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Docket Nos. 50-361
and 50-362

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Gentlemen:

SUBJECT: FIRST ROUND QUESTIONS ON THE FINAL SAFETY ANALYSIS REPORT FOR THE SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

As a result of our review of the Final Safety Analysis Report (FSAR) for the San Onofre Nuclear Generating Station, Units 2 and 3 (SONGS 2 & 3), we find that we need additional information to complete our evaluation. The specific information required is listed in the Enclosure. The numbering system in the Enclosure is a continuation of the system used in the acceptance review questions. Please note that we have not completed our first round review of certain areas of your FSAR. These areas are instrumentation and control, power systems, structural engineering, core performance, geology-seismology, and reactor systems. Additional information in these areas will be requested later.

Our review has identified certain areas of your FSAR which are not acceptable to the staff. To avoid future delays, we wish to advise you of our positions in these areas. These positions, identified by the notation (RSP) next to an item number in the Enclosure, reflect our resolution of safety issues acceptable for a decision concerning the issuance of a construction permit. Accordingly, we request that you amend the SONGS 2 & 3 FSAR to clearly state your intent regarding compliance with each of these positions. We are prepared to meet with you to assure your complete understanding of our positions and our bases for them, if necessary.

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SURNAME ➤						
DATE ➤						

To maintain our licensing schedule for the SONGS 2 & 3 FSAR, we need your responses to the items in the Enclosure by August 19, 1977. If you cannot meet this requirement, please inform us of the date you plan to meet so that we may revise our schedule accordingly.

Please contact us if you have any questions about the information requested.

Sincerely,

Original Signed by

Karl Kniel, Chief
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosure:
Request for Additional
Information

cc w/enclosure:
See page 3

OFFICE➤	DPM:LWR #2	DPM:LWR #2				
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FIRST ROUND

REQUEST FOR ADDITIONAL INFORMATION

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 AND 3

DOCKET NOS. 50-361, 50-362

005.0 AUXILIARY SYSTEMS BRANCH

005.1
NONE

In Table 3.2-1 and Figure 9.3-5, Sheets 1 and 2, the letdown line of the chemical and volume control system from isolation valves 004-C-105 and 023-C-105 (adjacent to backpressure control valves 2PV-0201A and 2PV-0201B) to the volume control tank outlet valve 2LV-0227B, is designed to non-seismic Category I requirements. Demonstrate that in the event of a Safe Shutdown Earthquake there is an adequate plant procedure for this situation. Provide a detailed discussion of your cold shutdown procedure for this event and identify any differences in this procedure from that of normal plant shutdown.

005.2
NONE

In Figure 9.3-6 of the Primary Sampling System, the four sample lines from the reactor coolant piping and pressurizer within containment are incorrectly classified as non-seismic Category I. It is our position that these sample lines within containment which are part of the reactor coolant pressure boundary should be designed to seismic Category I requirements. Revise Figure 9.3-6 accordingly.

010.0 AUXILIARY SYSTEMS BRANCH - SECTION B

010.20
(3.4)

You state in Section 4.3.1.2 of the FSAR that rooms containing non-seismic Category I system components and pipes whose rupture could result in flood damage to equipment important to safety have level alarms that alarm in the control room. Provide design classifications of the level monitoring and alarm systems and their capabilities of meeting the single failure criterion. Also describe the action that will be taken to prevent safety related equipment from being flooded, taking into account time for operator manual action; namely, 20 minutes manual action time should be assumed if only a single action is required inside the control room or 30 minutes manual action time should be assumed if there is more than one operator action required inside the control room.

010.21 (RSP)
(3.6)

Your response to our request 010.4 is not acceptable. It is our position that you must follow the guidance provided in the December, 1972 letter from A. Giambusso and the corrections in the errata sheet for "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" that was transmitted to Mr. J. Moore and Mr. Engler from Mr. K. Goller dated January 22, 1973. Also, an analyses made in conformance with B.3 of our Branch Technical Position APCSB 3-1 must be presented to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided.

010.22
(9.0)

Provide a tabulation of all valves in the reactor pressure boundary and in other seismic Category I systems (per Regulatory Guide 1.29) e.g., safety valves, relief valves, stop valves, stop-check valves, and control valves whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of a transient or accident. The tabulation should identify the system in which it is installed, the type and size of valves, the actuation type(s), and the environment of conditions to which the valves are qualified.

010.23 (RSP)
(9.1.1)

It is our position that the spacing between fuel assemblies in the new fuel storage racks should be sufficient to maintain the array, when fully loaded and flooded with non-borated water, in a subcritical condition, i.e., K_{eff} of less than 0.95. Provide additional information in your FSAR to demonstrate that the above criteria are met.

