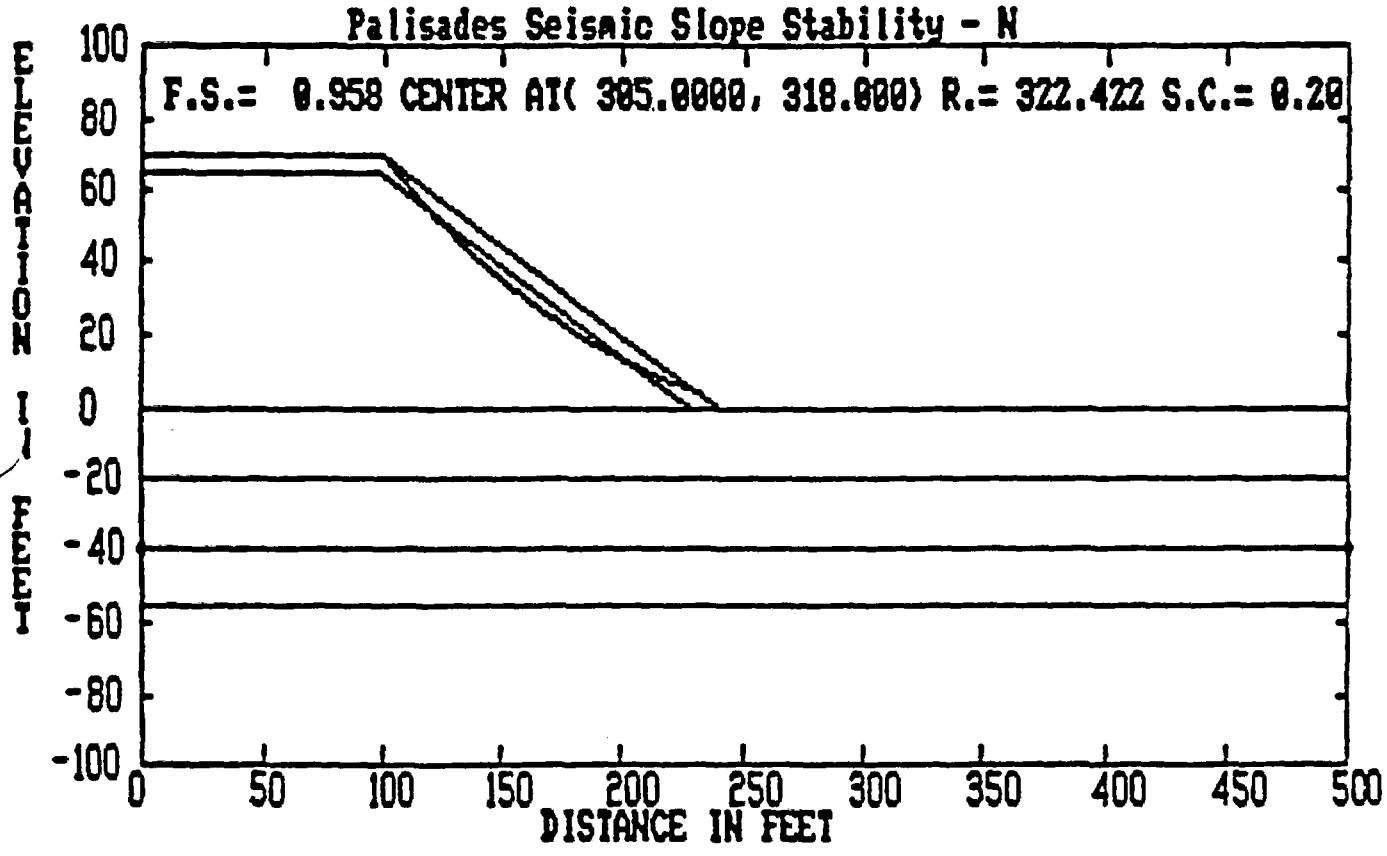


FIG. II.5: NORTH SLOPE (PGA 0.2 g)

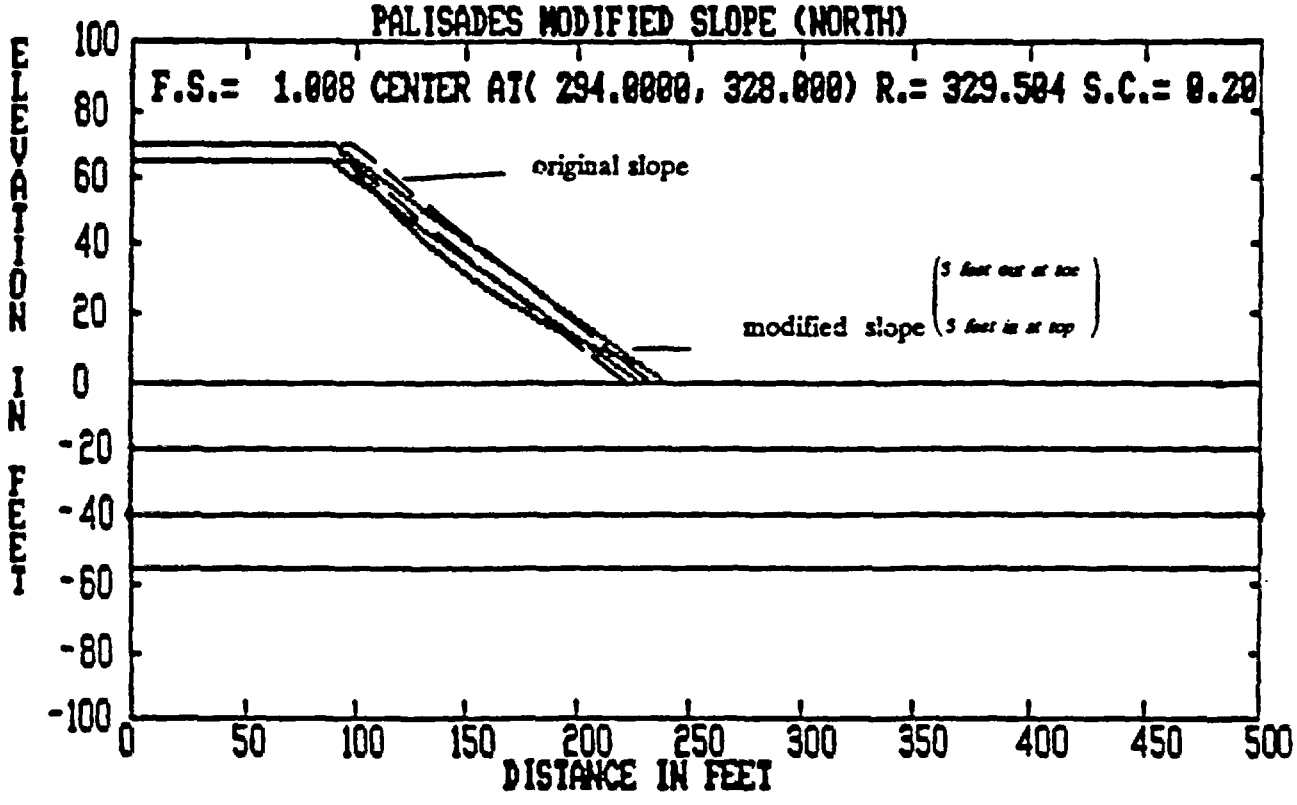


FIG. II.6: FLATTENED NORTH SLOPE (PGA = 0.2 g)

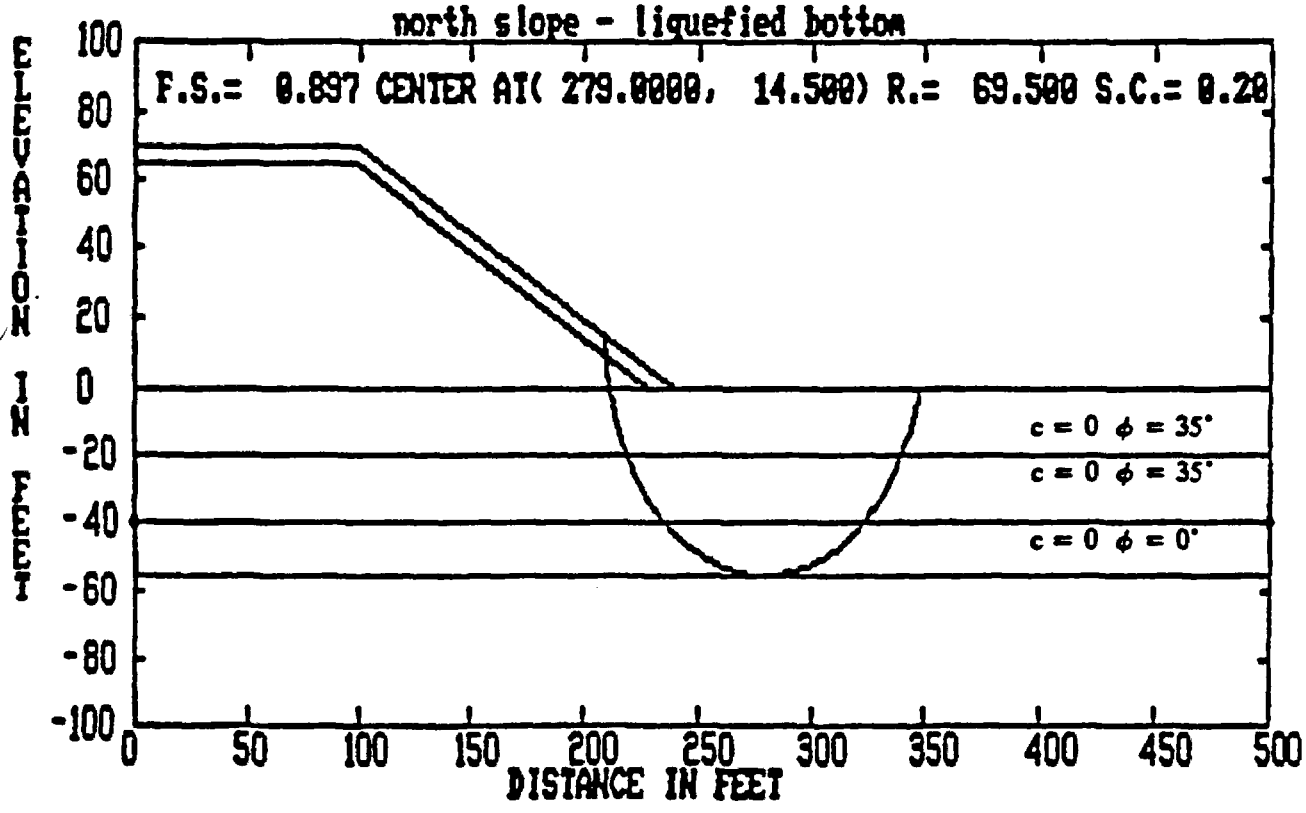


FIG. II.7: LIQUEFIED BOTTOM ZONE ($c = 0, \phi = 0°$)

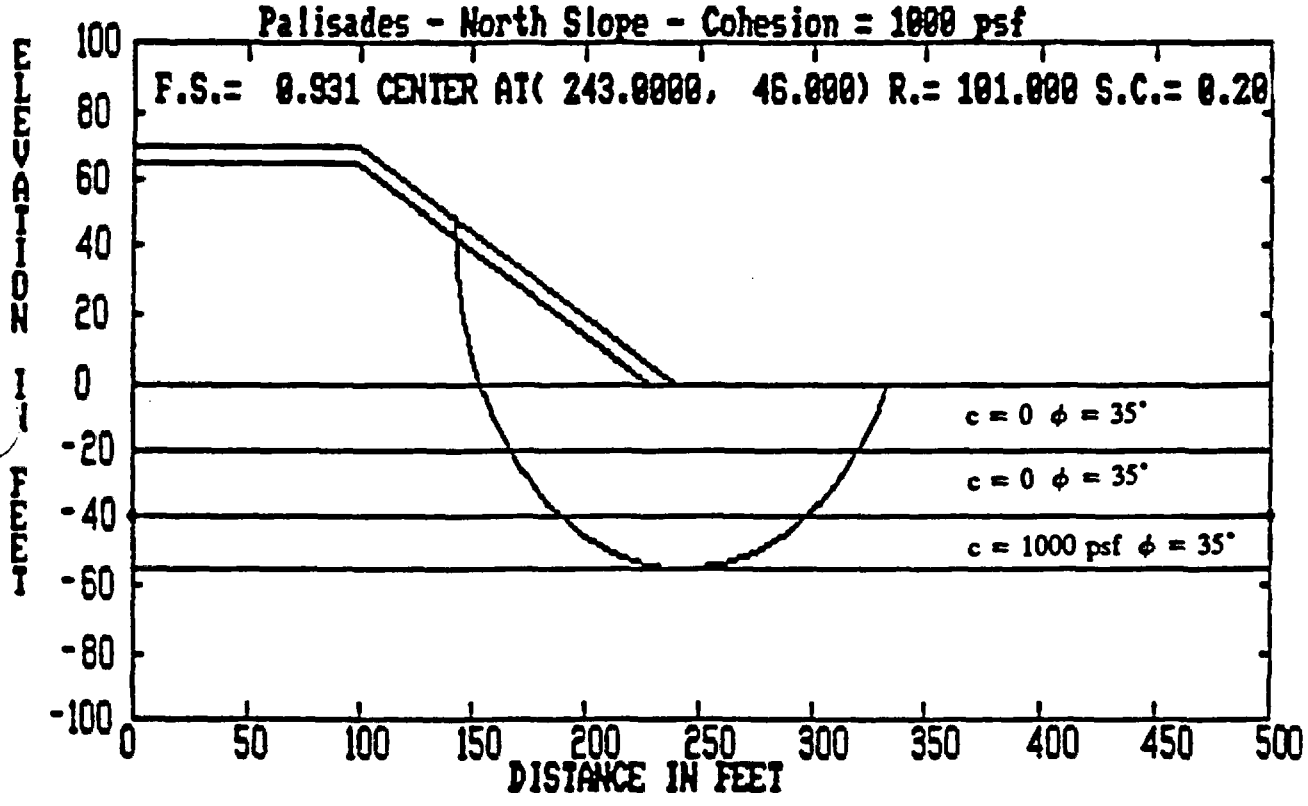


FIG. II.8: LIQUEFIED BOTTOM ZONE WITH RESIDUAL STRENGTH

SOUTH SLOPE

For the south slope the failure conditions for S.C. = 0.2 have been calculated and shown below (Fig. II.9). The safety factor is calculated to be S.F. = 0.9 which is similar to the condition found for the north slope (Fig. II.5). A flattening of the slope (similar to that of the north slope shown in Fig. II.6) would bring the safety factor back to unity.

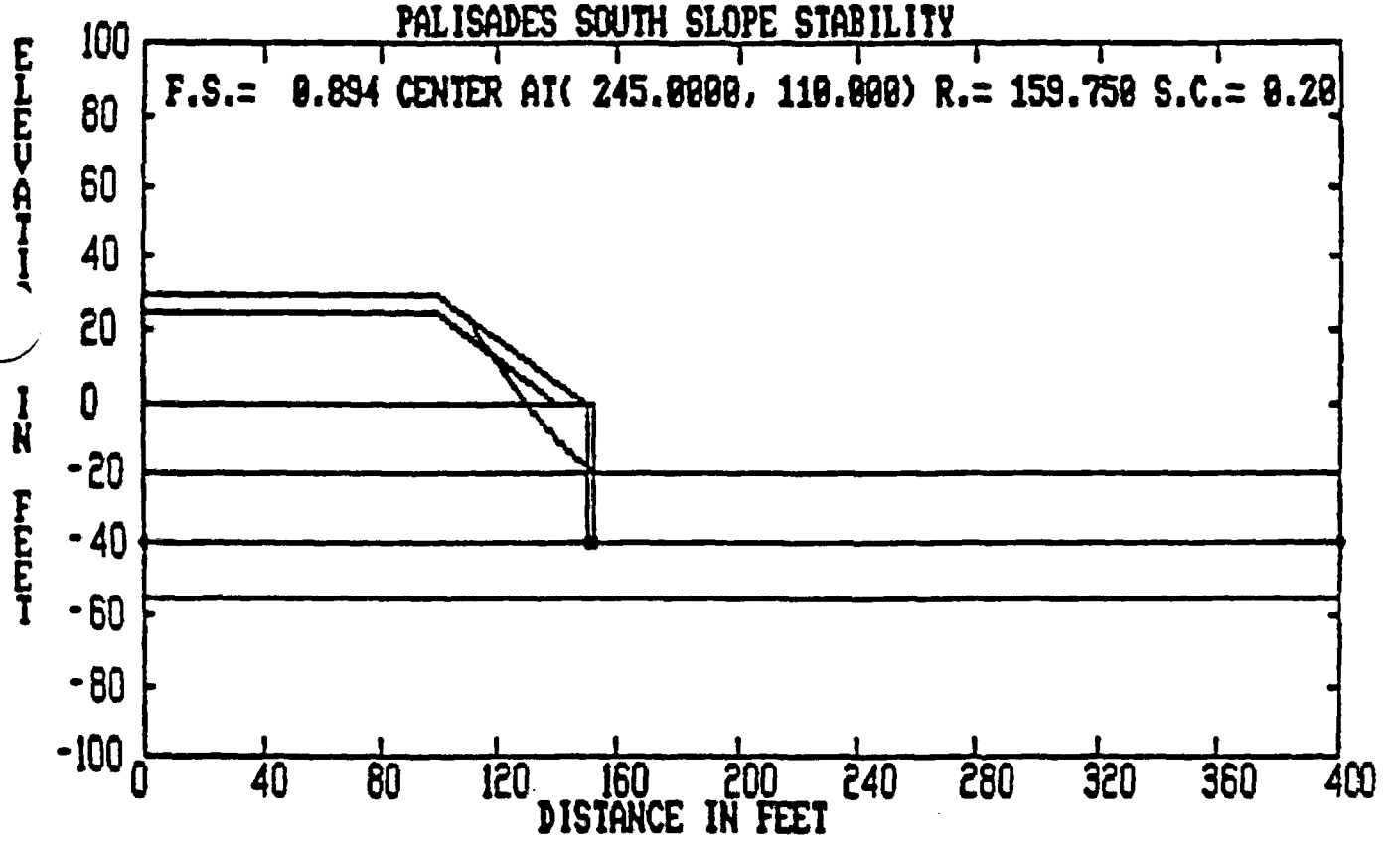


FIG. II.9: SOUTH SLOPE (PGA = 0.2 g)

APPENDIX III TO ATTACHMENT 1

STABILITY OF NORTH SLOPE: POROSLAM ANALYSIS

The computer code POROSLAM which is capable of treating the soil as a two-phase medium (i.e., soil skeleton & pore water) is employed to study the slope stability of the north slope.

