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October 17, 1984

Mr. H. R. Denton
Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Subject: Docket No. 50-206
Seismic Withstand Capability
San Onofre Nuclear Generating Station
Unit 1

SCE has recently met with the NRC to discuss the seismic backfit program for San Onofre Unit No. 1. During those meetings, SCE was requested to provide additional information regarding the seismic capability of those systems which have not been completely upgraded to 0.67g as part of the current return to service seismic upgrade program. On October 10, SCE provided specific information which demonstrates that the plant can be safely shut down following a 0.67g earthquake and qualitative information which demonstrates that equipment similar to the systems on which backfits have not been completed have substantial seismic capability. On October 16, SCE provided the results of a detailed quantitative evaluation of portions of the systems whose upgrade is not complete. The purpose of this letter is to document the information provided in those meetings.

In 1983, SCE submitted a program to return SONGS 1 to service. This program was documented in a letter from SCE dated December 23, 1983 and approved by the NRC in a letter dated February 8, 1984. The basic goal of this program is to complete the evaluations and modifications necessary to assure that the plant can be safely shut down following a 0.67g earthquake and that such an earthquake would not cause an accident. A summary of the scope of this effort is provided in Enclosure 1. As indicated in this enclosure, all modifications are being implemented to ensure the reactor coolant system integrity and the ability to maintain a safe hot shutdown condition using only systems upgraded to 0.67g.

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Completion of the evaluation and modification to 0.67g of the remaining safety-related systems has been deferred. However, compelling evidence is available which indicates that this is not a safety concern. Essentially the only remaining open item for these systems are the seismic withstand capability of piping equipment. (Although the south turbine building extension is not completely upgraded, SCE feels confident that modifications to date ensure it can withstand an earthquake on the order of 0.5g or higher). As indicated in Enclosure 2, data available from testing and actual earthquake experience indicate that piping and equipment have the capability to withstand dynamic loads three to four times larger than that for which they are designed.

The relative insignificance of piping as a concern is also demonstrated by probabilistic risk studies as shown in Enclosure 3. Such studies indicate that piping failure is not a dominant contributor to core melt frequency. Limited risk studies done for SONGS 1 indicate the seismic risk at SONGS 1 is about the same as at other nuclear plants. Finally by upgrading the safe shutdown systems for return to service, the risk has been improved by a factor of 35 to 125 whereas upgrading the remaining systems will reduce the risk by less than 10%.

Based on the above information, it is concluded that systems and equipment at SONGS have substantial seismic withstand capability. Notwithstanding this conclusion, for systems not completely upgraded as part of the return to service effort, detailed evaluations were performed on a sample of these systems to determine their seismic withstand capability. The purpose of this effort was not to validate that the original plant met its design basis but rather to provide additional assurance that the plant as it exists today is capable of withstanding a 0.5g earthquake using contemporary methodology. The results of this evaluation are summarized in Enclosure 4 which is a copy of the handouts provided at the October 16, 1984 meeting with the NRC.

The scope of the detailed seismic evaluations included thirteen large bore pipe analyses, twenty-one small bore pipe analyses, eight equipment anchorage analyses and two tanks. This sample was selected based on a review of the April 30, 1982 Balance of Plant Mechanical Equipment and Piping report. The piping and equipment analyses with the highest stresses from this report were selected and then supplemented with additional analyses to ensure an appropriate sample of the incompletely upgraded systems. On a quantitative basis, the sample covered 33% of the incompletely upgraded large bore piping, all of the small bore piping in the BOPMEP report, and most of the non-upgraded equipment. The sample analyzed is considered a reasonable representation of incompletely upgraded systems.

The results of these evaluations demonstrate the capability of the piping and equipment to withstand a 0.5g earthquake, in accordance with the same criteria used to evaluate the return to service systems wherever applicable. (Additional criteria were developed for the refueling water storage tank and cast iron pipe. In addition, in a limited number of large bore pipe analyses, credit was taken for some or all of the 20% conservatism in the instructure response spectrum.) All eight equipment items and the two tanks satisfy the evaluation criteria for 0.5g with an average margin of about 40%. In fact, with one exception, all equipment qualified at 0.67g. For the thirteen large bore pipe analyses, all were shown to have pipe stresses within the allowables. The average margin for these analyses was about 25%. Of the pipe supports on these analyses which had been evaluated all were shown to be able to withstand a 0.5g earthquake. Finally, the twenty-one small bore pipe analyses all satisfied the return to service criteria. These results are due to a number of factors including:

1) the substantial amount of modifications completed on the incompletely upgraded sytems during the current outage,

2) the current calculational techniques used for these analyses,

3) the application of the return to service criteria and

4) the fact that these calculations were done to a 0.5g level in lieu of a 0.67g level.

Based on the combination of all of the information discussed above and the fact that the design process, NRC and ACRS reviews, and hearing board process provide a high degree of assurance that the plant was built in accordance with its original design criteria, it is concluded that those systems not completely upgraded during the current outage have the capability to withstand an earthquake of 0.5g and in all likelihood can withstand an earthquake of 0.67g.

If you have any questions in this matter, please do not hesitate to call me.

Sincerely yours,

Kenneth P Baskin

Kenneth P. Baskin
Vice President

KB:am
Enclosures

Deterministic Justification for San Onofre Unit 1 Restart

I. Introduction

San Onofre Unit 1 was originally designed and constructed with the safety systems, those designated seismic category A, able to remain functional following a 0.5g seismic event. The Seismic Reevaluation Program has since been structured to upgrade the plant to be able to safely withstand a more severe 0.67g Housner earthquake. The emphasis of the Systematic Evaluation Program of which the Seismic Reevaluation Program has become a part, is to maximize the safety improvement of the plant.

Upon return to service from the current outage the plant will have substantial upgrades that will assure that it can safely shutdown following a 0.67g Housner earthquake. This document will delineate the basis for the current outage upgrades and will demonstrate the capability of the plant to safely shutdown following a 0.67g Housner earthquake.

II. Plant Status for Restart

The emphasis of the Seismic Reevaluation Program has been on a phased upgrade of (1) the Reactor Coolant System, (2) structures, and (3) remaining systems and components. The goal of the program has been to eliminate the possibility that a design basis type event would be caused by an earthquake of a magnitude even larger than the original plant design basis such that the plant would be able to safely shutdown following such an event. The modifications in the following table have been completed to achieve the program goal.