010.24 (9.1.1) (9.1.2) — Provide results of an analysis to demonstrate that the new and spent fuel storage racks and the anchorages can withstand the maximum uplift forces available from the crane without an increase in K_{eff} and verify that the vaults and racks have been designed to preclude damage from dropped heavy objects.

010.25 (9.1.3) — Section 9.1.3 of the FSAR states that the makeup to the spent fuel pool is from the seismic Category I refueling water storage tank. Show this makeup line in your Figure 9.1-1 accordingly.

010.26 (9.1.4) — In addition to the fuel handling arrangement in Figure 9.1-16, provide additional drawings including section views that show more clearly the relationship between the spent fuel pool and the spent fuel cask area. These drawings should be sufficiently detailed to demonstrate that the spent fuel cask cannot be hoisted to a height that would permit possible tipping or swinging of the cask over the spent fuel storage pool during cask handling.

010.27 (3.2.1) (9.1.4) — Table 3.2-1 of the FSAR indicates that the refueling machine, spent fuel handling machine, new fuel crane and new fuel elevator are not designed to seismic Category I standards. Expand Section 9.1.4.3 of the FSAR to discuss the consequences of a failure due to a postulated SSE of the above fuel handling equipment to essential fuel handling and storage facilities in accordance with Position C.2 of Regulatory Guide 1.29.

010.28 (9.2.1) — Table 9.2-1 of the FSAR indicates that the salt water cooling system required flow for LOCA conditions is 34,000 gpm. Section 9.2.1.2 states that each pump with a capacity of 17,000 gpm is capable of providing 100 percent of the cooling flow required to mitigate the effects of any design basis accident including a LOCA. Provide clarification for this discrepancy.

010.29 (RSP) (9.2.2) — Your response to our request 010.13 is not acceptable. It is our position that the you must modify the design of the component cooling water system supplying cooling water to the reactor coolant pumps to follow the guidance set forth in our request 010.13. Your response also stated that you do not consider moderate energy line cracks to be design criteria for San Onofre Units No. 2 and No. 3. Refer to our request 010.21. Failures in moderate energy component cooling water piping systems should be considered for San Onofre Units No. 2 and No. 3 design.

010.30 (9.2.2) Section 9.2.2 of your FSAR indicates that the reactor coolant pumps and the spent fuel pool heat exchangers are supplied component cooling water (CCW) from a non-essential loop of the CCW system which will be isolated under accident conditions. Describe the provisions made to reopen the non-critical loop isolation valves so that the CCW can be supplied to the fuel pool heat exchangers before boiling in the spent fuel storage pool occurs.

010.31 (9.2.2) Describe and provide the design bases for the makeup water source to the surge tank of the component water system. Include the seismic category classification of the makeup system.

010.32 (9.2.5) Your response to our request 010.11 indicates that the traveling water screen for the salt water system is not designed to seismic Category I requirements. Also Section 9.2.5 of the FSAR states that the offshore intake and outfall conduit structures are not designed to seismic Category I requirements. Provide results of an analysis to demonstrate that the salt water cooling pumps will be provided with required suction flow under a postulated LOCA assuming the effects of failure of non-seismic designed components and loss of offsite power conditions. The analysis should also include the effects of a postulated failure of non-seismic designed seawall which is located above the intake and outfall conduits.

010.33 (9.2.6) Section 9.2.6.3 of your FSAR indicates that the condensate storage capacity for the auxiliary feedwater supply is based on a cooldown time of about 3 1/2 hours. Your response to our request 010.14 refers to a cooldown time of 4.2 hours assumed in the accident analysis. Clarify this deviation to assure that the condensate storage capacity is sufficient for plant cold shutdown.

010.34 (9.3.4) Provide page 9.3-36 of the FSAR for our review. This page is now missing from the FSAR.

010.35 (9.4.2) Figure 9.4-8 of the FSAR indicates various points of the control room complex emergency HVAC system where the air enters or leaves the system on the same drawing. Provide additional information to clearly identify the points that should be interconnected on continuation drawings (i.e. match points).

010.36 (9.4.2) Figure 9.4-8 of the FSAR indicates that there are no isolation dampers between the emergency air cooling system and the normal air cooling system for the ESF switchgear rooms. Revise the P&ID and the description of the ESF switchgear room cooling system to demonstrate that provisions are made to isolate the normal cooling system when the emergency cooling system is in operation.

010.37 (9.4.2) Section 9.4.2 of the FSAR indicates that two emergency exhaust fans serve all four battery rooms for each unit. Each emergency fan connects to two battery rooms. A single failure on one emergency exhaust fan will result in loss of exhaust capability of two battery rooms. This is not acceptable, since hydrogen may buildup in these battery rooms. Modify the system to provide continuous exhaust for all battery rooms under loss of offsite power conditions and meet the single failure criterion.