The finite element discretization is shown in Fig. III.1, while the in-situ vertical and shear stresses of the section are shown in Figs. III.2 & 3, respectively. Fig. III.4 shows the location of the two selected failure surfaces.

The earthquake input is a generated acceleration time history based on a 0.2 g Housner Spectrum. The target spectrum and the generated acceleration are shown in Fig. III.5. For the two slopes, a shallow one and one that penetrates the zone under the water table, the safety factor is calculated as a function of time (duration of earthquake = 10 sec). Fig. III.6 depicts the safety factor variation of the shallow slope. It is apparent from Fig. III.6 that the more sophisticated analysis produces safety factors in the shallow slope that are above 1.0 (the REAME analysis safety factors were less than 1.0 for the same failure surface). The analysis also indicated that when the weak zone in the foundation layer holds firm (does not liquefy) the safety factors for the deep failure surface are well above 1.0.

Assuming, however, that the critical zone in the foundation has liquefied, the safety factor for the deep failure surface drops below 1.0 (see Fig. III.7). For this assumed liquefied state, both the frictional angle and the cohesion of the soil were considered to be zero.

By allowing the liquefied zone to retain some strength in the form of cohesion, the safety factors for the same deep failure surface are evaluated. The results of this step are shown in Fig. III.8.

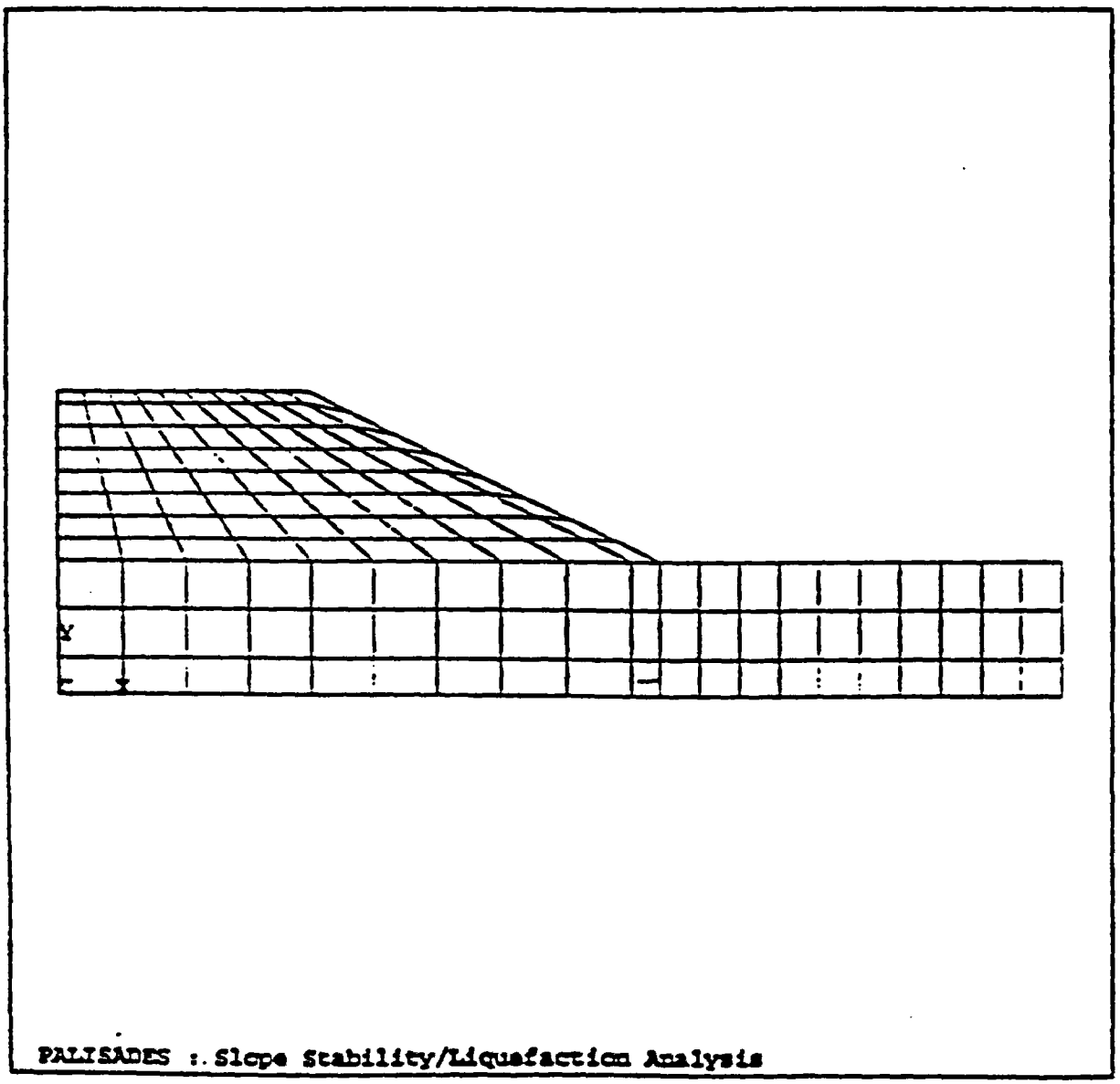


FIG. III.1: FINITE ELEMENT DISCRETIZATION OF THE NORTH SLOPE

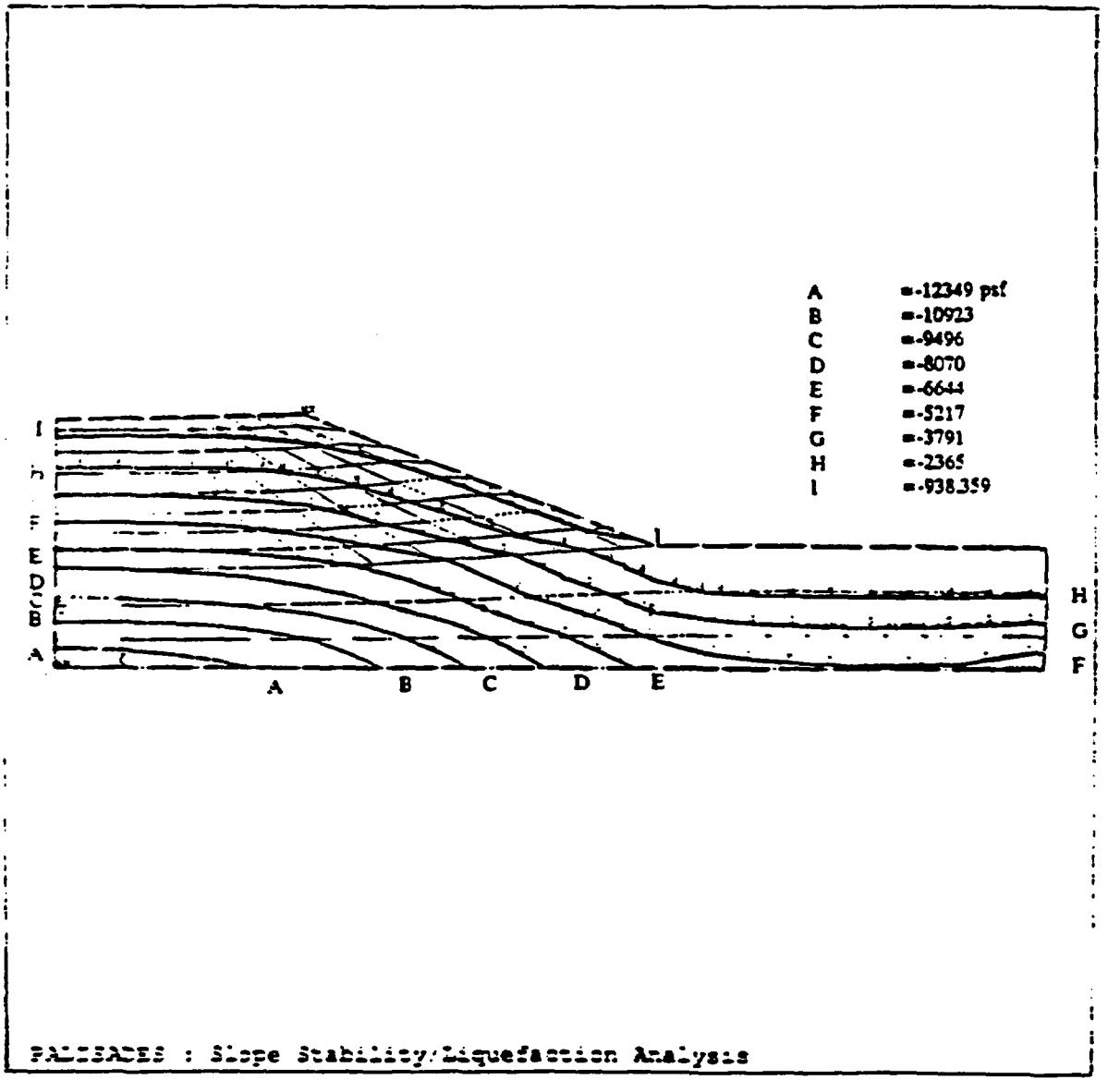


FIG. III.2: IN-SITU VERTICAL STRESS

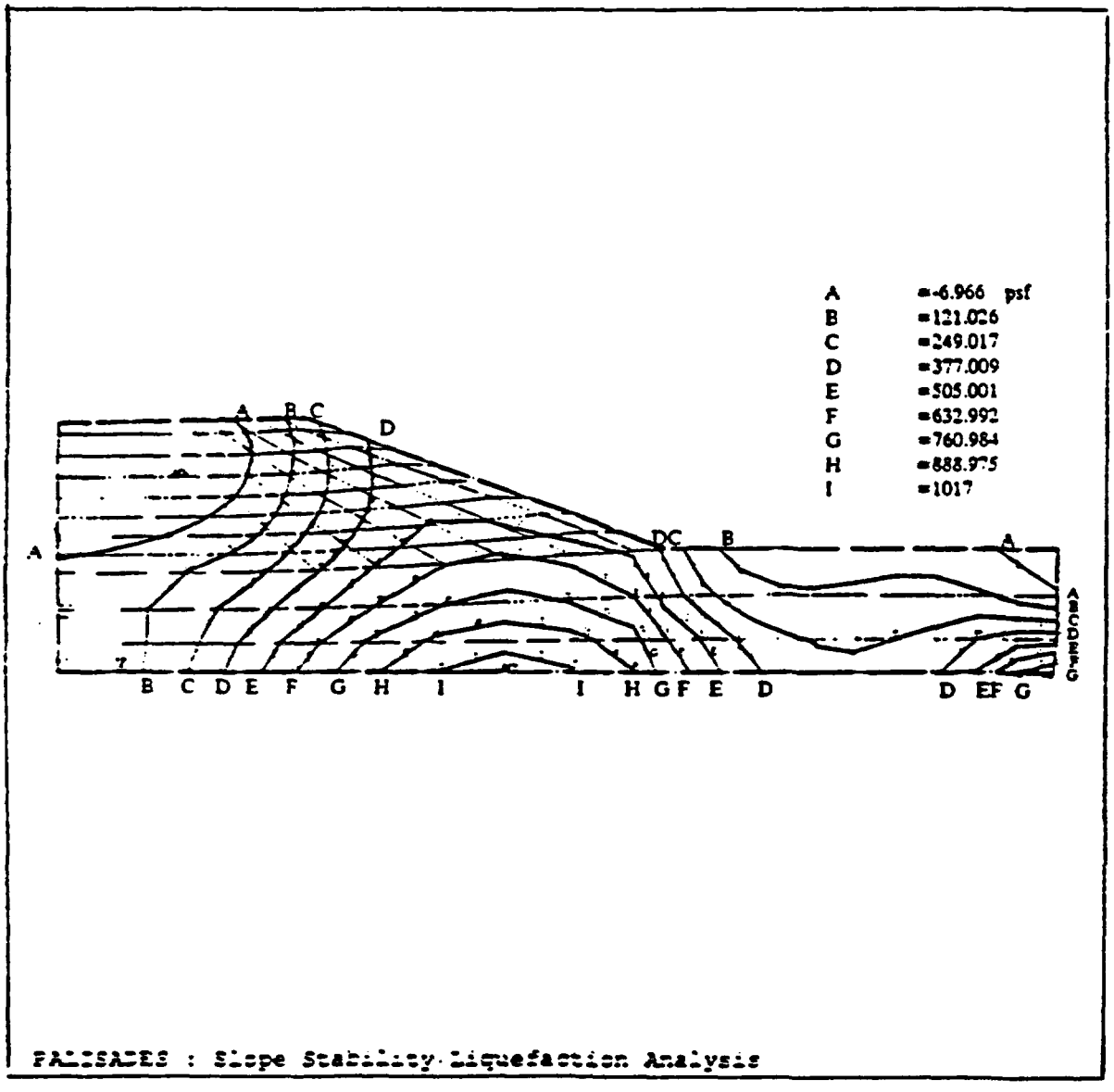


FIG. III.3: IN-SITU SHEAR STRESS

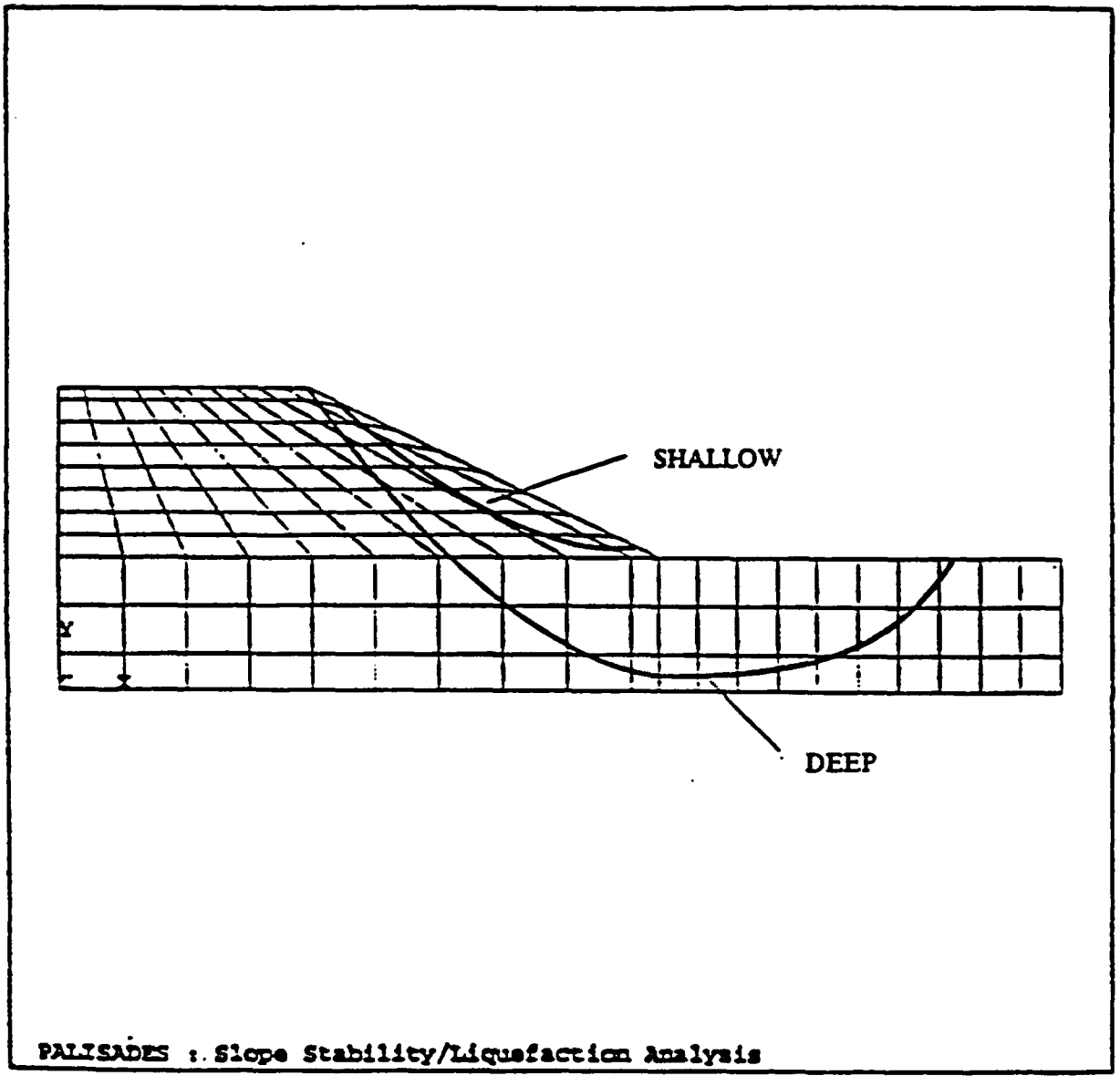
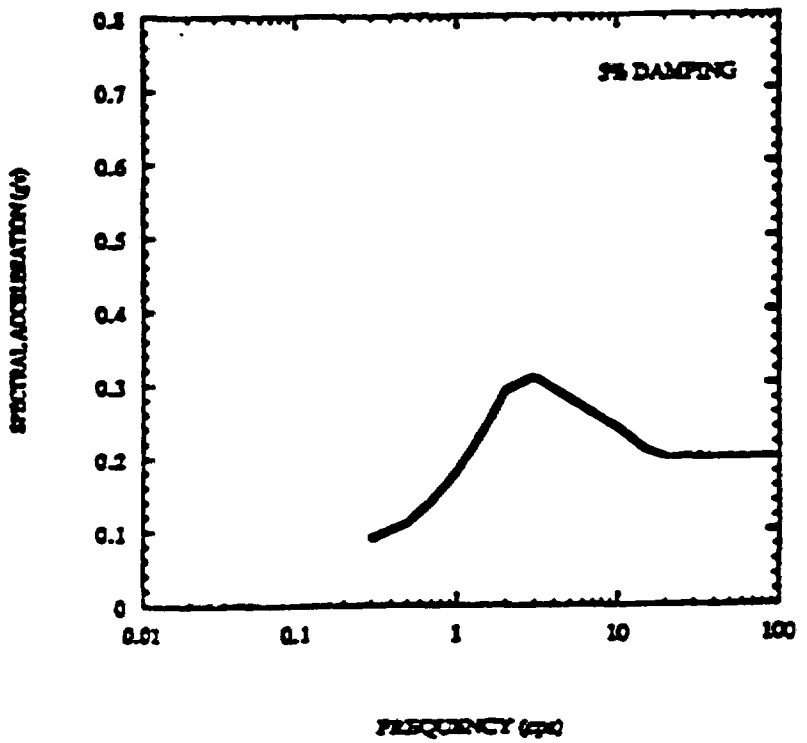


FIG. III.4: LOCATION OF SELECTED FAILURE SURFACES



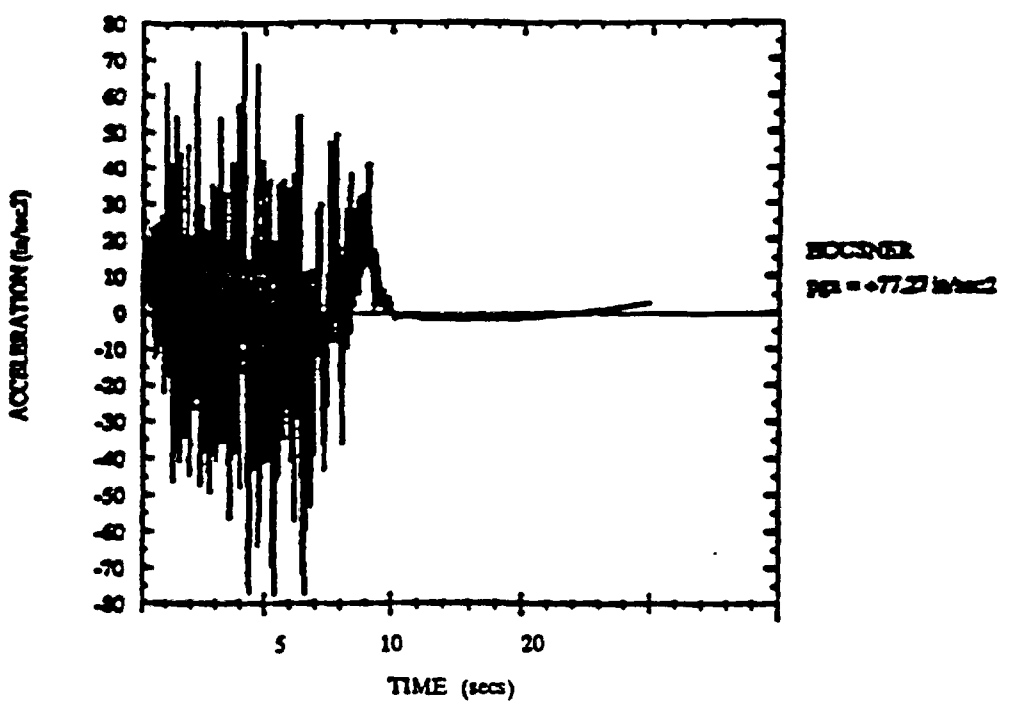


FIG. III.5: TARGET RESPONSE SPECTRUM AND GENERATED ACCELERATION TIME HISTORY

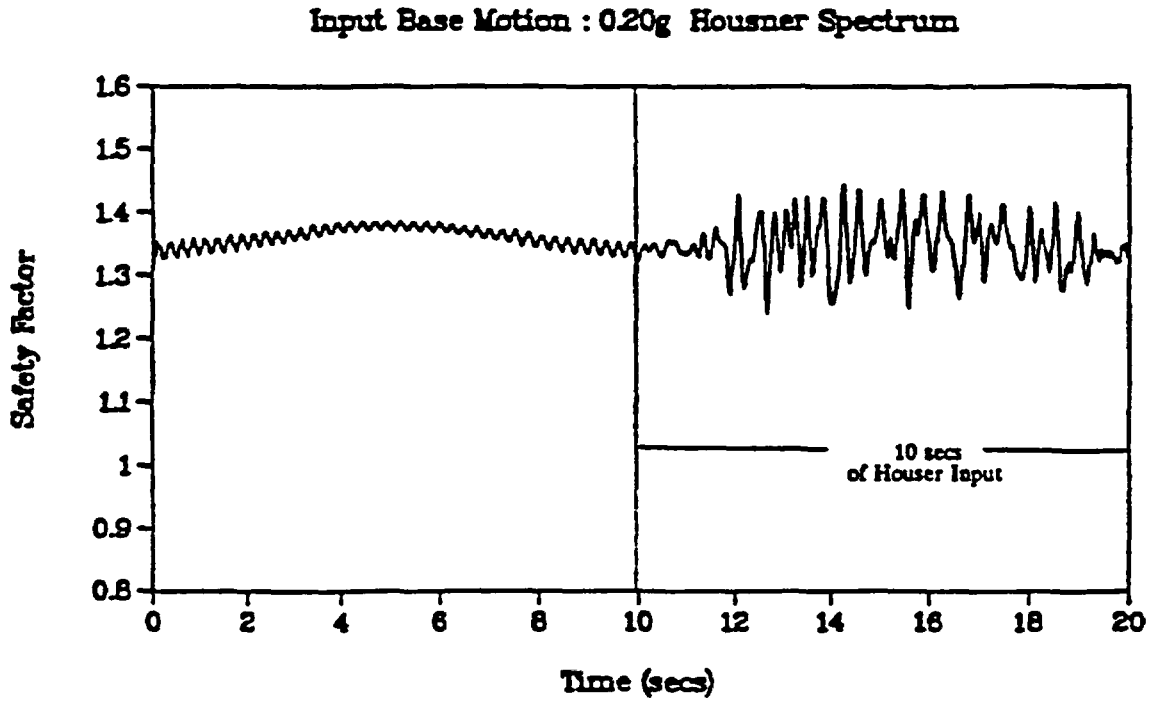


FIG. III.6: TEMPORAL VARIATION OF SAFETY FACTOR
SHALLOW FAILURE SURFACE/NORTH SLOPE

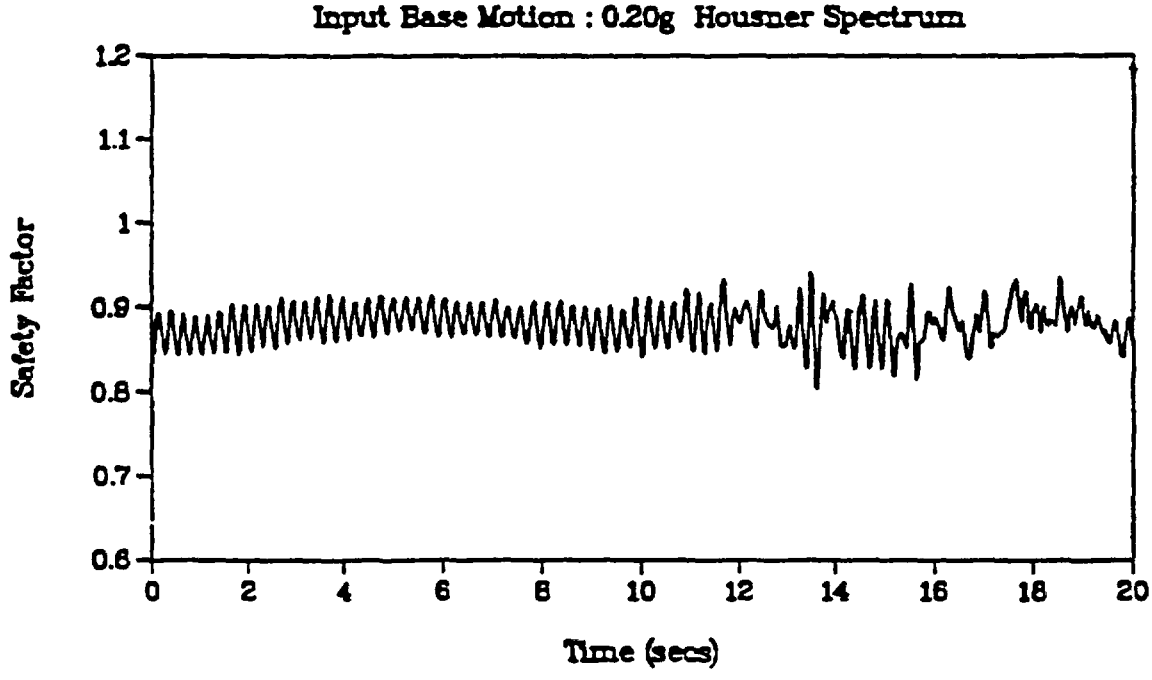


FIG. III.7: TEMPORAL VARIATION OF SAFETY FACTOR OVER FAILURE SURFACE PENETRATING INTO LOWER FOUNDATION ZONE WHICH HAS LIQUEFIED (fr. angle = 0)