Table 1

Structures, Systems and Components Designed or Upgraded to Withstand a 0.67g Housner Seismic Event

Structures

Containment
Sphere Enclosure Building
Diesel Generator Building
Turbine Deck, North, West and East Platforms
Control Building
Seawall
Masonry Walls in Ventilation Building, Reactor Auxiliary Building, Fuel Building and Turbine Building
Fuel Storage Building
Service Water Reservoir

Systems and Components

Reactor Coolant Pressure Boundary
Standby Power System
Auxiliary Feedwater System (Including New Tank)
Portions of Charging and Letdown Systems Necessary for Safe Shutdown
Intertie Between Spent Fuel Pool and Charging System
Portions of the Main Feedwater System Necessary for System Integrity
Portions of the Main Steam System Necessary for System Integrity and to Provide Steam Dump Capability.

In addition to the modifications listed, major improvements to other systems have been initiated, but will not be completed during the current outage.

III. Plant Response to Severe Seismic Event

The modifications that have been completed are intended to satisfy the requirement to achieve safe shutdown following a severe seismic event.

With the complete upgrade of the Reactor Coolant Pressure Boundary and portions of the Main Feedwater and Main Steam Systems necessary to maintain integrity, the possibility of a severe seismic event rupturing one of these systems has been removed. Since the Emergency Core Cooling System is required only to flood the core under conditions of loss of coolant or shrinkage due to rapid cooling, upgrades to the system are not necessary.

Following a seismic event, reactor shutdown will be initiated manually (if not automatically caused by the event) and the reactor will be maintained safely in a Mode 3 or Mode 4 condition. The shutdown will be achieved using the charging pumps (the test pump and one normal charging pump with cooling supplied by its own fan) to provide makeup due to shrinkage and minor leakage from the Reactor Coolant System. The Component Cooling Water System, which normally cools the charging pumps will not be required due to the capability of the charging pump cooling fan. The test pump does not require external cooling.

Makeup water for the charging system will be supplied by a new intertie to the spent fuel pool. This path will be made available following the event by action of an operator who will be dispatched to manually align the system. All other connections that might otherwise supply makeup to the charging pumps will be isolated. The spent fuel pool will be used as the makeup source and 80,000 gallons of borated water will be available which is sufficient makeup for at least 1 week assuming worst case conditions. Should additional borated water be required it can be manually added directly to the spent fuel pool.

The reactor will be cooled through the use of the completely upgraded Auxiliary Feedwater System through its connection to a new Auxiliary Feedwater Storage Tank. The system is completely redundant. The steam generators will act as the heat exchangers for cooling the RCS through natural circulation with the steam being dumped through redundant steam dump valves. The requirement to remove heat from the Reactor Coolant

System will be met by the Auxiliary Feedwater System with water supplied by the new Auxiliary Feedwater Storage Tank. With the minimum required 150,000 gallons in the tank there is at least 32 hours of cooling capability. With the maximum amount of water in the tank, over 240,000 gallons, there is at least 50 hours of cooling capability. Once this water source diminishes, additional supplies can be made available from the 3 million gallon service water reservoir. With this water the plant can be safely cooled for over an additional 21 days. In this period of time an indefinte amount of cooling water can be made available from external sources.

IV. Conclusion

With completion of the current upgrade program, San Onofre Unit 1 will have the capability to withstand a severe seismic event without the loss of reactor coolant system integrity and can be maintained indefinitely in a safe shutdown condition using only those systems upgraded for a 0.67g Hounser event.

V. Reference

Letter, Walt Paulson, NRC, to R. Dietch, SCE, regarding SEP Topic VII-3, Systems Required for Safe Shutdown, dated November 12, 1982.

GEH:2630F

Enclosure #2

SEISMIC CAPACITY OF PIPING SYSTEMS

BASED ON EXPERIENCE

I. INTRODUCTION

Structures, systems and components have substantial reserve capacity with respect to their earthquake withstand capability. In the case of piping this is demonstrated by risk assessments, test results, and actual earthquake experience. In general, piping can be shown to be relatively rugged with respect to its seismic capability.

The purpose of this report is to summarize experience related to the seismic capacity of piping systems. This is done by examining (1) the contribution of piping systems to the overall core melt frequency in probabilistic risk assessments, (2) the results of dynamic testing of piping and (3) experience of piping systems in actual earthquakes.

II. PROBABILISTIC RISK ASSESSMENTS

Published probabilistic risk assessments (PRA) done at other nuclear plants were reviewed to identify the dominant contributors to seismic core melt frequency. This review was documented in Reference 1. The first step in this evaluation was a review of three published PRA's (Zion, Indian Point Units 2 and 3, and Limerick) to attempt to characterize the dominant contributors. The list of dominant contributors to core melt was supplemented based on discussions with PRA experts as well as to include seismically induced failures which are major contributors to offsite consequences.

Based on this review of existing seismic PRA studies, the dominant contributors which were identified were broadly categorized as (1) onsite power, (2) essential water supplies, (3) structures and (4) reactor coolant system. In addition, it was concluded that failure of ductile steel piping is not a dominant contributor to seismic risk. The only types of piping systems that were identified as potential contributors were non-ductile pipe, threaded joints and piping routed between structures. These special circumstances are discussed below with respect to the design of SONGS 1.

With the exception of a small amount of cast iron pipe, all SONGS 1 safety-related piping systems are welded ductile steel in nature. Historical experience has shown that such systems have a very high seismic withstand capability. This experience is reflected in seismic PRA fragility data, all of which indicate that the median ground acceleration capacities of welded ductile steel piping systems are sufficiently high that - even considering uncertainties - the probability of failure at ground accelerations of 0.67g and lower is small and not a major contributor to seismic core melt frequency even for piping systems which were designed for a Safe Shutdown Earthquake much lower than 0.67g.

SONGS 1 contains a limited amount of buried cast iron pipe associated with the salt water cooling (SWC) system. Cast iron pipe is known to be more susceptible to failure under seismic load. To ensure the safe shutdown capability of SONGS 1, alternate means of cooling the systems required for hot safe shutdown have been provided. Thus, this item has been eliminated as a potential contributor to seismic core melt frequency.

SONGS 1 utilizes no threaded joints in process piping for those systems required to get to a safe shutdown. Therefore, this item is eliminated as a potential contributor to seismic core melt frequency.

SONGS 1 contains some piping spanning buildings on separate foundations. The evaluation of piping spans between structures that are required to attain safe shutdown has been specifically addressed in the return to service evaluations and upgrades. Thus this item has been eliminated as an important contributor to seismically induced core melt frequency.

Based on this review of seismic PRA's it is concluded that piping is not a major contributor to seismic risk at SONGS 1. Even the special circumstances of potential problem areas do not exist or have been eliminated at SONGS 1 for the safe shutdown systems.

III. DYNAMIC TESTING OF PIPING

Researchers at UCLA undertook a program to dynamically test nuclear related piping and equipment in the late 1960's at San Onofre. These tests were performed with small shakers on the operating deck of the reactor building and were very low amplitude. Though these tests demonstrated the capability of piping under dynamic loads, the nature of the tests and the state of the art at the time are a limitation on the usefulness of the results.