010.38 (9.4.3) In Section 9.4.3.1 of the FSAR you state that the fuel handling building (FHB) normal ventilation system isolation dampers are pneumatically operated. Figure 9.4-9 indicates that these isolation dampers are motor operated. Clarify this discrepancy in your FSAR. If they are air operated, describe the safety classification of the supply air and the failure mode of the isolation dampers in case of loss of air supply. If they are motor operated, describe the electric power and instrumentation supply to the isolation dampers and demonstrate that a single electric failure assuming loss of offsite power will not prevent positive isolation of the FHB normal ventilation system.

010.39 (9.4.3) Section 9.4.3.1 of the FSAR states that the post-accident cleanup units and fuel pool pump room cooling units are powered from a vital bus. Clarify this statement to confirm that the redundant cleanup units and redundant cooling units are powered from separate vital buses to meet the single failure criterion.

010.40 (9.4.3) You stated that the description of the design of the diesel generator building ventilation system would be submitted in April, 1977. This information has not be received. This information is necessary for us to evaluate this system.

010.41 (10.3) Identify on Figure 10.1-1 of the FSAR, the seismic Category I portion of the control air system for operation of the power operated atmospheric steam dump valves and identify the interface between the seismic Category I nitrogen supply line and the non-seismic Category I instrument air supply line.

010.42 (RSP) (10.4.5) In your response to our request 010.17, you have assumed an operator reaction time of 5 minutes and 38 seconds from the first flood alarm to trip the circulating water pump motor in a failed line. This is not acceptable. It is our position that 20 minutes manual action time should be assumed (from the time of break) if only a single action is required inside the control room to trip the circulating water pump motor or 30 minutes manual action time should be assumed if there is more than one operator action required inside the control room. Re-evaluate the postulated circulating water system failure based on the above stated manual action times.

010.43 (10.4.9) Provide additional detailed P&IDs of the auxiliary feedwater systems to indicate design features such as the following:

(1) The alternative water supplies to the auxiliary feedwater pumps.

(2) Show the steam supply lines from the main steam lines upstream of the MSIVs to the turbine driven auxiliary feedwater pump turbine and indicate that the motor operated valve on the steam supply line is powered from D/C power sources and the air operated valves are operated by safety grade air supplies.

010.44 (10.4.9) Provide the results of an analysis to demonstrate that your auxiliary feedwater pump size and condensate storage capacity for the auxiliary feedwater supply is sufficient to prevent overheating and subsequent overpressurization of the primary coolant system under all postulated accident conditions including the following situations:

(1) Main steam or feedwater line failure and both auxiliary feedwater pumps start to pump water through the break until the operator manually isolates the steam generator with the broken line. It is our position that 20 minutes manual action time should be assumed if only a single action is required inside the control room to isolate the line break or 30 minutes manual action time should be assumed if more than one operator action is required inside the control room.

(2) A moderate energy line crack at the condensate supply line close to the tank T 121.

022.0 CONTAINMENT SYSTEMS BRANCH

022.14 In response to Question 022.1 reference was made to CESSAR to justify
(6.2.1.1) that it was conservative to assume maximum safety injection for the design basis LOCA. The San Onofre 2 & 3 LOCA analysis, however, does not utilize CESSAR mass and energy release data. Additionally, the acceptability of the assumption of maximum safeguards is contingent upon plant dependent parameters such as containment net free volume and containment active and passive heat removal capability. Therefore, provide containment response analyses assuming minimum safety injection.

In addition to the assumption of safety injection system performance, the mass and energy release to the containment following a LOCA is dependent upon the containment backpressure. The conservatism of a high or low containment backpressure assumption is dependent on plant parameters similar to those impacting the assumption of safety injection system performance. Therefore, provide justification that the assumed backpressure for both reflood and post reflood mass and energy release calculations is conservative for determining the design basis LOCA, or provide containment response analyses using mass and energy release rates conservative for San Onofre 2 & 3.

022.15 (6.2.1.1) The response to Question 022.6(a) is inadequate. A spectrum of MSLB cases including various break sizes at different power levels should be provided to assure identification of the worst case for both maximum containment atmosphere temperature and pressure. Therefore, provide the originally requested information.

022.16 (6.2.1.1) The response to Question 022.7 contends that consideration of single active failures of a main steam isolation valve or main feedwater isolation valve is unnecessary because no single active failure could render the valves inoperable. It is our position, however, that a valve is an active component and any failure must be considered an active failure in the single failure analysis for the calculation of mass and energy release to the containment following a postulated MSLB. Therefore, provide MSLB analyses considering single active failures of the main steam and main feedwater isolation valves.

022.17 (6.2.1.1) Describe and justify the method of calculating the discharge of the volume of fluid between the main feedwater line isolation valve and the ruptured steam generator to the containment for the postulated MSLB accident. Provide a table of the additional mass and energy releases.