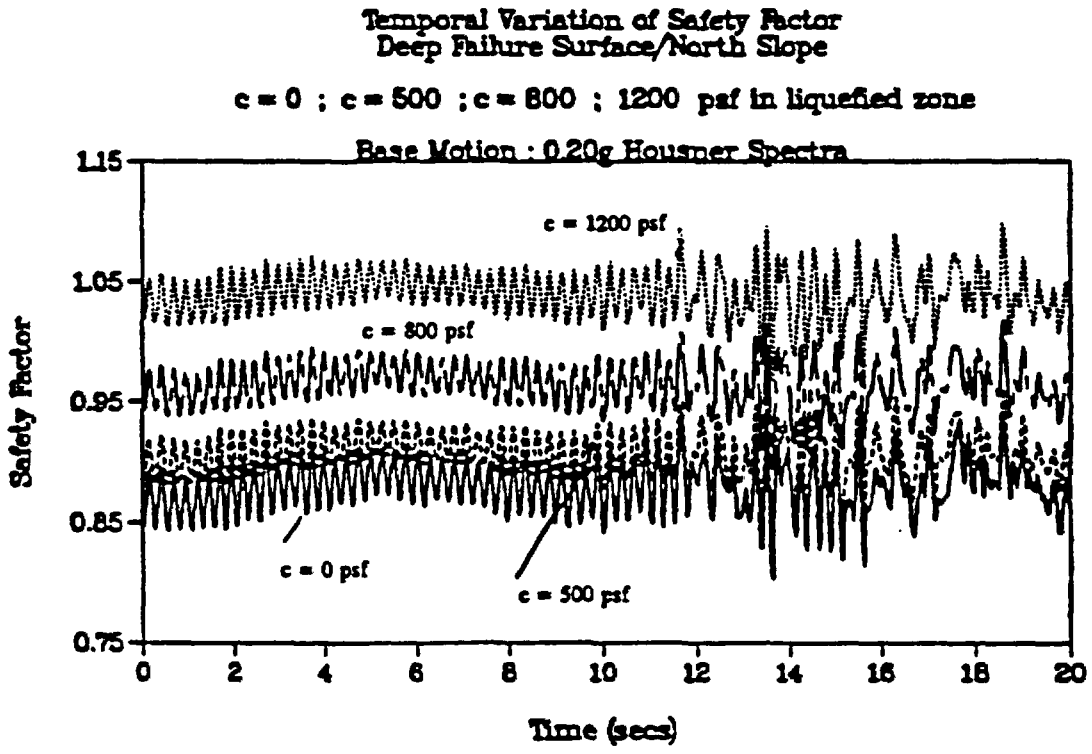


FIG. III.8: STABILITY OF DEEP SLOPE WITH RESIDUAL STRENGTH IN THE LIQUEFIED ZONE

APPENDIX IV TO ATTACHMENT 1

EAST-WEST SLOPE STABILITY ANALYSIS BASED ON QUASI-STATIC "REAME" ANALYSIS

The stability of the E-W slope is investigated using the "REAME" computer program. The analysis is based upon the simplified Bishop slip circle failure analysis. A simplified slope configuration was used for the analysis with the objective of obtaining conservative estimates of slope stability due to assumed liquefaction conditions in the loose to medium dense sands at the level of the ground water table.

As indicated in Fig. IV.1, the simplified slope condition assumed consists of a slope of 1 on 10 extending from the edge of the pad to the lake's edge. The liquefied zone is assumed to be 15 feet thick (as assumed in NS slope analyses) extending from the top of the GWT at a depth of 34 feet below the pad. This loose zone is assumed to extend uniformly from this elevation to the top of the water at the lake's edge, a most conservative configuration. It should be noted that for the purpose of the analysis, the elevation of the lake was arbitrarily set at Elev. 60.

Again assuming fully developed liquified conditions throughout this soft zone from the pad to the water's edge, the shear strength (cohesion and friction angle) are assumed to be zero. It is obvious that for this assumed condition that the slope becomes unstable. Fig. IV.1 indicates that a local failure condition will develop at the toe of the slope under seismic loading of 0.2 g, while Fig. IV.2 indicates similar behavior under static conditions post liquefaction.

Fig. IV.3 indicates that the safety factor increases to above unity for residual strengths of 600 psf, with the critical surface becoming a deeper circle starting further up the slope away from the toe. Fig. IV.4 indicates that the required residual strength to provide stability decreases as the thickness of the soft zone decreases.

Figs. IV.5, 6, 7 indicate similar behavior for deeper failure surfaces extending from the top of the slope. The thinner the soft zone, the lower the residual strengths needed to prevent failure of the circular failure surfaces. If a wedge analysis is used rather than the circular failure, a residual strength of 300 to 400 psf is required for quasi-static stability.