Recently, a number of test programs intended to investigate the performance of typical nuclear piping systems have been undertaken. These include testing performed by EPRI, Hanford Engineering Development Laboratory, the Earthquake Engineering Research Center at UC Berkeley, and ANCO Engineers. There are similarities in all of the results reported. These are summarized as follows:

(1) piping systems have withstand capabilities well beyond the limits of the design

(2) damping in piping systems tends to be higher than used in current design practice

(3) flexible piping systems have substantial withstand capability that goes will beyond what is predicted by linear analysis and even exceeds that predicted by nonlinear analysis.

A summary of some recent piping tests is described in the following paragraphs.

A. Small Bore Piping

ANCO Engineers has performed a number of small bore piping tests for Kraftwerk Union (KWU) in the Federal Republic of Germany, EPRI, and the Bechtel Power Corporation. Of particular interest are the tests performed by ANCO to qualify small bore piping for KWU, (References 2 and 3). These tests were used to generically qualify flexible small bore piping without the need to perform sophisticated computer analyses. The qualification of existing small bore piping in KWU nuclear power plants to withstand low frequency loading (SSE) and high frequency loading (aircraft impact) was successfully demonstrated by ANCO Engineers for KWU.

A series of full-scale tests, using small bore piping systems typical of those installed in KWU nuclear power plants, was conducted on a shake table. Nine small piping configurations were selected for testing. These configurations are representative of the large majority of piping systems in nuclear power plants and are the most critical sections for each type. Trapeze-supported, hung, and horizontally restrained systems were included in the test program. A variety of boundary conditions, such as one-dimensional restraints, hangers, stops, pressure ranges, and added masses, were also investigated.

These tests clearly showed that small bore piping is capable of surviving low and high frequency loads where large displacements, accelerations and even plastic strains occurred. Some accelerations were in excess of 50g, displacements were in excess of 50cm and plastic strains were in excess of about 0.6%. The tested piping configurations survived without collapse, leaks or loss of pressure. Stress analyses performed on these lines indicate large overstresses, as much as 300 to 400% over the code allowable stress levels.

B. Large Bore Piping

ANCO Engineers has also performed a number of tests of large bore piping systems. These tests were performed to obtain benchmark data for piping computer codes and to demonstrate piping design margins for dynamic loads.

In one test program (Reference 4), two 20 foot runs of 4-1/2 inch carbon steel piping were tested. High level dynamic loads above the elastic range of the piping material and above the Code Class 2 Level D stress limit were induced in the piping system with peak input accelerations ranging above 10g. The piping systems successfully withstood repeated dynamic loading at input levels from three to four times greater than those necessary to exceed the ASME Class 2 Level D stress limit for primary loads.

In another test program (Reference 5), two configurations were tested. One piping system was a single run of carbon steel about 70 feet long comprised of six inch and eight inch schedule 40 piping. The second system consisted of two 3 inch schedule 40 branch lines tied into a mainline similar to that in the first test. The tests included time histroy input with a peak acceleration of 8.4g. The test inputs were a factor of 4 greater than that necessary to match the Level D stress limits. The piping system withstood these severe dynamic tests with no gross distortion or loss of pressure retaining capacity.

IV. EARTHQUAKE EXPERIENCE

A formidable quantity of contemporary evidence is available to demonstrate that piping systems and equipment designed with controlled flexibility have the capacity to withstand forces far in excess of those for which they were designed. Reference 6 includes data collected from more than twenty power plants and industrial facilities which were subject to severe seismic motion. A typical example is the ESSO refinery in Managua, Nicaragua which was designed to meet provisions of the Uniform Building Code for a 0.2g seismic acceleration. During the 1972 Managua, Nicaragua earthquake, the peak acceleraton measured at the refinery was 0.39g E-W and 0.34g N-S. Despite the fact the ground acceleration exceeded by nearly 100% the acceleration for which the systems were designed, virtually no damage was sustained by the piping systems and equipment. The plant was shut down for inspection but was operating at full capacity within 24 hours. Even more impressive evidence can be found at the ENALUF Power Plant which was subject to an estimated 0.6g ground motion during the same earthquake. This plant sustained no damage to its piping and equipment, despite a probably non-existent seismic design.

In addition to the survey presented in Reference 6, a more comprehensive study was made of the response of the El Centro steam plant to the 1979 Imperial Valley earthquake by Murray, et. al. The results of this study were published in NUREG CR-1665 (Reference 7), and seem to demonstrate that a conventional plant probably designed for a 0.1g to 0.2g seismic acceleration, successfully withstood a much higher seismic acceleration, probably on the order of 0.5g. Significant conclusions of this study that relate to the piping are excerpted as follows:

(1) "No high-temperture or high-pressure piping failed during the earthquake."

(2) "General observations indicate that the piping systems are hung in a more flexible manner than that which would be required by current NRC criteria."

(3) "In most cases, the piping is supported in a similar manner to older operating nuclear power plants, and it may be inferred that the seismic response would be similar. These observations are, on the surface, encouraging since in all cases the circumstances leading to failure are dissimilar to nuclear applications in that damage occurred at weld repaired areas of past corrosive attack or at nonwelded pipe joints."

The evidence of earthquake experience clearly indicates that piping systems that are well laid out and anchored according to industry practice have an inherent resilience that permits them to withstand substantially greater seismic inputs than would be indicated by current design practice.

V. CONCLUSION

Based on the information presented above it is concluded that piping failure is not a dominant contributor to seismic risk. Further, based on both testing and actual experience, piping can withstand dynamic loading several times its design level and several times the minimum acceptable ASME Code level.

VI. REFERENCES

1. Letter from M. O. Medford (SCE) to D. M. Crutchfield (NRC), "Return to Service, Seismic Reevaluation Program, San Onofre Nuclear Generating Station, Unit 1," December 23, 1983.

2. Ibanez, P., Keowen, R. S., Rentz, P. E., "Experimental Study of Dynamic Behavior of Piping Systems Under Maximum Load Conditions--Testing," 1982 ASME Orlando Conference.

3. Sand, Lockau, Schoor, Haas, "Experimental Study of Dynamic Behavior of Piping Systems Under Maximum Load Conditions--Analysis," 1982 ASME Orlando Conference.

4. Howard, G.E., Johnson, B.A., Walton, W.B., "Piping Extreme Dynamic Response Studies," Nuclear Engineering and Design 77, pp 405-417, 1984.