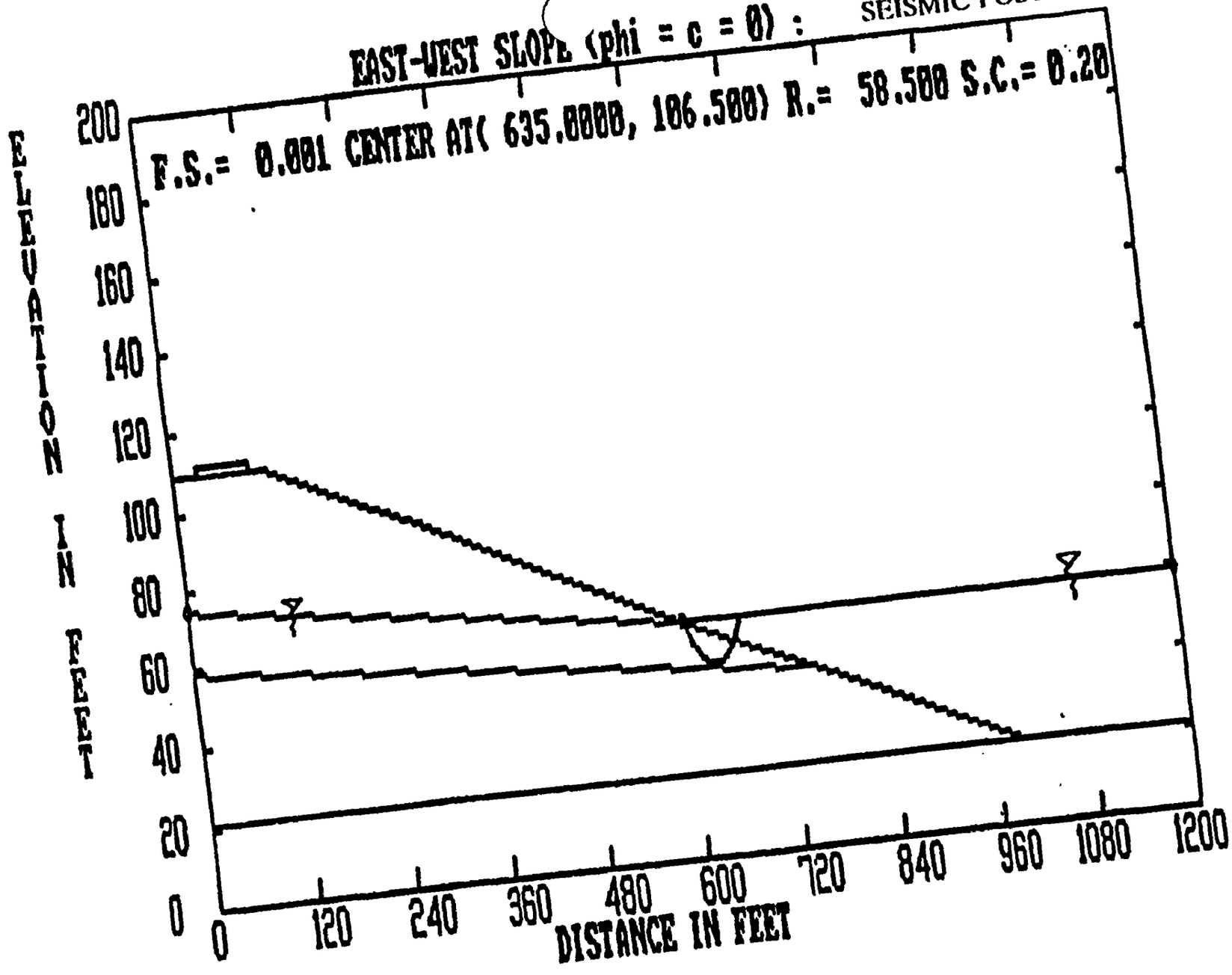

EAST-WEST SLOPE (phi = c = 0) :
F.S.= 0.001 CENTER AT(635.0000, 106.500) R.= 58.500 S.C.= 0.20
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET

FIG. IV.1

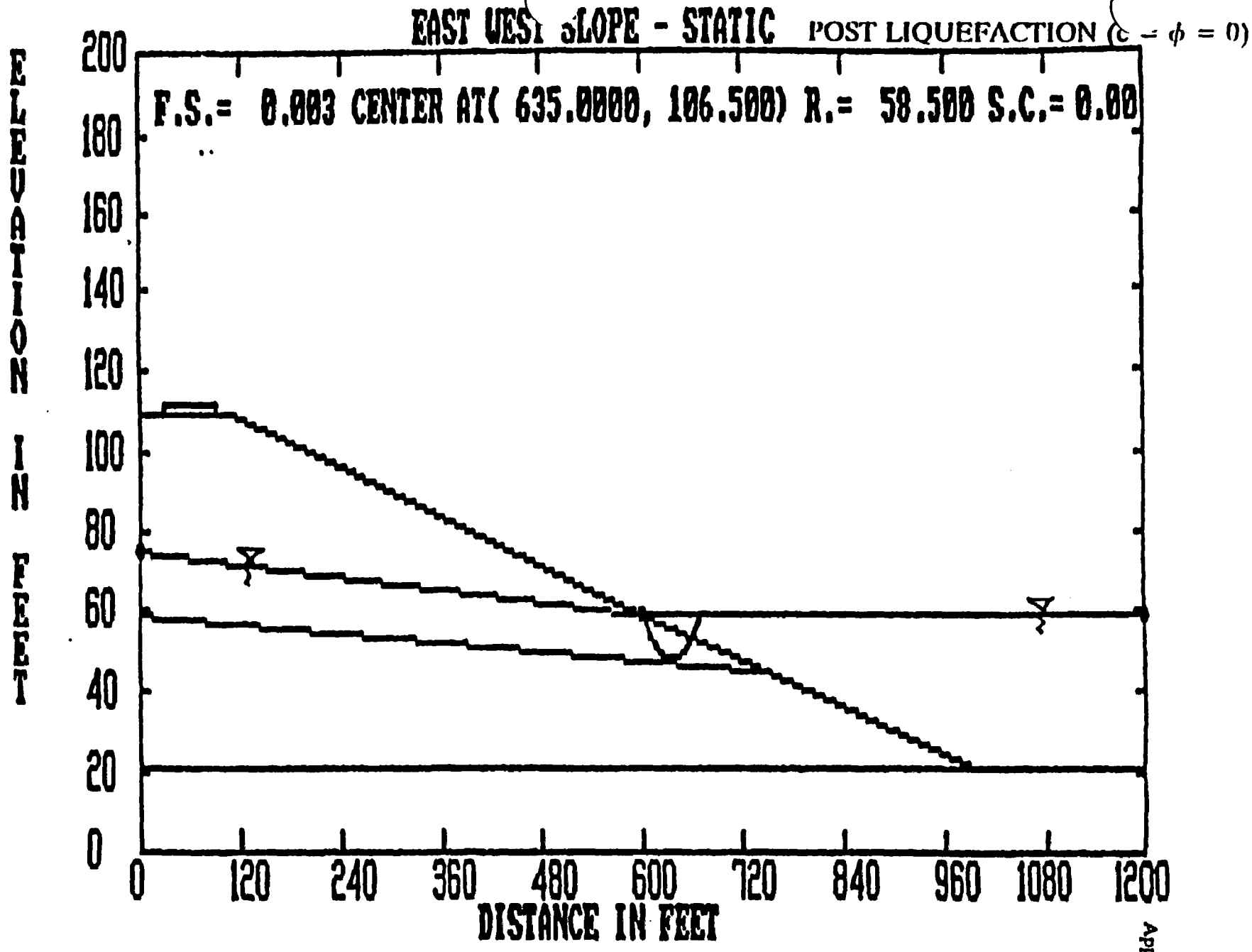

EAST WEST SLOPE - STATIC POST LIQUEFACTION (c = φ = 0)
F.S.= 0.003 CENTER AT(635.0000, 106.500) R.= 58.500 S.C.= 0.00
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET

FIG. IV.2

SEISMIC POST-LIQUE CTION
Cohesion= psf $\phi^{\circ}=0^{\circ}$

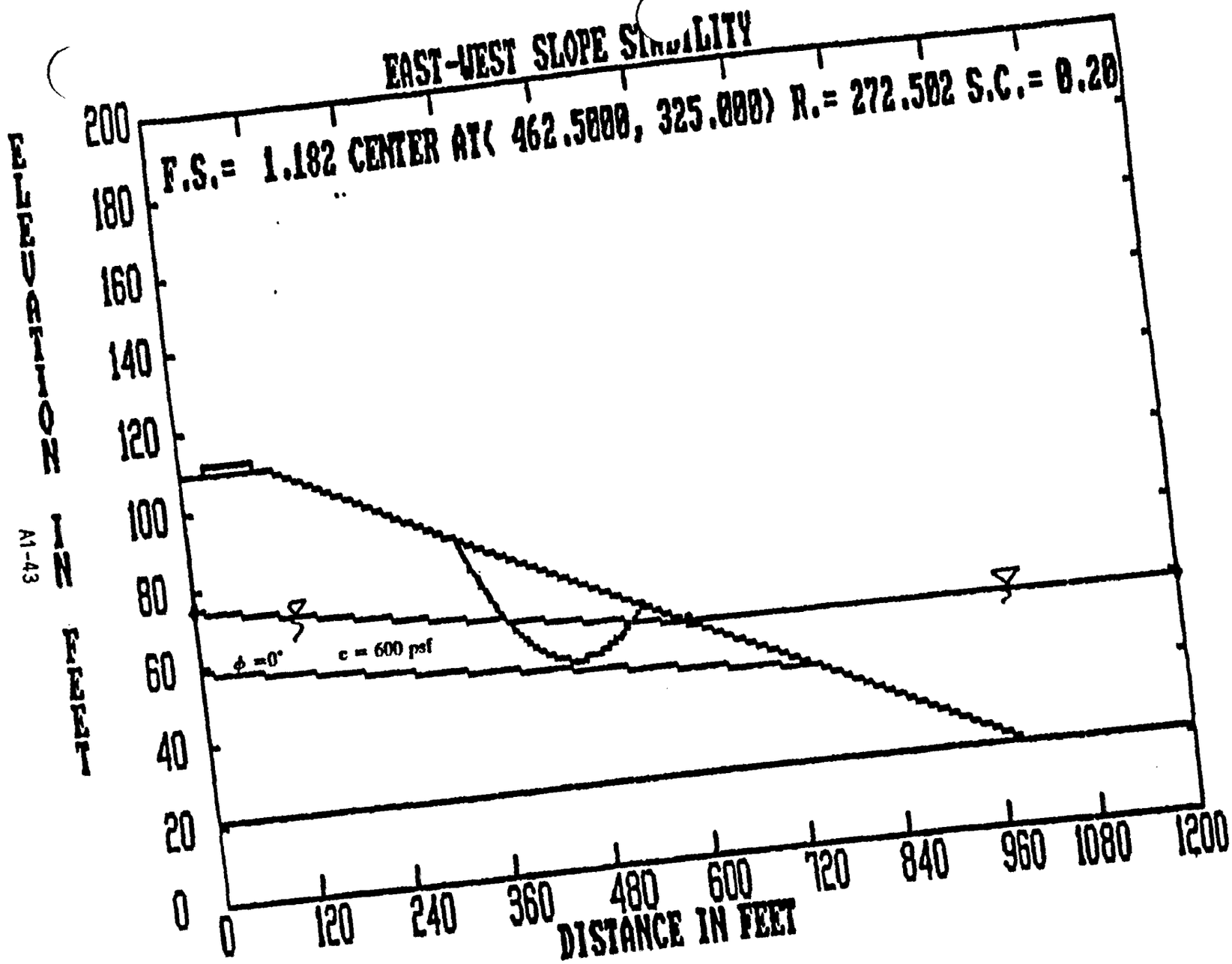


FIG. IV.3

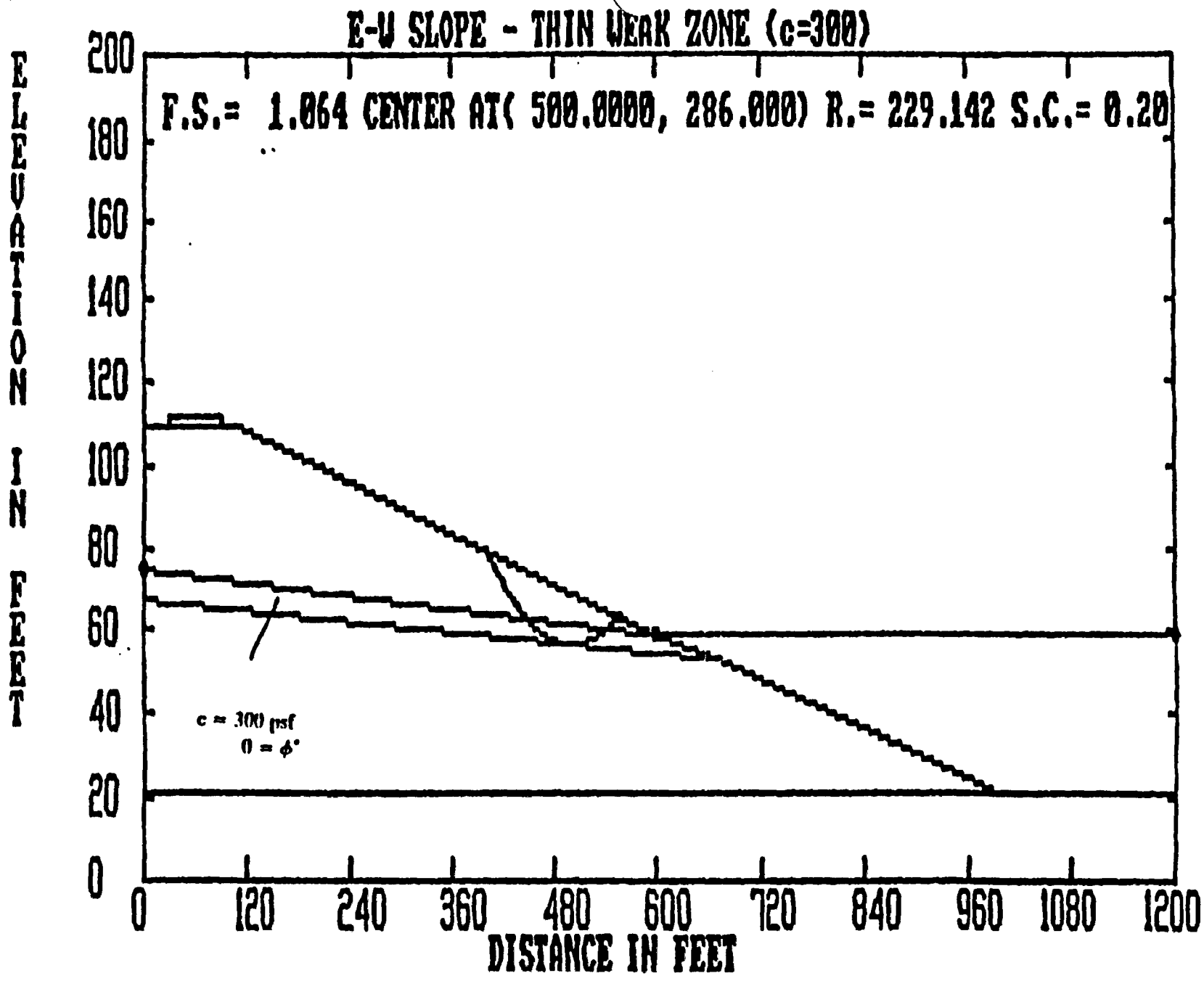

E-W SLOPE - THIN WEAK ZONE (c=300)
F.S.= 1.064 CENTER AT(500.0000, 286.000) R.= 229.142 S.C.= 0.20
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
c = 300 psf
0 = φ°
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET

FIG. IV.4

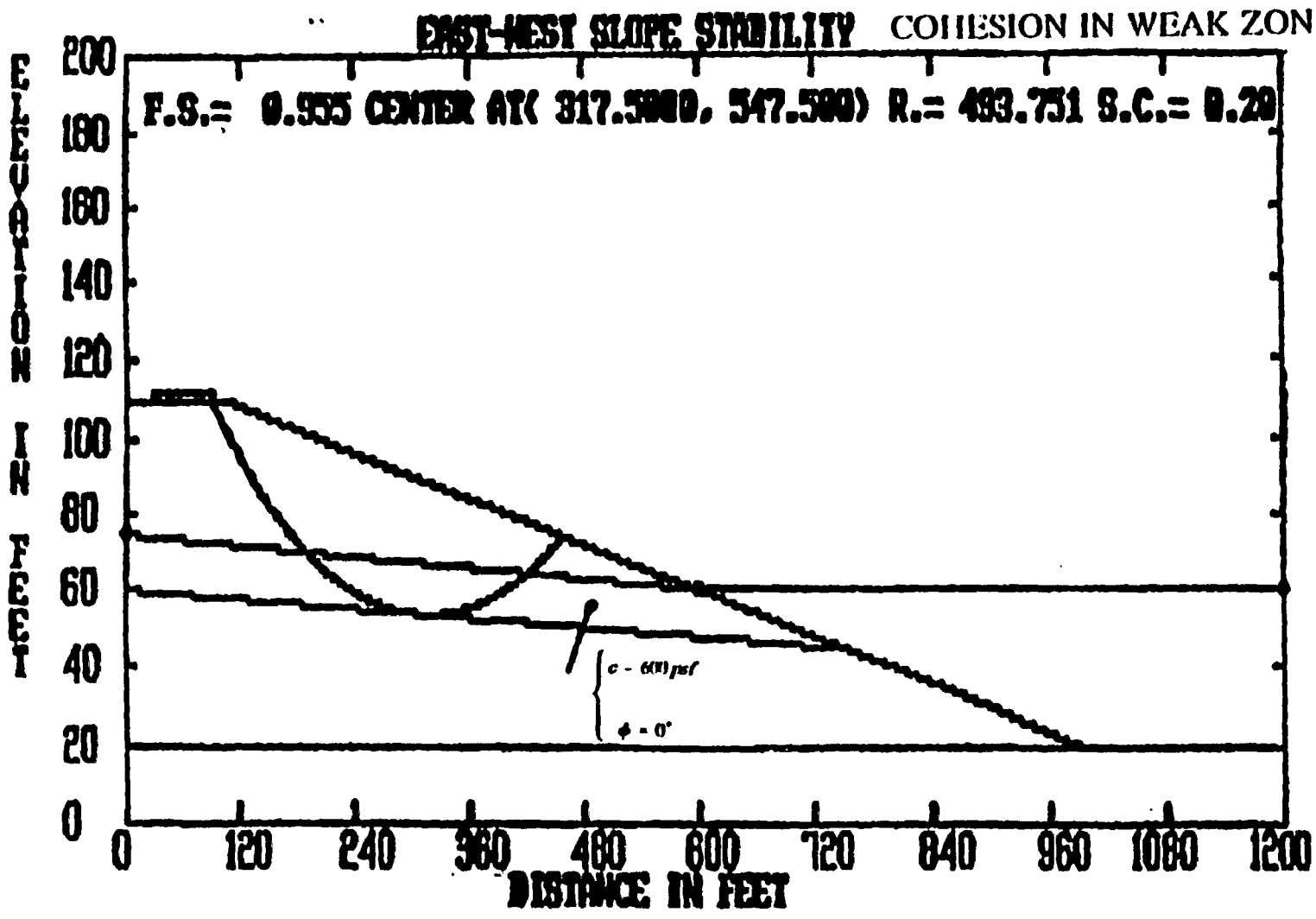

EAST-WEST SLOPE STABILITY COHESION IN WEAK ZONE = 600 psf
F.S.= 0.955 CENTER AT(317.5000, 547.500) R.= 493.751 S.C.= 0.20
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET
c - 600 psf
φ - 0°

FIG. IV.5

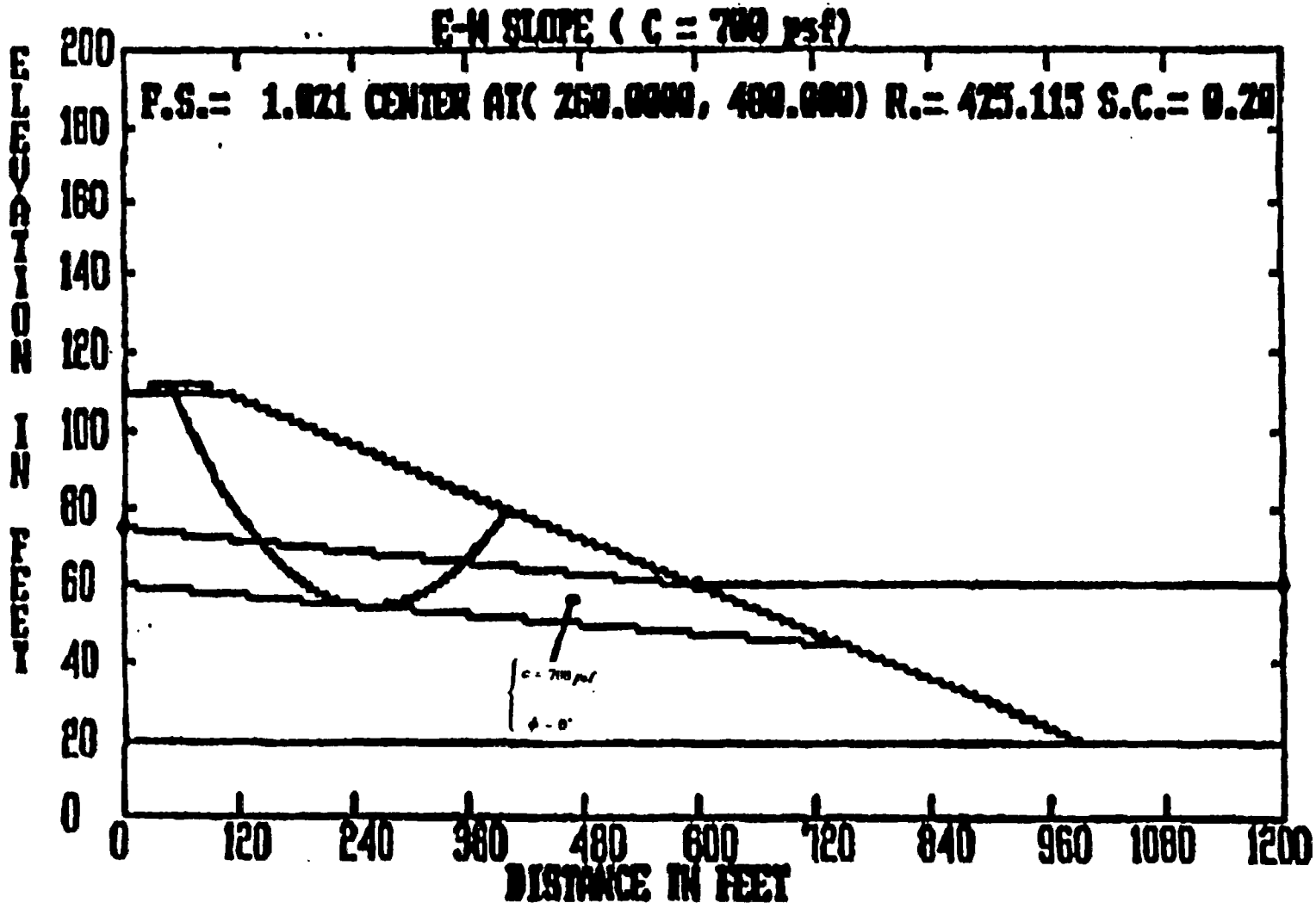

E-W SLOPE (C = 700 psf)
F.S.= 1.021 CENTER AT(250.0000, 400.000) R.= 425.115 S.C.= 0.20
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET
c = 700 psf
φ = 0°

FIG. IV.6

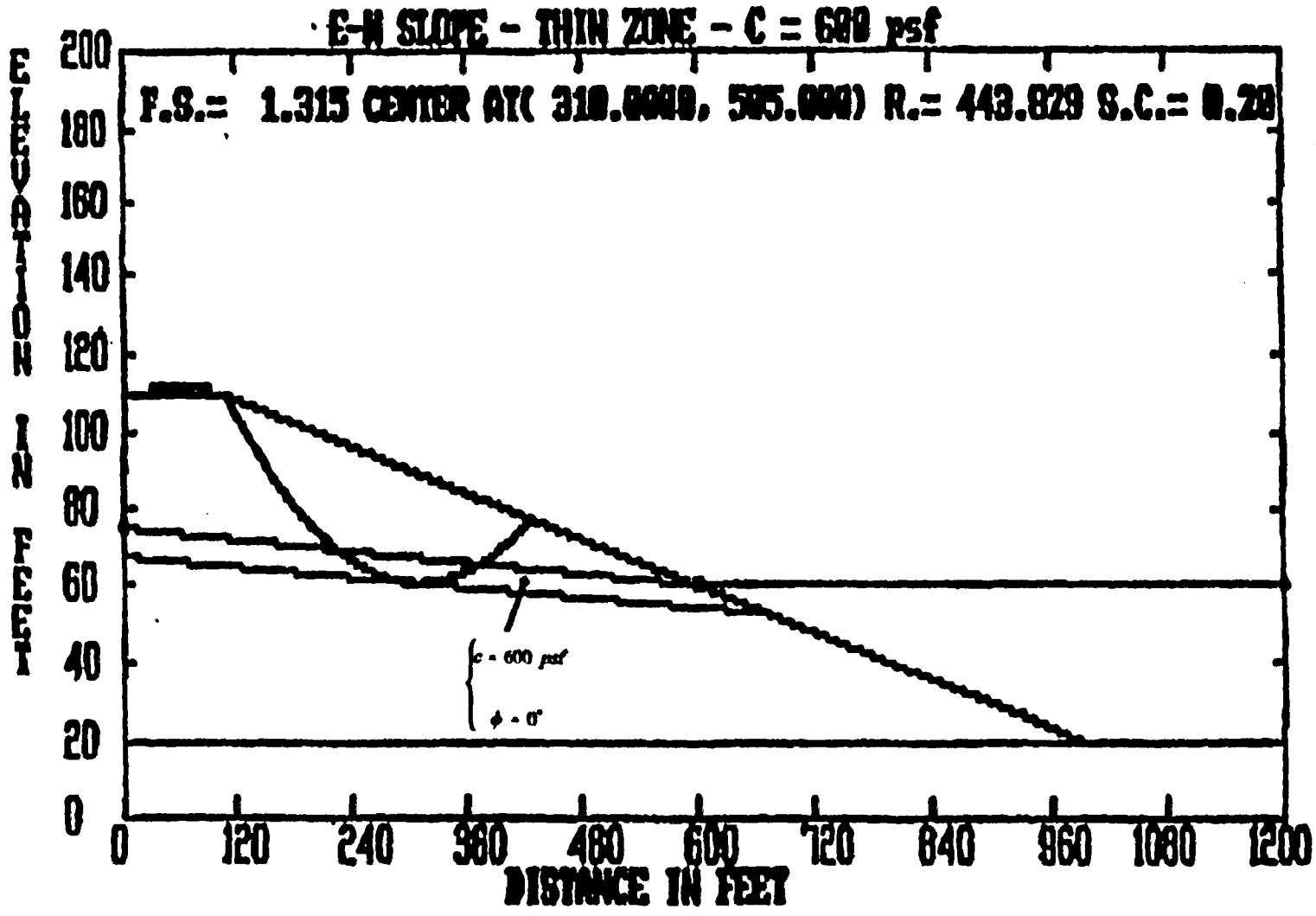

E-W SLOPE - THIN ZONE - C = 600 psf
F.S.= 1.315 CENTER AT(310.0000, 585.000) R.= 443.829 S.C.= 0.20
ELEVATION IN FEET
200
180
160
140
120
100
80
60
40
20
0
0
120
240
360
480
600
720
840
960
1080
1200
DISTANCE IN FEET
c = 600 psf
φ = 0°

FIG. IV.7

ATTACHMENT 2

Answers to Public Comments and Questions

Regarding

Palisades ISFSI

The first 40 comments and questions that follow come from the public meeting held on May 23, 1994. The page numbers in parentheses note where to locate the comment or question in the official transcript. See Attachment 3 for the official transcript.

Comments and questions 41 through 57 were taken from correspondence sent to the NRC by members of the public. These letters are in the NRC Public Document Room, 2121 L Street, N.W., Washington, D.C. 20037, and are available to the public.

From 5/23/94 public meeting transcript

1. "[T]he agency has in effect conceded a central issue of our lawsuit. That is, this is a unique environment of great importance to many people, and should have had a site specific evaluation" (p. 59).

Answer: The commenter appears to misunderstand that NRC's review in no way conflicts with its position in the current lawsuit.

As stated in NRC correspondence to Dr. Mary Sinclair, the NRC determined to independently review questions about the Palisades ISFSI pad (as documented herein) mainly because of the allegations raised by Dr. Sinclair.

After issuing the final rule approving the VSC-24 storage cask for use under the NRC general license, NRC received allegations from Dr. Sinclair reflecting a concern that, because the Palisades ISFSI pad was constructed on sand dunes near Lake Michigan, it could be damaged by shoreline erosion. While NRC reviewed the allegations, an NRC technical staff member, Dr. Ross Landsman, NRC Region III, also raised questions about the stability of the soil and slopes next to the ISFSI pad under seismic conditions. The licensee responded by conducting additional analyses related to the concerns about the storage pad in March 1994. In March 1994, NRC began to independently review the Palisades ISFSI pad to address the questions raised and to better review the licensee's additional analyses.

This NRC action confirmed rather than undercut the principle underlying the NRC rule under attack in the Sixth Circuit case: the rugged design of the cask, intended to enable it to withstand an extremely broad spectrum of environmental conditions, should

make it nearly impervious to any event even remotely likely to occur at Palisades or at any other licensed nuclear plant within the environmental parameters of the cask design. Thus, this review did not detract from NRC's position that generic approval is an appropriate procedure, and that NRC need not analyze each site before casks are installed. In this instance as in many others, the NRC, in its role as regulator and overseer, was following its customary and prudent practice of examining thoroughly allegations received concerning the safety of already licensed and operating facilities (see 10 CFR 2.206).

2. Mr. Adamkus letter to Mr. Taylor of NRC in 12/93 asking for more environmental info...."This action supports our position that a site specific NEPA review was required" (p. 59).

Answer: This comment apparently refers to a December 30, 1993, letter from an EPA Regional Administrator, asking for copies of the NRC's environmental documents on dry cask storage in order to review them. On January 30, 1994, NRC submitted the requested documents and an explanatory letter. Since that date, there has been no further correspondence between EPA and NRC on the subject. Copies of the above-referenced letters between EPA and NRC are enclosed as Appendix A.

3. Army Corps of Engineers report on Great Lake shoreline and in depth study authorized by State of Michigan Low Level Radioactive Waste Authority in 1988 conclusions appear to directly contradict the findings of NRC's draft safety assessment. These facts indicate that a public hearing should be in order (pp. 59, 60).

Answer: NRC staff reviewed the Army Corps of Engineers report. The NRC staff has concluded that the ISFSI is adequately protected from the effects of wave action and shoreline erosion. This conclusion is based on an evaluation of the location and elevation of the ISFSI facility and the protection provided by a revetment and beach area fronting the plant and the ISFSI pad area.

It is important to note that the reports discuss and delineate general areas of erosion and indicate that several factors contribute to both shoreline stability and shoreline erosion. The reports show that high-risk erosion areas include those that are not protected by armored revetments and those where dunes are located very close to the shoreline. For the Palisades plant area, an armored revetment is provided to protect the immediate shoreline fronting the plant site. Directly north of the plant, a wide beach separates the dunes from the shoreline. Both of these factors contribute to the stability of the shoreline and the dunes.

The comment also references a study by the State of Michigan on the suitability of Palisades as a low level waste (LLW) site, "Final Report, An Evaluation of the Four Licensed and Operating Nuclear Power Plant Sites in Michigan for Co-Location of a Low-Level Radioactive Waste Isolation Facility," May 24, 1988. The commenter appears to misunderstand the scope of that study, which addressed only LLW disposal. It did not cover any site and did not contain any conclusions for interim spent fuel storage in dry casks. Moreover, the conclusions regarding LLW disposal sites were based on the conservative assumption that engineered systems will fail and that natural barriers must be relied on, whereas cask approvals are based on the assumption that natural barriers are irrelevant, since reliance is placed solely on engineered systems.

For example, the Michigan researchers stated: "These locations offer little or no *natural protection to prevent release of radioactive waste materials to the environment* (p. ES-2). This statement could be important for LLW disposal because a safe LLW disposal system, as defined in NRC regulations, could include a disposal site that has the ability to contain any waste that is released. However, this statement is not important for dry cask storage because NRC regulations define waste containment to be a function of the engineered cask and not of the site conditions.

Similarly, consider the criterion reflected in the following statement on page 1-2 of the study: "Generally speaking, a site for a low-level radioactive waste disposal facility should have *natural features such that waste breaching any of the engineered barriers would be naturally isolated and contained at the site.*" This siting criterion is important for a safe means of LLW disposal, which NRC regulations require to safely contain the waste for as long as 500 years. NRC regulations assume that active institutional controls cannot be relied on with confidence to monitor, deter, or remediate a breach of engineered barriers for 500 years; therefore, the site itself must have features to contain the LLW. However, this siting criterion has no significance for dry cask storage. The dry cask system is a robustly designed and well-engineered system which itself will prevent release of any waste. It can also be monitored and maintained over the 20-year license term to ensure safe performance and complete containment of the waste throughout its licensed life.

A statement on page 3-1 of the Michigan study shows that the study's conclusions cannot be applied to dry casks: "The criteria [used for site evaluation] are formulated *assuming no engineered containment*. Based on the siting criteria, the potential site selected is to serve as a natural means of isolation for the waste." The assumption in the statement is exactly the opposite of the assumption that applies under NRC regulations for the licensing of dry cask storage. Under NRC regulations, dry cask storage can only be licensed based on engineering design of the

robust cask; since containment is to be provided by the cask itself, it is assumed that the site will not isolate waste.

4. "In addition, we can now raise questions about whether the VSC-24 cask was indeed a generic cask as is claimed and as it was licensed. We can point to the fact that at both reactor sites where there are plans to use the VSC-24, at Point Beach and Arkansas 1, changes must be made to the cask design in order to be able to use it" (p. 61).

Answer: Although the VSC-24 cask is indeed a generic cask, it does not necessarily follow from that statement that the cask can be used to store the particular spent fuel assemblies at every reactor site. For example, the type or size of the fuel at a particular plant might not conform to the conditions in the cask certificate of compliance (C of C) which, in turn, would preclude its use. The changes requested by the cask vendor, Sierra Nuclear Corporation (SNC), to the safety analysis report (SAR) and C of C for the VSC-24 cask to accommodate longer versions of Combustion Engineering (CE) 16 X 16 fuel assemblies generally apply to all reactors that use the longer CE fuel, one of which is the Arkansas Nuclear One (ANO) Unit 2 plant. The VSC-24 cask, as approved, could still be used at the ANO Unit 1 plant. Likewise, the changes requested by SNC to accommodate control components, while useful at the ANO site, would also generally apply to those pressurized-water reactors (PWR) that also want to store control components integral to the fuel assemblies. Although changes to the C of C are not necessarily required for use of the VSC-24 cask at the Point Beach nuclear plant, SNC requested certain changes to clarify the requirements for non-critical cask handling operations. Those changes would apply to any utility using the VSC-24 cask.

5. "[T]he casks were licensed before all the safety questions were resolved" (p. 62).

Answer: All safety considerations for approving the VSC-24 cask for use under the general license provisions of Part 72 were resolved before NRC issued the C of C. Subsequent to the NRC's approval, the vendor proposed changes to the cask design for NRC approval, to allow use of the cask for additional types of spent fuel. The NRC staff is evaluating the proposed changes and will resolve all safety considerations associated with them before issuing an amended C of C.

6. "With so many requests for changes in the design to suit unique site specific requirements at other reactor sites, it plainly does not have a generic cask for high level nuclear waste disposal and for a good reason. U.S. reactors are of so many different designs and they have different fuel types and vary so much in how they are constructed on site, that a generic cask system apparently cannot be designed to accommodate all these differences. Nor do we have a generic environment in this country" (p. 62).

Answer: The VSC-24 is a generic cask, and design bases of the cask envelop site conditions at most reactor sites in the U.S. However, as noted in the response to comments 4 and 5, the vendor has proposed certain changes to address spent fuel specifications and handling operations. The changes do not propose to modify the site conditions enveloped by the cask design.

7. "[T]hese casks have never been built or tested before....they were built before there was a certificate of compliance which would have established their criteria for construction, and this is one of the aspects of our lawsuit" (p.63).

Answer: NRC addressed these issues in the response to comments 37 and 60 in the final rule that approved the VSC-24 cask (58 FR 17948) and are included here for completeness.

Response to Comment 37

Although the data available to support certification of the VSC-24 cask do not include results of full-scale tests, the data are more than sufficient to show that a licensee's use of the VSC-24 cask will not place power plant workers, the public, or the environment at any undue risk. Also the conditions of use for the VSC-24 cask in the certificate of compliance ensure adequate protection of the workers, the public, and the environment. Further, the VSC-24 cask has been designed and will be fabricated to well-established criteria of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* and the *American Concrete Institute Code*. The construction materials in the cask have well-known and documented properties to provide the necessary structural strength and radiation shielding to meet regulatory requirements.

While the NRC has not relied on testing of the VSC-17 cask (a smaller version of the VSC-24 cask design) for approval of the VSC-24 cask, the Department of Energy (DOE) tested the VSC-17 cask at its Idaho National Engineering Laboratory and found that ventilated storage cask technology can safely store spent fuel. The report, "Performance Testing and Analysis of the VSC-17 Ventilated Concrete Cask," EPRI TR-100305, May 1992, concluded that the VSC-17 cask can be safely used at reactor sites. While the VSC-24 cask approval does not rely on the VSC-17 cask, the designs are similar and many parallels in design and function can be drawn. Thus, although the commenter's observation that the VSC-24 had not been fully tested under climatic conditions is technically correct, the cask has been designed for ambient temperature extremes from -40°C (-40°F) to over 37.78°C (100°F) and meets the ASME and ACI requirements.

Response to Comment 60

The NRC granted Sierra Nuclear Corporation's request for an exemption to fabricate a limited number of the casks at its

financial risk before NRC issued the certificate of compliance under its NRC-approved quality assurance program. The staff reviewed the SAR for the VSC-24 cask and concluded that beginning fabrication before receiving the certificate of compliance would pose no undue risk to public health and safety. Licensees cannot use these casks until they satisfactorily complete NRC's certification process.

8. "Is there any research being done which could utilize the spent fuel for a use[ful] purpose thus eliminating the need for casks all together" (p. 65)?

Answer: See transcript, line 22 of p. 65 to line 17 of p. 66.

9. "At one point we heard that the pads was [three feet]---and then we heard it was between two and three feet thick." (p. 66)

Answer: See transcript, line 11 of p. 67 to line 4 of p. 68.

10. "[I]f you would like public participation which I am not really sure about, you need to make sure that we know where the meeting is....You don't get up and show charts and acronyms and expect people to understand. I guess I am concluding that you didn't want us to understand" (p. 69).

Answer: The meeting time and location were published in a meeting notice on May 6, 1994. All meeting notices are available in the NRC and local public document rooms, as well as telephone recordings of NRC public meetings. We regret that some persons might not have learned of the meeting in time to attend, or might have needed more specific directions on how to get to the meeting location. The NRC presenters for the May 23 public meeting were trained engineers accustomed to using technical terminology when discussing technical issues. Time was allotted at the end of the meeting for people to ask questions on points they may not have fully understood. For future public meetings, the NRC staff will try to better communicate technical issues in lay terms and where possible, use everyday analogies to provide appropriate insights.

11. "Is there a list of parties consulted somewhere that I can get hold of?" "That would be a complete list of parties consulted in this six week review?" "Was public comment sought during any time of the review" (pp. 72, 73)? "[W]as it solicited? Was it asked for and solicited" (p. 74)?

Answer: See transcript, lines 2 to 15 of p. 73 and line 21 of p. 74 to line 9 of p. 75.

12. "What documents were reviewed, and is there a public list of the documents that are reviewed" (p. 75)?

Answer: See transcript, lines 13 to 19 of p. 75.

13. "Did the 1979 EIS address erosion and earthquakes" (p. 75)?

Answer: Environmental impact statements do not address the safety of constructed facilities; rather, they address the impact of constructed facilities on the environment. The safety analysis report, written before the construction permit was issued for the Palisades plant review, and the safety evaluation report, written before the operating license was issued for the Palisades plant review, both addressed the site geology, foundation conditions, hydrology, and earthquake seismology.

In Sections 4.1.1.2 and 4.1.1.3 of the "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel," August 1979, NUREG-0574 (GEIS), NRC discussed the effects that storage facilities could have on erosion:

> Earth-moving operations expose soils to erosion....Various measures, such as...topsoil stockpiling, revegetation, etc. are usually implemented either to reduce initial impacts or to facilitate rapid recovery." The construction and operation of [the facility]...involve excavation and replacement of soils. Occasional...soil erosion problems have been encountered. Soil disposal areas have been contoured to conform to existing topography and reseeded so that the...erosion impacts are reduced....Thus the spent fuel storage facility does not appear to have any ecological impact on the surface or groundwater environment.

Appendices B and D of the GEIS address earthquakes as general design criteria for storage facilities. NRC also analyzed a range of accidents and natural phenomena events in Section 4.2.3.

14. "[H]ow do your findings of this six week study relate to your other, earlier, environmental assessments" (p. 76)?

Answer: As stated at the meeting (see transcript, line 23 of p.76 through line 2 of p.77), the "Draft Independent NRC Staff Assessment of Dry Spent Fuel Cask Storage Facility at Palisades Nuclear Power Plant Site" is a safety review to address an allegation of possible safety problems and had no direct connection to earlier environmental assessments. Nothing in this final report prompted the staff to reassess its previous environmental assessments or safety evaluations related to the approval of the VSC-24 cask.

15. "I know someone who, in the last ten years, lost 70 of their 158 frontage....You have to look at what is happening on the shore line and relate that to Palisades."... "I would urge you to make a real commitment to public participation" (p. 77).

Answer: See transcript, lines 16-23 of p. 77.

As discussed in the staff's Final Safety Assessment (FSA), some shoreline erosion should be expected to occur in many areas along Lake Michigan, particularly those areas in which protection by revetment, or otherwise, is not provided or where dunes are close to the shoreline. During a site visit to the plant, the NRC staff observed several areas of erosion in the Palisades Park area. Based on those observations, it appeared that most of the damage occurred to dunes and beaches located within about 61 meters (200 feet) of the shoreline. Apparently, the dunes were damaged because large waves were able to break directly against them. As indicated in the FSA, the staff has concluded that large breaking waves would not reach the ISFSI area. This conclusion is based on the location and elevation of the ISFSI, which is 137.25 meters (450 feet) from the shoreline, 12.2 meters (40 feet) above the highest recorded lake level, and 9.15 meters (30 feet) above the estimated probable maximum lake level. Large waves and lake levels would not reach the ISFSI facility due to this elevation difference. Large waves would break either on the revetment in the immediate plant area or in the beach area to the north of the plant.

16. "Either abide by your generic system or do a full fledged environmental impact statement that would certainly take a lot more time and would involve the public, as it should" (p. 78)...."What is wrong with doing a complete environmental impact statement and really involving the public as they need to be. People are just tired of having decisions handed down to them" (p. 79).

Answer: The Palisades ISFSI uses the VSC-24 storage cask design, which was reviewed and approved by NRC in a final rule issued in April 1993. That final rule is in litigation before the U.S. Court of Appeals for the Sixth Circuit in Cincinnati. One of the issues contested before the court is the NRC environmental review. Therefore, this matter will likely be resolved by the courts.

As a regulator, NRC has the obligation to address safety concerns about a licensed activity under NRC jurisdiction irrespective of the process used initially to license the activity. Therefore, we cannot accept the commenter's "either-or" premise which would have NRC ignore allegations about the safety of a cask at a specific site if the cask was installed at the site under a generic approval process.

17. "What physical protection is provided from the more likely unnatural hazard similar to bombing of the World Trade Center during the long term anticipated "temporary" storage of the VSC-24 casks at Palisades" (p. 79, 80)? Other comments from lines 3 to 17 of p. 80 of the transcript deal with the protection against an attack on a cask by use of a car bomb, a shoulder-fired missile from land or from a boat, and bombs dropped from small aircraft.

Answer: Similar questions regarding postulated sabotage scenarios were raised during the rulemaking to amend 10 CFR Part 72 regarding storage of spent nuclear fuel at nuclear power reactor sites. Responses to these questions were included in the July 18, 1990, *Federal Register* notice (55 FR 29181) which was published with the final amendments. The response referred to an NRC-sponsored study done before the proposed amendments. The researchers considered a range of sabotage scenarios. After considering various technical approaches to radiological sabotage, and experiments and calculations, they concluded that radiological sabotage would be successful if carried out using large quantities of explosives, not a small bomb such as one that could be dropped from an airplane, and that the consequences to public health and safety would be low because most of the resultant contamination would be limited to the storage site. The researchers found that substantial protection against sabotage scenarios resulted in part from the inherent protection derived from the massiveness of the storage casks required to both shield from radiation and protect the spent fuel from earthquakes and tornado missiles. Thus, the security threat to be protected against is the possible protracted loss of control of the storage area. For that reason, security protection requirements (1) provide for early detection of malevolent moves against the storage site and (2) give a means to quickly summon response forces to protect against a loss of control of the storage area. Security at Palisades for the spent fuel storage casks ensures both early detection and ability for quick response. Although someone might be able to approach the protected area bounding the dry spent fuel storage area, any attempt to penetrate the protected area would be detected and met with immediate response.

The staff regularly consults with law enforcement agencies and intelligence-gathering agencies to obtain their views of any adversary groups interested in radiological sabotage of commercial nuclear facilities. None of the information the staff has collected confirms the presence of a domestic threat to dry storage facilities.

NRC recently approved a final rulemaking to require power reactor licensees to implement specific measures to protect their facilities against vehicle intrusion and a vehicle bomb. This rulemaking does not apply to ISFSIs. The staff believes that the nature of the fuel and the degree of protection afforded by the approved storage means for spent fuel give adequate protection. However, as a matter of prudence, staff is studying the consequences of a vehicle bomb detonated near an ISFSI. The results of the study will assist the staff in determining whether any specific measures to protect ISFSIs from the threat of a vehicle bomb are needed.

Questions 18-21 were directed to Consumers Power Company

18. "[N]uclear power is not safe...Why has this company not explored a comprehensive plan to reduce electric consumption and thus eliminate the need for this plant and every other plant like it" (p. 82)?

Answer: **See transcript, lines 9 to 18 of p. 82.**

19. "How does Consumers Power explain the series of mishaps and accidents resulting in serious violations of state and federal law which have occurred continuously since it was built" (p. 82)?

Answer: **See transcript, lines 2 of p. 83 to line 15 of p. 84.**

20. "What will Consumers Power Company do if the Federal Government does not establish a national nuclear dump site" (p. 85)?

Answer: **See transcript, lines 11 to 19 of p. 85.**

21. "How much has these casks cost, who is paying for it, who will pay for their eventual removal and transportation" (p. 87)?

Answer: **See transcript, lines 10 to 16 of p. 87.**

22. "What can we as citizens do to move the government to create a site far away from the water" (p. 87).

Answer: **See transcript, line 24 of p. 96 to line 25 of p. 98.**

23. "I have concerns there, about the wind erosion and potential blow out as the Army Corps of Engineers have suggested" (p. 88).

Answer: **The staff agrees that some wind erosion could occur in the immediate area of the ISFSI. However, the dunes in the area are considered to be relatively stable, due to the presence of heavy vegetation. Regardless of the amount of wind-deposited material in the vicinity of the ISFSI, the licensee will rely on a comprehensive monitoring program to detect the presence of sediments and wind-blown material and has committed to remove those sediments in a timely manner. Additional information on the staff analysis of the licensee's monitoring programs may be found in the Final Safety Assessment.**

24. "I walked up to the fence and took photographs and nobody was around."..."There is nothing preventing somebody from coming in on a boat and taking a TOW missile or what have you. There are literally hundreds of these things floating in the underground market" (p. 89).

    "Somebody with a hang glider could go overhead and drop a bomb. I mean somebody in a little plane" (p. 94).

    "Concrete is very easily [sic] for somebody to destroy with a bomb or any kind of mild explosive....it would be real hard to get a crew in there to do repairs" (p. 117)

Answer: See also answer to question 17.

The staff addressed radiological sabotage in response to comment 33 in the final rule that approved the VSC-24 cask (58 FR 17948) and the response, in part, is included here again for completeness.

NRC reviewed issues regarding possible radiological sabotage of storage casks at reactor site independent spent fuel storage installations (ISFSIs) in the 1990 rulemaking that added Subparts K and L to 10 CFR Part 72 (55 FR 29181). NRC regulations in 10 CFR Part 72 establish physical protection and security requirements for an ISFSI within the owner-controlled area of a licensed power reactor site. Section 72.212(b)(5) requires that the spent fuel in the ISFSI be protected against the design-basis threat for radiological sabotage using provisions and requirements comparable to those for other spent fuel at the associated reactor subject to certain additional conditions and exceptions described in 10 CFR 72.212.

Each utility licensed to have an ISFSI at its reactor site is required to write security plans and install a security system to protect against unauthorized activities that could pose an unreasonable risk to the public health and safety. The security systems at an ISFSI and its associated reactor are similar in design features to ensure the detection and assessment of unauthorized activities. All alarms at the ISFSI are monitored by the security alarm stations at the reactor site. Response to intrusion is required. Each ISFSI is periodically inspected by NRC and annually audited by the licensee to ensure that the security systems are operating within their design limits. The validity of the threat is continually reviewed, and NRC performs a formal evaluation every 6 months.

NRC has anticipated actions a licensee might take to recover from an accident that degrades the concrete shielding or renders it inoperable. However, the ventilated concrete cask (VCC) is a massive concrete construction that would not likely be completely lost as a result of any event, such as sabotage. If an unlikely but credible event were to damage the VCC, little concrete would likely be lost because it is steel reinforced. If crushed or impacted by a wind-driven object, the concrete would be captured and contained by the reinforcing steel. However, if an unspecified event caused a large piece of the concrete shielding to be lost, the hazard to workers and the general public would be an increase in the surface area dose rate resulting from direct radiation emanating from the sealed multi-assembly sealed basket (MSB) inside the VCC. This condition would be detected during the normal daily inspections, after which, radiation protection personnel, or health physicists (HPs), would do the required radiological surveys. The HPs would approach the cask from a side that remains intact. If no side remained intact, the workers could use a portable shield for protection. Upon

evaluating the radiation survey data, the licensee would fabricate temporary shielding using lead blankets or sheet lead[1], which are generally available at all commercial nuclear facilities. Once the shielding was in place, the immediate threat to the public health and safety would be eliminated. The next step would be to repair or replace the damaged VCC by following special procedures for the specific conditions at the time.