5. ANCO Engineers, "Laboratory Studies: Dynamic Responses of Prototypical Piping Systems," NUREG/CR-3893, August 1984.

6. Cloud, R. L., "Seismic Performance of Piping in Past Earthquakes," Proceedings of the Specialty Conference on Civil Engineering and Nuclear Power, September 15-17, 1980.

7. Murray, Nelson, et. al., "Equipment Response at the El Centro Steam Plant during the October 15, 1979 Imperial Valley Earthquake," NUREG CR-1665, October 1980.

SUMMARY OF PROBABILISTIC ANALYSES SUPPORTING SAN ONOFRE UNIT 1 RETURN TO SERVICE

Introduction

The risk associated with seismic events at San Onofre Unit 1 is dependent upon several contributors as follows:

Seismic Hazard - the probability of an earthquake, the associated magnitude, and the type of structure being affected.

Plant Design Criteria - The magnitude of earthquake selected as a basis for plant design.

Plant Seismic Capability - The actual capability of a plant to withstand a seismic event.

Relative System Importance - Each system contributes more or less risk according to its need for maintaining the reactor in a safe condition.

Each of these areas have been reviewed and estimates have been placed on the magnitude of the risk associated with seismic events for San Onofre Unit 1.

Seismic Hazard

The seismic hazard for San Onofre Unit 1 has been estimated based on measurements and analysis applied to local faults (References 1, 2, and Attachment 1 to Reference 3). The studies include many conservatisms, including assumptions on the actual characteristics of nearby faults and on the methods of measurement of ground acceleration (References 4 through 15). The result of the studies is that the Instrumental Peak Ground Acceleration (IPGA) for the San Onofre Unit 1 site is as follows:

Ground Acceleration	Corresponding Return Period
0.5g	1.5×10^4 years (6×10^{-5}/year)
0.67g	$4 \times 10^5 - 1.5 \times 10^6$ years ($2.5 \times 10^{-6} - 6 \times 10^{-7}$/year)

These low numbers in themselves, especially considering the conservatisms used in their calculation, demonstrate the low probability of an event that could challenge the plant seismic capability.

Plant Design Criteria

The original San Onofre Unit 1 design was intended to assure the function of safety systems for a 0.5g seismic event. The current upgrade program will assure the function of equipment for a safe shutdown of the plant following a 0.67g Housner event.

Plant Seismic Capability

Notwithstanding the criteria used in the design of the plant, experience at industrial facilities similar to San Onofre Unit 1 has clearly demonstrated that plants designed for a particular level of earthquake can withstand events of much greater magnitude (See References 16 through 23). As described in SCE's December 23, 1983 submittal, and described in detail in Reference 21, recent experience at a generating facility demonstrates a safety margin of at least two. (The El Centro Steam Plant in the Imperial Valley of Southern California experienced an earthquake in 1979 estimated to have caused ground acceleration of approximately 0.5g. Though the plant was probably designed for a ground acceleration of between 0.1g and 0.2g, it experienced no damage to high pressure or high temperature piping.)

Relative System and Component Importance

The upgrade program for San Onofre Unit 1 recognizes the relative importance of those systems required to maintain their function following an earthquake. The program has completely upgraded the Reactor Coolant System Pressure Boundary and those systems necessary to assure a safe shutdown. This is supported by a review of the dominant contributors to seismic risk identified through Probabilistic Risk Assessments performed at other nuclear facilities as described in Appendix B to Reference 3. At San Onofre Unit 1 all those dominant contributors will have upgrades completed prior to return-to-service from the current outage. The primary remaining items are on systems that do not significantly contribute to the risk and are associated with those portions (primarily piping) of these systems that have been identified as being resistant to seismic loads far in excess of design.

Results of Risk Studies

In order to combine the effects of each of the above areas, a limited PRA study has been done for San Onofre Unit 1 and was included as Appendix C to Reference 3. The study constructed fault trees specific to San Onofre Unit 1 and applied data from other PRA's and the Seismic Safety Margin Review Program (SSMRP). The result is a core melt frequency due to seismic events on the order of 2×10^{-5} per year. This number reflects that the plant has been significantly upgraded and that the upgrades have been concentrated on those systems most important to seismic risk. This number is well within the NRC Safety Goal for large-scale core melt of less than 10^{-4}/year.

Since any estimate of the seismic risk is subject to uncertainty, and because the emphasis of the Systematic Evaluation Program is to maximize relative plant improvement, a further study (Reference 24) has been performed. This study has estimated that the risk associated with seismic events at San Onofre Unit 1 has been reduced by a factor of between 35 and 125 times when compared with the plant as it existed at the beginning of the current outage. It was also noted that because of the relative unimportance of the systems which have not been completely upgraded and because of the low contribution to risk of those portions of those systems which have not been upgraded (piping), the further improvement due to their upgrade would be less than 10%.

Conclusion

The seismic risk at San Onofre Unit 1 has been evaluated and quantified to be approximately 2 x 10^{-5}/year which is comparable to that at other nuclear facilities. The relative improvement in plant safety due to the modifications performed during the current outage is on the order of 35 to 125 times. Further improvements would not sigificantly improve these numbers.

The return-to-service of San Onofre Unit 1 will not represent an undue risk to the health and safety of the public and represents the successful completion of a major upgrade program which has met its objective of improving plant safety.

References

1. Wight, L. H., "Evaluation of Peak Horizontal Ground Acceleration Associated with the Offshore Zone of Deformation at San Onofre Nuclear Generating Station," TERA Corp., Exhibit No. LHW-1, Direct Testimony, ASLB Hearings, 1981, report dated August 1980.

2. Campbell, K. W., "Near-source Attenuation of Peak Horizontal Acceleration," BSSA, v711, pp. 2039-2070, 1981

3. Letter, M. O. Medford, SCE, to D. M. Crutchfield, NRC, dated December 23, 1983

4. Boore, D. M., "The Effect of Simple Topography on Seismic Waves: Implications for the Accelerations Recorded at Pacoima Dam, San Fernando Valley, California," BSSA, v63, pp. 1603-1609, 1973.