25. "My question is to the people from Consumers Power. Why aren't you publicly and loudly raising hell with the NRC to get rid of your waste storage problem" (p. 95)?

Answer: See transcript, lines 5 to 15 of p. 96.

26. "To the NRC, is it a truly political problem that you can't use the sites out west. You own the land, you made the studies, why can't you move the waste out there" (p. 95)?

Answer: See transcript, line 24 of p. 96 to line 23 of p. 99.

27. "How is the waste moved" (p. 99)?

Answer: As indicated in the response at the meeting (see transcript, line 25 of p. 99 through line 7 of p. 100) the spent fuel would be moved off site in a certified transportation cask. Under the Nuclear Waste Policy Act of 1982 (NWPA) the Department of Energy (DOE) is responsible for removing the fuel from the reactor sites for further interim storage or disposal when a facility is available. In its work on the Civilian Waste Management Program, DOE is pursuing the multi-purpose canister (MPC) initiative and the development of the transportation capability that will meet its needs in the near term.

28. "You have not addressed how you are going to get the waste out and how you are going to transport it" (p. 101).

Answer: The general procedures for unloading the cask are addressed in the VSC-24 cask SAR, Chapter 8. See the answer to question 27 on how the waste is to be transported.

29. "[The] International Joint Commission for the Great Lakes for the United States and Canada has said in this last meeting they had, they said they recommend to both governments that they phase out all radionuclides that have a half life longer than eight weeks. Where does that put the nuclear power plants" (p. 102)?

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[1] The VCC is designed to hold the weight of the 100-ton transfer cask, so even in a damaged condition it is reasonable to assume that temporary shielding would have a negligible effect on the structural integrity of the cask.

Answer: **See transcript, line 24 of p. 102 to line 2 of p. 103. The staff does not have information regarding the recommendations from the International Joint Commission. However, the environmental assessments and NRC regulations state that dry cask storage under NRC license does not involve the release of any liquid radioactive effluents. Therefore, the casks are not a source of any radionuclide releases to the Great Lakes or to Lake Michigan.**

30. "I called the NRC and they said no, we can't help you. We have seen that study but we can't get you that study because it is not in the NRC document. That means there is a whole realm of studies that is not open to the public. I would challenge you to open your books and let the public in" (p. 105).

Answer: **All documents which form the basis of this NRC final safety assessment are in the NRC public document room and are available to the public.**

31. "First of all, if helium can leak out, what can leak in. Second of all, if the helium does leak out, what effect does that have on the casks" (p. 107)?..."What happens if water leaks in" (p. 108)?

Answer: **See transcript, line 6 of p. 107 to line 14 of p. 108.**

32. "You did say that you would not put any fuel into the cask that had known fuel leaks. What did you mean by known....How do you test for that" (p. 110)?

Answer: **See transcript, lines 13 to 24 of p. 110.**

33. "Did you do that to the fuel that was loaded in the first cask, rod by rod" (p. 111)?

Answer: **See transcript, line 3 of p. 111 to line 17 of p. 112.**

34. "You know, on Lake Michigan the sun reflects off the water and it reflects off the sand. It gets so hot that you can not walk on the asphalt or the concrete. What about a meltdown. There is no real cooling system on those. They are passive" (p. 113)...."Couldn't there be a meltdown when they are doing that kind of stuff" (p. 114)?

Answer: **See transcript, line 11 of p. 113 to line 7 of p. 116.**

35. "Last year during the federal register process.... we had a bunch of questions that were never answered by the NRC" (p. 116). "If you are so open to the public, give us public hearings" (p. 117).

Answer: **In the final safety assessment, the staff stated that the Palisades ISFSI uses the VSC-24 storage cask design which was reviewed and approved by NRC in a final rule issued in April 1993. The final rule is in litigation before the U.S. Court of Appeals for the Sixth Circuit in Cincinnati. The issues contested before the court include questions raised during the**

rulemaking process and the request for public hearings by some commenters. Therefore, these issues will likely be resolved by the courts.

36. "[I]s there going to be underground seepage? Can you give us some of the dimensions of the cask, how thick is the concrete and can water penetrate" (p. 118)?

Answer: The VSC-24 dry cask storage system at Palisades is a vertical cask system composed of a steel multi-assembly seal basket (MSB) and a ventilated concrete cask (VCC). The welded MSB provides confinement and criticality control for the storage and transfer of irradiated fuel. The VCC provides radiation shielding while allowing cooling of the MSB and fuel by natural convection during storage. The MSB consists of a steel cylinder shell with a thick shield plug and steel cover plates welded at each end. An internal fuel basket is designed to hold 24 spent fuel assemblies. The VCC is a reinforced-concrete cask in the shape of a hollow right circular cylinder. The VCC has four penetrations for air entrance, which are located at the bottom and four outlets located at the top. The air flow path is formed by the air inlet ducts, the gap between the MSB exterior and the VCC interior, and the air outlet ducts. The internal cavity of the VCC as well as inlets and outlets are steel lined. After loading the spent fuel assemblies into the MSB, the MSB is seal welded, dried, and backfilled with helium and then structurally welded. After the loaded MSB is inserted into the VCC, a shield ring is placed over the MSB/VCC gap and the cask weather cover is installed. Ceramic tiles are placed between the bottom of the MSB and the steel liner of the VCC to prevent potential corrosion. The response at the meeting (see transcript, lines 9 to 25 of p. 118) indicated that because the design consists of a sealed alloy steel container inside the concrete cask, there should be no leakage. The fuel is stored in a dry, inert atmosphere inside the MSB which is a 2.54-centimeter (1-inch)-thick steel canister that prevents leakage to the environment. The MSB is designed to also prevent inleakage of liquids and has been evaluated against flood conditions. The VCC has an inner steel liner 4.45-centimeter (1.75-inch)-thick and 73.7 centimeters (29 inches) of concrete used for shielding the fuel contained in the MSB. The concrete of the VCC is not relied upon as a barrier to inleakage of liquids. Liquids may be absorbed on the outer surface of the concrete but would not seep through the concrete and steel to interfere with the safe storage of the spent fuel. Liquids entering the upper outlet vents would not come into contact with the fuel but would either drain through the bottom inlet vents or evaporate because of the high temperatures on the outside of the MSB.

Detailed descriptions of the VSC-24 cask and its components can be found throughout the Safety Analysis Report. But generally, the VSC-24 cask when loaded with fuel weighs about 120 tons

(265,360 pounds), stands up to 5.4 meters (18 feet) tall, and is about 3.4 meters (11 feet) in diameter.

37. "Are you prepared to return the waste as you pledged to the Court, in the event that you lose the lawsuit" (p. 119)?

Answer: See transcript, line 16 of p. 119 to line 8 of p. 120.

The premise of the question is not entirely accurate. The issue of returning spent fuel from dry casks to the spent fuel pool was discussed in a 1993 lawsuit in the context of whether that action would physically be possible if ordered by the court. (The utility and the NRC stated that it would be possible.) In a July 1994 filing with the court, Consumers Power stated that there would continue to be room in the spent fuel pool until 1997 for all fuel in dry cask storage at the Palisades site. More recently, Consumers Power notified the Court and the public its intention to unload one of the casks containing spent fuel assemblies due to two weld flaws discovered on the metal canister. The licensee has concluded that the cask is structurally sound but chose to unload as a conservative measure.

38. "What would be--if the concrete was broken around the cask what would be the radiation field in there" (p. 122)?

Answer: The dose rates that could result from the loss of concrete shielding are estimated to be as much as about 2 sieverts/hour (200 rem/hour) at the outer surface of the cask and about 0.2 sieverts/hour (20 rem/hour) at 1 meter (3.3 feet) from the cask surface. As high as these are, dose rates of this magnitude are routinely dealt with at commercial nuclear facilities, and are well within the capabilities of specially trained personnel. These dose rates are estimated based on the very conservative assumption that all of the concrete is lost from a section of the VCC. However, as described in the answer to comment 24 above, the complete loss of concrete is unlikely. In the case of a credible event involving cask damage, the loss of concrete is unlikely to be very large. Therefore actual dose rates would be expected to be much less.

39. "I read two reports where corrosion is intensified because of radiation in a wet climate....there was another report that said that if there is corrosion between the metal basket and the metal liner of the cask, that you can never take that waste out, even if the cask now functioned or whatever" (p. 121).

Answer: These issues were previously addressed in the responses to comments 3 and 43 in the final rule that approved the VSC-24 cask (58 FR 17948) and are included here again for completeness.

Response to comment 3 (Final Rule): There are numerous ceramic tiles arranged on the base of the VCC which serve as a separator

between the flat bottom surface of the MSB and the parallel surface of the VCC liner to prevent the possibility of localized corrosion. Although these tiles could break, there is a substantial margin of safety to prevent breakage. However, if some tiles break, the tiles will still perform their function of providing a slight gap between the MSB and the VCC. Although it is not necessary, the certificate of compliance has been revised to include a statement that the operating procedures for handling the MSB over the VCC should include the consideration for reducing the likelihood of fracturing the ceramic tiles by impact load.

Response to comment 43 (Final Rule): The VSC-24 system has been evaluated for the possible effects of harsh environmental conditions and the MSB has been evaluated for the possible effects of corrosion due to humid and marine environmental conditions. As a result of the corrosion analysis of the MSB, the NRC found the design acceptable with the consideration of localized corrosion mechanisms (i.e., pitting, stress corrosion cracking, crevice corrosion, and galvanic corrosion) as well as general corrosion. Localized corrosive attack on the MSB surfaces is minimized by choice of materials and design features, such as the ceramic tiles between the VCC liner and the bottom surface of the MSB. Furthermore, the NRC allows no credit for the attributes of the paint.

The MSB structural evaluation considered the effects of degradation due to corrosion over the storage period so that the MSB could be safely handled in order to retrieve the fuel. Additional precautions, to reduce the possibility of corrosion due to metal contact between the MSB and the VCC inner liner, are provided by using the ceramic tiles as separators and by applying corrosion-protective coatings to the MSB outer surfaces and the VCC inner liner.

40. "We have asked questions about the tiles at the bottom of the cask on which you are going to rest the metal basket. We have asked how you check those tiles for their strength. We have tried for a long time to get some answer as to how those tiles are tested....The only report that we got was that they had calculated the strength of these tiles based on a 200 pound woman in two inch high heels, standing on them....I would like a response to that" (p. 121, 122).

Answer: The question about the strength of tiles at the bottom of the cask was addressed in the response to comment 3 in the Final Rule that approved the VSC-24 cask (58 FR 17948) and was included above in the response to comment 39. The tiles arranged on the base of the VCC separate the flat bottom surface of the MSB from the parallel surface of the VCC liner to prevent localized corrosion. These tiles were selected for their ability to support the weight of the loaded MSB. Their structural strength properties are based on the tile manufacturer's specifications. The calculations referred to were offered to demonstrate that

ceramic tiles intended for pedestrian traffic, were also structurally adequate to support the weight of the MSB. Licensees' and vendors' quality assurance programs are used to ensure that the proper materials are procured and that any required testing be performed.

The "American National Standard Specifications for Ceramic Tile," American National Standards Institute (ANSI) A137.1-1988, is a reference standard for buyers and specifiers of ceramic tile, and is a guide to producers in maintaining quality control of the manufacture of ceramic tiles. Section 4 of the standard includes procedures for sampling and testing and describes the basis for acceptance.

From correspondence between NRC and the public

41. "The Agency has, in effect, conceded a central issue of our lawsuit...that this is a unique environment of great importance to many people and should have a site specific evaluation" (letter from M. Sinclair to R. Bernero, May 20, 1994).

Answer: See answer to comment 1.

42. "On December 30, '94, Mr. Valdas Adamkus, Administrator for Region 3 of EPA out of Chicago, wrote to James M. Taylor of the NRC asking for more environmental information because of the importance of Lake Michigan and the Mississippi River as natural resources for the nation and the source of drinking water for many millions of people. This action supports our position that a site specific NEPA review was required" (same as comment 41).

Answer: See answer to comment 2.

43. "[A] generic cask system can not be designed [to] accommodate all these differences. Nor do we have a generic environment in this country" (same as comment 41).

Answer: See answers to comments 4 and 6.

44. "We can now also advise the Court about two important in-depth studies that were made of the Palisades site....The Corps of Engineers ...[stated] 'Erosion and bluff recession will continue, regardless of lake level controls or structural shore protection measures.'...The State of Michigan's Low Level Radioactive Waste Authority...found that none of these reactor sites was suitable for such a facility....The conclusions of both of these studies appear to directly contradict the findings of NRC's independent evaluation which was disclosed on May 19" (same as comment 41).

Answer: See answer to comment 3.

45. "[An] example of poor quality control and management is found in the fabrication control for the metal basket. Copies of the inspection

report are enclosed....How can you speak so reassuringly of the design basis for leakage from the metal basket" (letter from M. Sinclair to R. Fenech, May 25, 1994).

Answer: In Inspection Report 72-1007/92-01 (May 6, 1992), which the commenter attached to her letter, the NRC discussed several nonconformances in fabrication controls. These included fabrication travelers approvals, measurement and record of critical dimensions on fabrication travelers, maintenance and identification of radiographic films, general nondestructive examination requirements, and traceability between the materials used for fabrication and the certificate of compliance for the materials.

In letters dated June 1 and August 10, 1992, Pacific Sierra Corporation (PSNA), since renamed Sierra Nuclear Corporation (SNC) responded to NRC and proposed corrective actions. The NRC determined that those corrective actions were successfully completed, and that the items identified in the 1992 inspection report were taken care of satisfactorily. The NRC so advised PSNA in letters dated July 23 and September 1, 1992.

Since that time, however, other quality assurance (QA) issues have arisen that the NRC considers significant. These were identified in the course of the NRC's continuing regulatory oversight over the activities of the cask vendor, SNC. In late June 1994, NRC conducted a QA inspection at SNC and two of its prime fabrication contractors. (See Inspection Report No. 94207/Notice of Violation, dated August 23, 1994.) The Notice of Violation, to which SNC must respond within 30 days, identified a number of issues that need to be resolved, including fabrication controls and management deficiencies.

As a result of the NRC's findings, CPCo decided to suspend cask loading and fabrication activities until it could be assured that all pertinent QA issues had been resolved satisfactorily. NRC continues to monitor both SNC's follow-up actions and CPCo's cask validation activities to resolve QA issues and will independently evaluate the effectiveness of these corrective actions.

46. "[A]n important point you are missing about why this area is geologically unstable....Dunes are in an almost constant state of change. Moreover, they are affected by processes other than wind action, such as wave erosion lake level, groundwater changes, and climatic conditions. Thus, their history is very complex....This area does not meet NRC's siting objectives and criteria" (letter from M. Sinclair to R. Fenech, May 25, 1994).

Answer: The staff is aware that dune stability is a complex issue and that dunes along Lake Michigan in many locations may change in character as a result of various natural phenomena and natural processes. However, the staff has evaluated the various phenomena that could result in dune erosion and instability and

has concluded that the ISFSI design is acceptable at this specific location. Results of the staff review of Lake Michigan water levels, wave action, seismic design, slope stability, and groundwater considerations may be found in the Final Safety Assessment. In addition, see response to comments 3 and 23.

47. "Why was it impossible to get specifications for something as basic as the construction of the storage pad prior to the licensing of the VSC-24 cask since in 10 CFR 72.210(2)(ii), states that the general licensee shall 'Perform written evaluations, prior to use, that establish that cask storage pads and areas have been designed to adequately support the static load of the stored casks'? Why weren't these specifications placed in the public documents room" (attachment to letter from M. Sinclair to R. Fenech, May 25, 1994)?

Answer: Although there are no regulatory requirement that these specifications be placed in the NRC Public Document Room, in response to this request, they were put there.

48. "In 1988, Michigan Low Level Radioactive Waste Authority commissioned an independent study to be made of all reactor sites in the state, including that of Palisades in its efforts to co-locate a 'low-level' radioactive waste facility for the Midwest Compact. It hired a consulting firm from Ann Arbor, Michigan. Their conclusion was that none of the four nuclear power plant sites in Michigan were suitable for co-location of a low-level radioactive waste storage facility. Furthermore, they pointed out that such a facility 'would not meet the goals of the NRC's siting objectives and criteria and the overall goals of the NRC's performance objectives.' If this site is not suitable for a low-level waste facility according to NRC's own criteria, how can you now ask us to believe it is suitable for a high level nuclear waste facility? Have you applied these criteria mentioned in this report to this cask storage facility? Where is the data? Why does it not appear in your report" (same as question 47)?

Answer: See answer to comment 3.

49. "In the Final Rule, the storage pad is referred to as an 'elastic pad.' Can you explain what this means? What assurances do you have that it will support the load as you say it will given the immense amount of weight this relatively small pad will have to hold? What real world examples can you give of the successful operation of an 'elastic pad' holding this much weight for a period of at least 20 years or longer in the severe climate conditions such as can be expected in the Midwest" (same as question 47)?

Answer: Response to comment 8 in the final rule refers to the reinforced-concrete bearing pad behaving as a pad on an elastic foundation. When a pad is supported on a soil foundation, the pad is analyzed as a beam supported on a soil foundation and, therefore, there are continuous reactions that are proportional to the load at each position along the beam. This is the reason for the name

"elastic pad." The pad has been analyzed and designed to carry the cask loads.

The "elastic pad" merely distributes the weight of the casks to the foundation. If the weight of the casks is distributed over the contact area, the resulting pressure on the foundation material is well within the allowable bearing pressure of the foundation material. The foundation material at Palisades is such that it settles quickly and stabilizes within days after the initial loading. Thus, the settlement does not accumulate with time.

The licensee did the static design of the pad, and the NRC staff is satisfied with that design.

As discussed in this final safety assessment, NRC independently analyzed the pad for seismic loads and liquefaction conditions, and found it satisfactory.

50. "The storage pad is relatively small (195 ft. long x 30 ft. wide x 3 ft. deep). It will have to bear 25 casks, each of which will weigh 130 tons. There do not appear to be any calculations that indicate that this concrete slab can hold that much weight without cracking. Has this been done? Where is the data? For how long can this size concrete storage pad support this weight without failure that might cause casks to tilt or tipover? Where are these data, and how have they been verified? In Michigan, we have a good deal of experience with concrete basements cracking over time just because of severe climatic conditions and bearing the weight of a home. We see this happening to our concrete highways all of the time because of severe climate conditions and bearing of weight. Why do you believe the concrete storage pad will not crack for the same reasons and cause damage to the casks" (same as question 47)?

Answer: The storage pad rests directly on the foundation material, and the weight of the casks is transmitted to the foundation, which is capable of withstanding about 8 times the bearing stress imposed by the casks. The dead weight of the casks creates a bearing stress on the concrete pad of only 0.14 megapascal (20 pounds per square inch). This stress is very low compared to the allowable bearing stress of concrete of 16.55 megapascal (2400 pounds per square inch). In a submittal of May 12, 1994, the licensee calculated that the slab would perform acceptably under all applicable load combinations including earthquakes.

In common types of basement slabs, cracks appear in areas that carry weights from furniture, equipment or people, and not in areas which carry the weight of the building. The parts of the slab that act as structural elements of the foundation do not crack unless the foundation material is not competent or receives uneven settlement. None of these factors is present in the case of the pad. Highway beds crack because of the effect of concentrated wheel loads being dispersed over a narrow area on the road slab and because the wheel loads are from moving

automobiles. The load from constant automobile traffic causes the concrete slab to flex and eventually crack when a portion of it loses contact with the foundation. This situation does not apply to the pad.

Even if the concrete slab cracks as a result of flexing, the foundation will remain generally level, and the heavy casks would not tip over. The staff performed a safety evaluation of the VSC-24 and found that the cask will tip over when one edge of the cask bottom is lifted over 152 centimeters (5 feet) while the opposite edge is on the pad. This scenario is not credible under normal conditions and is very unlikely under earthquake conditions.

51. "Even a tilting of the cask could result in the metal basket hitting the metal lining of the concrete storage cask. This could result in corrosion that would make it impossible to remove the nuclear waste in the event of the malfunction of the cask. How do you propose to take care of this potential problem" (same as question 47)?

Answer: See the response to comment 39. Corrosion is not likely to make it impossible to remove the spent fuel. If the cask was tilted, the MSB and VCC inner liner could make contact. Therefore, a shield ring over the gap between the MSB and VCC will reduce the possibility of corrosion if the MSB and the VCC inner liner make contact. This ring would keep the MSB from tilting into the VCC inner liner. Any tilting would result in a small area of contact with the ring that, with the corrosion protective coatings, should not prevent the removal of the MSB. Surveillance requirements will ensure that the licensee prevents any major tilting of the cask that could cause the MSB to shift. The cask could be returned to the auxiliary building and the tilted conditions could be corrected before corrosion becomes a problem. The corrosion caused by contact between the MSB and VCC liner should the cask tilt is unlikely and would not involve a large area. Therefore, removing spent fuel at the end of the storage period should not be a problem.

52. "On p. 7 [of NRC's Draft Safety Assessment of Independent Spent Fuel Storage Installation Support Pad, May 18, 1994], it is stated that there are some low blow counts of soil below the ground water table under the storage pad. It states that such sands can be susceptible to liquefaction, but it is not clear how this matter was resolved. Can this be explained? It is also stated that the soils beneath the storage pad are of variable densities? Isn't this the type of condition that would cause the storage pad to crack once it was bearing a huge weight" (same as question 47)?

Answer: The problem of liquefaction of the soils with low blow counts below the groundwater table under the storage pad was independently analyzed by the NRC. Blow counts are measures of the density of the tested soil in units of blows per foot. The

lower blow counts means less dense soil. The effects of such liquefaction on the stability of the slopes adjacent to the pad and on the integrity of the pad were also investigated and the results of such investigation were reported at the May 23, 1994, public meeting; they are also reported in the NRC staff's final safety assessment. As reported therein, even though the soils with low blow counts are likely to liquefy when subjected to an earthquake of Richter magnitude 5.25, the safety of the pad will not be adversely affected. The variable densities of the soils beneath the storage pad will not cause the pad to crack. However, the pad may undergo a differential settlement of 7.62 centimeters to 10.16 centimeters (3 to 4 inches) after an earthquake which might result in some cracking of the pad. Such cracking of the pad will not adversely affect the pad's ability to carry the weight of the casks, as discussed in Attachment 1 to the staff's Final Safety Assessment.

53. "The evaluation of the effect of natural hazards, wind and wave action does not coincide with the U.S. Army Corps of Engineers [COE] latest study of that area which states 'Erosion and bluff recession will continue, regardless of lake level controls or structural shore protection.' You state you have consulted local dune erosion experts. Have you asked the Army Corps of Engineers for their data on this site? This is a comprehensive study paid for by the public and performed by experts who were competent to use very sophisticated equipment. What effort have you made to get these data and include them in your own report? Is there any reason these data was not included in this report? 72.92 (c) states that 'Appropriate methods must be adopted for evaluating the design basis external events based on the characteristics of the region and the current state of knowledge about such events.' The Corps of Engineers report would appear to be the best current state of knowledge of these events. Why wasn't it used even though the NRC was made aware of this study" (same as question 47)?

Answer: The staff reviewed detailed site studies prepared by the COE and presented in reports of the International Joint Commission. Results of this review may be found in the Final Safety Assessment. In addition, see response to comment 3.

54. "You state that aerial surveys and topographic maps of the site area show little change to the dunes area from 1965 to 1992. But we have photographs of the area in the aftermath of a 1985 storm that took out a whole dune, to a depth of 32 to 34 ft. lost during this one storm. These photos were provided by a resident of the Palisades Park area. What is the source of your data" (same as question 47)?

Answer: See response to comment 15.

55. "The NRC report discusses many natural hazards that might affect the dunes, and concludes that there will be little effect from them. This is in contradiction to the most comprehensive geological report made of Michigan's geology carried in the book. *The Geology of Michigan*, by Dorr and Eschman. These experts state, 'Dunes are in an almost constant state

of change. Moreover, they are affected by processes other than wind action, such as wave erosion, lake level, groundwater changes, and climatic conditions. Thus, their history may be very complex.' How do your conclusions coincide with those of Dorr and Eschman" (same as question 47)?

Answer: **The staff has evaluated the various phenomena that could cause dune instability and potentially affect the ISFSI. See responses to comments 3, 23, and 46.**

56. "Mr. Haughney is attempting to resolve a problem he has in the use of the VSC-24 cask,...changes must be made to this cask to permit them to be used at Point Beach and Arkansas One reactor sites....I would like an explanation of all the changes that must be made to those sites, and the accompanying documents" (letter from M. Sinclair to I. Selin, April 5, 1994).

Answer: **See answers to comments 4 and 6.**

57. "The study found that none of the four nuclear power plants in Michigan are suitable sites for co-location of low-level radioactive waste isolation facility....the nuclear power plant sites and immediately adjacent areas did not meet several key exclusionary criteria."..."the shoreline setting of each of the nuclear power plants does not offer the safety and security of alternative non-shore sites. Wind-driven flooding and seiches will undoutably play an important role in the integrity and longevity of the site and facility throughout its life" (May 24, 1988, report, "An Evaluation of the Four Licensed and Operating Nuclear Power Plant Sites in Michigan for Co-Location of a Low-Level Radioactive Waste Isolation Facility."

Answer: **See answer to comment 3.**

APPENDIX A

to

ATTACHMENT 2

12/30/93 letter from EPA to NRC

1/30/94 letter from NRC to EPA

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
REGION 5
77 WEST JACKSON BOULEVARD
CHICAGO, IL 60604-3590
DEC 30 1993

REPLY TO THE ATTENTION OF

13D18/AHH ME-19J

Mr. James M. Taylor
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, D.C. 20046-0001

Dear Mr. Taylor:

We are writing in regards to the proposed dry cask storage proposal for Consumers Power Company's Palisades Nuclear Power Plant near South Haven, Michigan and Northern States Power, Prairie Island Nuclear Power Plant. We have recently become aware of both of these proposals. From our understanding, both Consumers Power and Northern States Power have been granted approval by your agency to use a dry cask storage system to store spent nuclear fuel on a concrete pad on each facility's property. The license for storage of these spent fuel rods is for 20 years with the potential of extending the license for an additional twenty years.

The dry cask storage system was authorized by the Nuclear Waste Policy Act of 1982, Section 133. Your agency has assessed dry cask storage systems generically and has also evaluated the environmental impacts of them generically. The Consumers Power Company's dry cask storage site will be adjacent to Lake Michigan which is a valuable resource providing drinking water and recreational opportunities for many people. Similarly, the Prairie Island Nuclear Power Plant is situated on an island in the Mississippi River, another valuable natural resource. We believe the potential for significant adverse impact to either Lake Michigan or the Mississippi River is real and was not fully assessed in the generic environmental assessment prepared for the dry cast storage process. Therefore, under our authority under Section 309 of the Clean Air Act, we are requesting to review the environmental documentation that you have used to determine that this action would not have a significant impact upon the human or natural environment. The site specific conditions and the valuable resources of Lake Michigan and the Mississippi River warrant a full and complete evaluation of the impacts and the review of this analysis by other Federal and State agencies as well as the interested public.

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In addition, the Prairie Island Nuclear Power Plant shares the island with the Prairie Island Dakota Community. It does not appear that the impact to the Tribe was assessed in the generic environmental impact statement. You may need to evaluate your agency's trust responsibilities, to the Indian Tribe, regarding the siting of the dry cast storage area at the Prairie Island Nuclear Power Plant.

We will appreciate your cooperation in this matter and look forward to a timely response. If you have any questions, please feel free to contact Mr. Robert Springer, Assistant Regional Administrator for Planning and Management, at 312/353-2024.

Sincerely yours,

Valdas V. Adamkus

Valdas V. Adamkus
Regional Administrator

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 30, 1994

Mr. Valdas V. Adamkus
Regional Administrator, Region 5
U.S. Environmental Protection Agency
77 West Jackson Boulevard
Chicago, IL 60604-3590

Dear Mr. Adamkus:

I am pleased to respond to your letter of December 30, 1993, in which you state that having recently become aware of proposals for dry cask storage at the Palisades and Prairie Island nuclear power plants, your office desires copies of the environmental documentation relating to those proposals. While we are of course happy to provide the documents you seek, you should be aware that all of the environmental documentation relating to cask design used at Palisades (the "VSC-24") was sent to the U.S. Environmental Protection Agency (EPA) headquarters some six months ago, with copies to a member of the EPA Region 5 staff. (See the attached letter of August 4, 1993, to Ms. Susan Offerdal, EPA.)

Since that time, there have been a number of developments relating to dry cask storage that may be of interest to you. Most notably, last fall the Government filed its brief in Kelley v. Selin, Nos. 93-1646 et al., in the U.S. Court of Appeals for the Sixth Circuit, involving the casks at Palisades.

The central issue in this case, which is now pending before the court (no date for oral argument has so far been established) was the procedure used by the U.S. Nuclear Regulatory Commission to approve the "VSC-24" spent fuel storage casks. The NRC's position was that its procedure for approving the cask design (by generic rulemaking, without the need for additional site-specific approvals) was consistent with the clear statutory directive of Congress in the National Waste Policy Act of 1982. Congress, seeking to foster the development of "off-the-shelf" at-reactor spent fuel storage technologies that could be used safely at any nuclear power plant site, directed NRC to use generic approvals, without additional site-specific approvals (and attendant adjudicatory hearings), "to the maximum extent practicable." NRC's technical judgment, reflected in a 1990 rulemaking, was that this approach was indeed fully "practicable."

The essence of the petitioners' challenge is that there is something unique about the Palisades site that warrants a site-specific environmental analysis, notwithstanding the statutory directive. The Government's brief argues in response that the NRC's 1990 rulemaking, in which the petitioners did not participate, established the principle that no site-specific analysis is necessary or desirable so long as the NRC can find, generically, that a particular cask design can withstand the range of environmental and climatic conditions representative of NRC-licensed nuclear plant sites for which its use is approved.

9402030057 3pp.

In fact, the environmental analysis underlying the decision on the VSC-24 cask is extensive. The record reflects a series of "tiered" analyses, beginning with a Generic Environmental Impact Statement in 1979 on the handling and storage of spent fuel, and encompassing, over a period of years, a number of related and progressively more specific findings, including environmental assessments of the 1990 and 1993 rulemakings. Moreover, the Palisades site was the subject of a full environmental impact statement at the time of the initial licensing of the plant.

In sum, because the generically approved VSC-24 cask can only be used on sites already approved for nuclear power plants, the result at Palisades is that a safe and exhaustively reviewed technology has been installed on a safe and exhaustively reviewed site.

A copy of the brief is enclosed for your information.

The installation of spent fuel storage casks at Prairie Island, on the other hand, took place through an individual licensing action rather than a rulemaking. As the NRC noted in its August 4, 1992, Federal Register notice on the subject, a copy of which is enclosed, an environmental assessment found that there would be no significant impacts from construction of the casks. Radiological impacts from gaseous and liquid effluents were found to be minimal, falling within the scope of impacts evaluated for licensed reactor operations and controlled by the existing technical specifications for the Prairie Island plant.

The 1992 notice noted that the environmental assessment relied on a number of previous environmental documents, including the 1973 Final Environmental Statement for the Prairie Island plant; the 1991 Final Environmental Impact Statement on the Prairie Island Independent Spent Fuel Storage Installation, prepared by the Minnesota Environmental Quality Board; EPA's Federal Guidance Report #11, EPA 520, 1-88-020; and the 1979 Final Generic Environmental Statement on the handling and storage of spent fuel. The environmental assessment took note of the presence of a nearby Indian Tribe community and of the Bartron Archaeological Site, an area including evidence of an Indian village and burial mounds, which was discovered at the southern boundary of the plant site and was added to the National Register of Historic Places in February 1971.

A further instance of NRC's due recognition of its responsibility to the Indian Tribe in the vicinity of the Prairie Island nuclear plant will also interest you, since you mention the issue in your letter. After the NRC published notice on October 19, 1990, of its consideration of issuance of a materials license for spent fuel storage at Prairie Island, a notice of intervention was filed by the Prairie Island Mdewakanton Sioux Indian Community. In March 1991, a stipulation agreement was signed by the Tribe, the NRC staff, the utility, and two State of Minnesota agencies which had also filed intervention petitions. Under it, the petitioners withdrew their intervention petitions, and the NRC and the utility agreed to furnish complete information, including notice of relevant meetings, and to perform additional analyses requested by the petitioners.

In sum, we believe that the NRC's handling of the issue of spent fuel storage at Palisades and Prairie Island has been above reproach: in its technical and environmental soundness, in its fidelity to Congressional directives, and its responsiveness to concerns of public commenters, State bodies, and affected Indian Tribes. We think that on review of the relevant documents, you will share our view.

Sincerely,

Original signed by
R. [illegible]

Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

Enclosures:
1. Ltr to S. Offerdal
   frm F. Sturz dtd 8/4/93 w/encl.
2. Brief dated 11/3/93
3. Federal Register Notice 8/4/92

ATTACHMENT 3

Official Transcript of Proceedings

for

May 23, 1994, Public Meeting Between

NRC and the Consumers Power Company