5. McNeill, R. L., "Some Possible Errors in Recorded Free-Field Ground Motions," Draft internal memorandum for Sandia National Laboratories, Albuquerque, 1981

6. Bycroft, G. N., "Forced Vibrations of a Rigid Circular Plate on a Semi-Infinite Elastic Half Space and on an Elastic Stratum," Phil. Trans, Roy. Soc. London, 1956

7. Bycroft, G. N., "The Magnificantion Caused by Partial Resonance of the Foundations of a Ground Vibration Detector," Trans. AGU, v38, 1957

8. Bycroft, G. N., "The Effect of Soil Structure Interaction on Seismometer Readings," BSSA, v68, 1978

9. Pasecnik, P., "Theoretical and Experimental Study of the Resonance of a Seismometer System (in Russian)," Izv. Akad. Nauk SSSR. ser. geofiz., 1952

10. Wolf, A., "The Equation of Motion of a Geophone on the Surface of an Elastic Earth," Geophysics, vIX, 1944

11. Washburn, H., and H. Wiley, "The Effect of the Placement of a Seismometer on its Response Characteristics," Geophysics, vVI, 1941

12. Safar, M. H., "On the Minimization of the Distortion Caused by Geophone-Ground Coupling," Geophysical Prospecting, v26, 1978

13. Fail, J. P., G. Frau, and M. LaVergne, "Couping of Seismographs to Soil (in French)," Geophysical Prospecting, vX, 1962

14. McNeill, R. L., "Machine Foundations: The State of the Art," Invited Paper, Proc., Spec. Session 2, Seventh Intl. Conf. on Soil Mech. and Fndtn. Engrg., Mexico City, 1969

15. McJunkin, R. D., "Strong-Motion Free-Field Site Design Characteristics," California Geology, Aug. 1979

16. Real, C. R., McJunkin, R. D. and E. Leivas, "Effects of Imperial Valley Earthquake," California Geoglogy, Dec. 1979

17. South Carolina Electric and Gas Company: "Seismic Confirmatory Program for Equipment and Components for the Virgil C. Summer Nuclear Station (with Addenda)," Docket No. 50/395, OL NPF-12, License Condition 2.C(25), March 9, 1983

18. Conference Proceedings, EERI, "Managua, Nicaragua Earthquake of December 23, 1972," 1973

19. Reconnaissance Report, EERI, "Managua, Nicaragua Earthquake of December 23, 1972," 1973

20. Steinbrugge, K. V., "Earthquakes, Volcanoes, and Tsunamis," Skandia America Group, NY, 1982

21. Murray, R. C., Nelson, T. A. Campbell, R. D. Young, J. A., Levin, H. A., Martore, J. A., and L. Reiter, "Equipment Response at the El Centro Steam Plant during the October 15, 1979 Imperial Valley Earthquake," NUREG/CR-1665, 1980

22. Yanev, P. I., Hou, S., Swan, S. W., Chan, T. K., and P. Ouborg, "Survey of the Effects of the May 2, 1983 Coalinga Earthquake on Industrial Facilities," Draft report expected to be updated and issued in the near future, 1983

23. EQE, Inc., "Analysis of failed transformer anchorage in Coalinga," letter report to R. L. McNeill, 1983

24. Guey, C. N., Hughes, E. A., and Lambert, H. E., "Evaluation of the Core Damage Risk Reduction Associated with Seismic Modifications at the San Onofre Nuclear Generating Station, Unit 1," submitted via letter from M. O. Medford, SCE, to D. M. Crutchfield, NRC, dated July 30, 1984

GEH:2623F

Enclosure #4

RETURN TO SERVICE

EVALUATION OF PIPING AND EQUIPMENT

SAN ONOFRE UNIT 1

OCTOBER 16, 1984

AGENDA

I.	INTRODUCTION	M. MEDFORD
II.	RESPONSE SPECTRA	J. EIDINGER
III.	EQUIPMENT	J. EIDINGER
IV.	LARGE BORE PIPING	W. GALLO/R. GAVANKAR
V.	SMALL BORE PIPING	R. GAVANKAR
VI.	CONCLUSION	M. MEDFORD

SCALE FACTORS TO DEVELOP REDUCED SPECTRA

EARTHQUAKE LEVEL (ZPA)	RATIO TO 0.67G	SCALE FACTORS USED	
		HORIZONTAL EQ	VERTICAL EQ
0.5G	0.75	0.825 (1)	0.800 (2)
0.4G	0.60	0.720 (1)	0.685 (2)

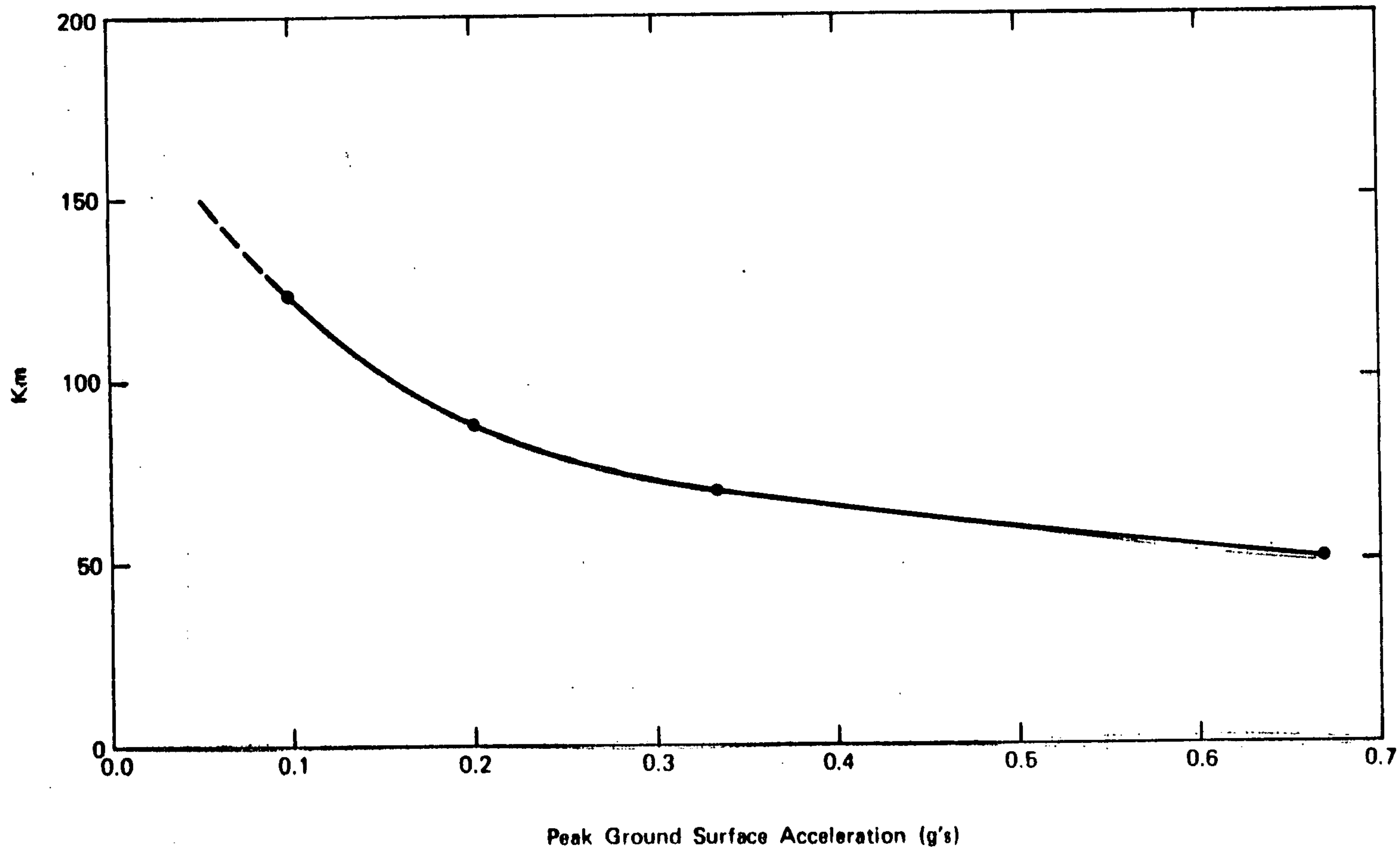
(1) INTERPOLATED FROM 0.65 HORIZONTAL FACTOR FOR OBE/SSE RATIOS AT OTHER NUCLEAR PLANTS

(2) INTERPOLATED FROM 0.60 VERTICAL FACTOR FOR OBE/SSE RATIOS AT OTHER NUCLEAR PLANTS

BROADENING FACTOR FOR RESPONSE SPECTRA

EARTHQUAKE LEVEL	STRAIN-ITERATED SOIL "K" FACTOR	SOIL STRAIN (%)	SOIL MODE BROADENING FACTOR
0.67	40 TO 55	0.25 TO 0.40	1.0
0.50	45 TO 62 (1)	--	1.06
0.40	48 TO 67 (1)	--	1.10
0.33	50 TO 70	0.08 TO 0.12	1.13

(1) INTERPOLATED FROM 0.67G AND 0.33G RESULTS



VARIATION OF Km WITH PEAK GROUND SURFACE ACCELERATION

SUMMARY OF SPECTRA DEVELOPMENT PROCEDURE

FOR 0.5G INPUT MOTION

STEP 1 - MULTIPLY "1983 - 0.67G" FLOOR SPECTRA ENVELOPES BY SCALE FACTOR (0.825 HORIZONTAL, 0.80 VERTICAL)

STEP 2 - FOR FIRST MODE PEAK, BROADEN PEAK BY 6 PERCENT FOR THE HIGHER FREQUENCY DIRECTION ONLY

CONSERVATISMS IN 0.5G SPECTRA

I. SCALE FACTORS FOR 0.5G EVENT

- 0 FACTOR OF 0.825 EXCEEDS 0.50G/0.67G = 0.75. FACTOR TO ALLOW FOR POSSIBLE LOWER DAMPING IN STRUCTURES AND SOILS IN SMALLER EARTHQUAKES.
- 0 THE 0.5G EVENT IS A DBE EVENT, AND CAUSES STRESSES NEAR YIELD IN STRUCTURES AND EQUIPMENT. THEREFORE, DAMPING FOR STRUCTURES AND EQUIPMENT IS AT DBE LEVELS.
- 0 THE 0.5G EVENT CAUSES LOWER SOIL STRAINS. THIS REDUCES SOIL MATERIAL DAMPING. FOR THE 0.67G EVENT, COMBINED SOIL MATERIAL AND RADIATION DAMPING IS NEAR 35%. FOR THE 0.5G EVENT, THE COMBINED SOIL MATERIAL AND RADIATION DAMPING IS NEAR 30%.
- 0 THE 1980 SEP SPECTRA CALCULATION USED A CUT-OFF DAMPING OF 20%.
- 0 THEREFORE, DAMPING FOR SOIL IS UNCHANGED BETWEEN 0.5G AND 0.67G EVENTS.

II. ARTIFICIAL VERSUS SMOOTH HOUSNER FREE FIELD SPECTRUM

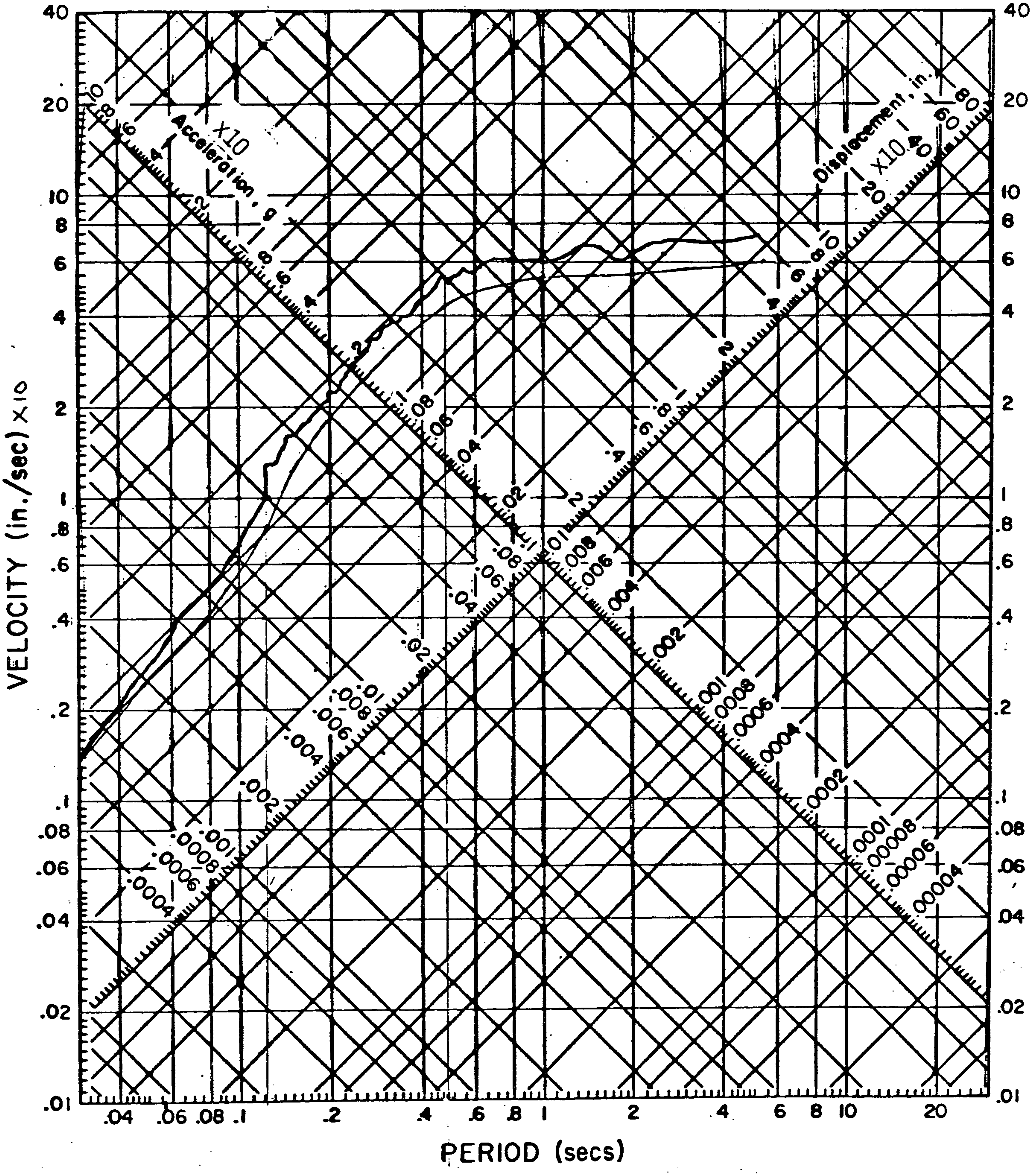
PERIOD RANGE	ARTIFICIAL/SMOOTH SPECTRA
0.25 TO 1.00 SECONDS (1)	115% TO 128%
0.07 TO 0.25 SECONDS	105% TO 108%
0.033 TO 0.07 SECONDS	108% TO 112%

(1) ALL FLOOR ENVELOPE SPECTRAL PEAKS ARE IN THIS RANGE.

III. SUMMARY

		CONSERVATISM IN ENVELOPE
0	SCALE FACTORS	1.10
0	ARTIFICIAL VERSUS SMOOTH (AVERAGE, 1 TO 33 HZ)	1.10 TO 1.16
	TOTAL	1.21 TO 1.27
	SIMPLIFIED REDUCTION FACTOR	0.80

FREE FIELD RESPONSE SPECTRA: COMPARISON OF ARTIFICIAL TIME HISTORY AND 0.67G HOUSNER SPECTRA



EQUIPMENT

- EVALUATION CRITERIA AND METHODOLOGY
- SAMPLE SELECTION CRITERIA
- EQUIPMENT ITEMS
- EQUIPMENT EVALUATION RESULTS

EQUIPMENT EVALUATION CRITERIA AND METHODOLOGY

1. EQUIPMENT (PUMPS, HEAT EXCHANGERS, SURGE TANK)

   - O EVALUATION CRITERIA IS THE RTS CRITERIA
   - O METHODOLOGY
     - STATIC ANALYSIS OF SUPPORTS
     - FREQUENCY CALCULATION TO DETERMINE ACTUAL SEISMIC ACCELERATION

2. REFUELING WATER STORAGE TANK CRITERIA AND METHODOLOGY

   - O EVALUATION CRITERIA (TWO APPROACHES)
     - API 650
     - ASME SECTION III
   - O COMPRESSIVE STRESS ALLOWABLES FOR SHELL
     - USE STANDARD COMPRESSIVE ALLOWABLES
     - INCREASE FOR INTERNAL PRESSURE EFFECTS
     - INCREASE FOR BENDING EFFECTS
   - O METHODOLOGY
     - SEISMIC LOADINGS CALCULATED BY MODIFIED HOUSNER METHOD
     - SETTLEMENT DUE TO IN-SITU BACKFILL UNDER PORTION OF TANK EVALUATED
     - ANCHORAGE AND BASEMAT EVALUATED FOR STRUCTURAL ADEQUACY USING STANDARD PROCEDURES

SAMPLE SELECTION CRITERIA

0 ALL EQUIPMENT IN BOPMEP

0 SELECTION FOR REVIEW CONSIDERED ALL MAJOR EQUIPMENT IN BOPMEP WHICH WAS OVERSTRESSED

EQUIPMENT ITEMS

0 RHR PUMPS (G-14A,B)

0 RECIRCULATION HEAT EXCHANGER (E-11)

0 COMPONENT COOLING WATER HEAT EXCHANGERS (E-20A,B)

0 RHR HEAT EXCHANGERS (E-21A,B)

0 COMPONENT COOLING WATER PUMPS (G-15A,B,C)

0 SALT WATER COOLING PUMPS (G-13A,B)

0 REFUELING WATER PUMP (G-27)

0 SAFETY INJECTION PUMPS (G-50A,B)

0 COMPONENT COOLING WATER SURGE TANK (C-17)

0 REFUELING WATER STORAGE TANK (D-1)

EQUIPMENT EVALUATION RESULTS

EQUIPMENT	ACCEPTABLE	NOT ACCEPTABLE
RHR PUMPS	X	
RECIRC. HEAT EXCHANGERS	X	
CCW HEAT EXCHANGERS	X	
RHR HEAT EXCHANGERS	X	
CCW PUMPS	X	
SALT WATER COOLING PUMPS	X	
REFUELING WATER PUMPS	X	
CCW SURGE TANK	X	
REFUELING WATER STORAGE TANK	X	
SAFETY INJECTION PUMPS	X	

O ASSESSMENT SHOWS THAT ALL EQUIPMENT SATISFY EVALUATION CRITERIA FOR 0.5G SEISMIC EVENT

LARGE BORE PIPING & SUPPORTS

CRITERIA & METHODOLOGY

- O RETURN TO SERVICE CRITERIA FOR HOT SAFE SHUTDOWN LARGE BORE PIPING AND SUPPORTS
- O 0.5 G GROUND ACCELERATION SPECTRA
- O PIPING ANALYSIS PERFORMED USING FULLY VERIFIED COMPUTER CODE SUPERPIPE
- O SUPPORTS QUALIFIED BY HAND CALCULATION OR STRUDL ANALYSIS

LARGE BORE PIPING & SUPPORTS

SAMPLE SELECTION CRITERIA AND CHARACTERISTICS

- SELECTED TEN ACCIDENT MITIGATION AND OUT-OF-SCOPE SAFE SHUTDOWN PIPING
- MOST HIGHLY STRESSED STRESS PROBLEMS FROM THE BOPMEP REPORT OTHER THAN HSS
- MAJOR SYSTEMS COVERED:

  FEED WATER

  AUXILIARY COOLANT

  MISCELLANEOUS WATER

  SAFETY INJECTION

  CONTAINMENT AIR CONDITIONING
- PIPE SIZES FROM 2" TO 16"
- TOTAL 192 SUPPORTS

LARGE BORE PIPING & SUPPORTS

SAMPLE

0	FW-05	6", 4", 3" AND 2" FROM CONDENSER E-2A TO PUMP G-3B
0	FW-06	3" AND 2" FROM CONDENSER E-2A TO G-3A
0	AC-05	14", 8", 6", 3", 2", 1" FROM UPPER AND LOWER BEARING OIL COOLERS AND OTHER HXS TO CCW HX
0	AC-03	4" AND 3" FRM CC SURGE TABK C-17 TO 14" AUX COOLING LINE AND MAKE-UP WATER
0	CA-55	6" FROM PENETRATION B-17B TO AC DUCT
0	MW-04	8" MISCELLANEOUS WATER FROM PEN. B-11 TO RECIRC. HX E-11
0	MW-05	8", 6" 14" FROM PEN. B-11 TO REFUELING CANAL SUMP AND FILTER AND SI RECIRC PUMPS G-45A AND G-45B
0	MW-51	8", 3" VAPOR CONTAINMENT COOLING AND VENTILATING UNITS THROUGH PEN. A-9A
0	SI-04	16", 14", AND 8" FROM SI PUMP G-50A TO THE RWST AND FW PUMP G-3A

LARGE BORE PIPING & SUPPORTS

RESULTS

ANALYSIS	PIPE STRESS	SUPPORTS
AC-03	QUAL. TO .5G	QUAL. TO .5G
CA-55	QUAL. TO .5G	QUAL. TO .5G
MW-04	QUAL. TO .5G	QUAL. TO .5G
MW-51	QUAL. TO .5G	QUAL. TO .5G
FW-05	QUAL. TO .5G	QUAL. TO .5G (1)
FW-06	QUAL. TO .5G	INCOMPLETE (2)
SI-04	QUAL. TO .5G	INCOMPLETE (2)
MW-05	QUAL. TO .5G	INCOMPLETE (2)
AC-05	QUAL. TO .5G	INCOMPLETE (2)
AC-06	QUAL. TO .5G	INCOMPLETE (2)

(1) ONE SUPPORT QUALIFIED TO 0.4G BY INITIAL EVALUATION

(2) THE INITIAL SUPPORT EVALUATION IS STILL IN PROGRESS FOR THESE PROBLEMS; SOME SUPPORTS EXCEED THE ACCEPTANCE CRITERIA IN THEIR INITIAL EVALUATION: REFINED ITERATIONS ARE IN PROGRESS FOR THESE.

BPC EVALUATED LARGE BORE PIPING

- O EVALUATED THREE (3) OUT-OF-SCOPE SAFE SHUTDOWN PIPE STRESS CALCULATIONS
- O SELECTION CRITERIA FOR 2 STRESS CALCULATIONS
  - O THOSE FROM THE 1982 BOPMEP REPORT THAT HAD STRESSES WHICH EXCEEDED THE BOPMEP REEVALUATION CRITERIA ($2.4S_H$)

BPC EVALUATED LARGE BORE PIPING

- O **PROBLEMS EVALUATED** (TWO "BOPMEP" PROBLEMS)
  - O AC-01 - 4 INCH AUXILIARY COOLANT LINE FROM SEAL WATER HEAT EXCHANGER (E-34) TO 14 INCH AUXILIARY COOLANT LINE 3037-14"-152N
  - O AC-23 - 6 INCH AUXILIARY COOLANT LINE FROM RHR PUMPS G14 TO HEAT EXCHANGERS E21
- O **METHODOLOGY**
  - O DYNAMIC ANALYSIS USING ME101 COMPUTER PROGRAM
  - O USED ENVELOPE OF APPLICABLE RESPONSE SPECTRA CURVES (.5G SCALED CURVES) IN EACH DIRECTION
- O **RESULTS**
  - O MEETS RTS ACCEPTANCE CRITERIA
    - PIPE STRESS
    - PIPE SUPPORTS
    - VALVE ACCELERATIONS

BPC EVALUATED LARGE BORE PIPING

SALT WATER COOLING PIPING WAS SELECTED BECAUSE IT IS A CAST IRON PIPE WITH A PORTION BURIED IN INSITU SOIL BACKFILL.

BPC EVALUATED LARGE BORE PIPING

- O **PROBLEMS EVALUATED** (BURIED CAST-IRON PIPE)
  - O SW-06 - BURIED PORTION OF 12 INCH CAST IRON PIPE IN SALT WATER SYSTEM FROM SALT WATER PUMP G-13B TO COMPONENT COOLING HEAT EXCHANGER E-20A
- O **METHODOLOGY**
  - O DURING EARTHQUAKE - (BC-TOP-4-A)
  - O POST-EARTHQUAKE USING ME101 PROGRAM TO ANALYZE PIPING FOR SOIL SETTLEMENT CONDITION

O **RESULTS**

- O DURING EARTHQUAKE CONDITION
  MAX CALCULATED PIPE STRESS = 13.5 KSI
- O POST-EARTHQUAKE CONDITION
  MAX CALCULATED PIPE STRESS = 12.1 TO 15.3 KSI
- O RESULTS ARE 70 PERCENT AND 85 PERCENT OF MIN. ULTIMATE TENSILE STRENGTH (S_U) OF CAST IRON.

BPC EVALUATED SMALL BORE PIPING

- O EVALUATED 21 OUT-OF-SCOPE SAFE SHUTDOWN PIPE STRESS CALCULATIONS
- O SELECTION CRITERIA
  - O ALL SMALL BORE STRESS CALCULATIONS IN THE 1982 BOPMEP REPORT WITH STRESSES WHICH EXCEEDED THE BOPMEP REEVALUATION CRITERIA ($2.4S_H$)
  - O EXCLUDED THOSE STRESS CALCULATIONS WHICH HAVE BEEN PREVIOUSLY EVALUATED AS PART OF THE "RTS" HOT SAFE SHUTDOWN SCOPE
- O EVALUATION CRITERIA & METHODOLOGY
  - O NRC APPROVED PROJECT CRITERIA 15691-583 "WALKDOWN CRITERIA FOR EVALUATION OF SAFETY RELATED SMALL BORE PIPING AND TUBING"
  - O USED "AS-IS" DESIGN INFORMATION.

BPC EVALUATED SMALL BORE PIPING

O SUMMARY OF RESULTS

NUMBER OF STRESS CALCULATIONS EVALUATED	SYSTEM	MEETS SMALL BORE RTS CRITERIA
8	AUXILIARY COOLANT	YES
1	CHEMICAL FEED SYSTEM	YES
1	MISC. WATER SYSTEM	YES(1)
3	REACTOR SAMPLE	YES(1)
1	COMPRESSED AIR	YES
3	SAFETY INJECTION	YES
1	MAIN STEAM	YES
3	CIRCULATING WATER	YES

(1) ONE PROBLEM IN EACH SYSTEM HAD SPANS WHICH EXCEEDED THE ALLOWABLE SPANS AND REQUIRED CALCULATIONS TO QUALIFY THE LINES TO THE "RTS" STRESS ALLOWABLES.