022.18 (6.2.1.1) Provide the following information regarding the environmental qualification of safety related equipment.

a. Provide a comprehensive list of equipment required to be operational in the event of a main steam line break (MSLB) accident to mitigate the accident consequences and assure a safe shutdown of the plant. The list should include, but not necessarily be limited to, the following safety related equipment:

1. Electrical containment penetrations
2. Pressure transmitters
3. Containment isolation valves
4. Electrical power cables
5. Electrical instrumentation cable
6. Level transmitters

Describe the qualification testing that was done, including the test environment, namely, the temperature, pressure, moisture content, and chemical spray as a function of time.

b. It is our position that the thermal analysis of safety related equipment which may be exposed to the containment atmosphere following a main steam line break accident should be provided based on the following:

1. A condensing heat transfer coefficient based on the recommendations in Branch Technical Position CSB 6-1, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation should be used.

2. A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period it is appropriate to use a conservatively evaluated forced convection heat transfer correlation. For example:

$$Nu = C(Re)^n$$

where

Nu = Nusselt No.

Re = Reynolds No.

C,h = emperical constants dependent on geometry and Reynolds No.

Since Reynolds number is dependent on velocity, it is necessary to evaluate the forced flow currents which will be generated by the steam generator blowdown. The CVTR experiments provide limited data in this regard. Convective currents of from 10 ft/sec to 30 ft/sec were measured locally. We recommend that the CVTR test results be extrapolated conservatively to obtain forced flow currents to determine the convective heat transfer coefficient during the blowdown period. After the blowdown has ceased or been reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable.

3. For each component where thermal analysis is done in conjunction with an environmental test at a temperature lower than the peak calculated temperature following a main steam line break accident, compare the test thermal response of the component with the accident thermal analysis of the component. Provide the basis by which the component thermal response was developed from the environmental qualification test program. For instance, graphically show the thermocouple data and discuss the thermocouple locations, method of attachment, and performance characteristics, or provide a detailed discussion of the analytical model used to evaluate the component thermal response during the test. This evaluation should be performed for the potential points of failure such as thin cross-sections and temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could result in the failure of the component mechanically or electrically. If the component thermal response comparison result in the prediction of a more severe thermal transient for the accident conditions than for the qualification test, provide justification that the affected component will perform its intended function during a MSLB accident, or provide protection for the component which would appropriately limit the thermal effects.

022.19 (6.2.1.1) Table 6.2-15 lists assumptions used in the analysis of inadvertent spray system actuation. Describe how the heat transfer coefficients are used to model the heat sinks, and justify the conservatism of the assumed values of the heat transfer coefficients.

022.20 (6.2.1.1) It is our position that instrumentation capable of operating in the post-accident environment should be provided to monitor the sump water temperature following an accident. This position is consistent with Standard Review Plan Section 6.2.1.1.A and Regulatory Guide 1.97. Therefore, discuss your plans for providing the appropriate post-accident monitoring instrumentation.

022.21 (6.2.1.2) The response to Question 022.8 concerning the subcompartment analysis is incomplete. It is our position that nodalization sensitivity studies should be performed to verify that differential pressures both for the structural design and for use in the design of component supports, have been conservatively calculated. Therefore, provide the results of nodalization sensitivity studies which show the above criteria have been satisfied.

022.22 (6.2.2) With regard to the design evaluation of the containment emergency fan cooler system, provide the following additional information:

(1) Justify the bases for the cooling water temperature used to determine the heat removal capacity presented in Figure 6.2-27.

(2) Provide a curve of heat removal capacity versus containment temperature assuming the highest component cooling water temperature. Provide justification for the assumed cooling water temperature.

022.23
(6.2.4)

Your response to Item 022.12 indicates that the large volume purge system will be used during modes of operation when containment integrity is required. It is our position that the technical specifications addressing limiting conditions for operation (Section 16.3.6.5.1) should include the reactor operational modes 3 and 4; i.e., hot standby and hot shutdown. Additionally, we will require that Item 1.a of Branch Technical Position CSB 6-4, regarding valve operability, be satisfied.

022.24
(6.2.5)

Provide the bases for the zinc and aluminum corrosion rates presented in Table 6.2-35. Your response should include references if the supporting literature is generally available, or a complete discussion of the test program conducted and the applicability of the test data.

022.25
(6.2.6)

It is stated in Section 6.2.6.1 that valves which remain open throughout the design basis LOCA need not be tested. It is our position that all containment isolation valves; i.e., valves provided to satisfy the requirements of the General Design Criteria, should be leak tested, either pneumatically or hydrostatically. Hydrostatic leak testing is permissible if a water seal similar to the seal water system described in Appendix J to 10 CFR Part 50 can be shown to exist. Therefore, discuss your plan for complying with this position.

Provide a table listing all containment isolation valves and indicate which will be pneumatically tested and which will be hydrostatically tested. Additionally, identify all valves for which the pneumatic test pressure will not be applied in the same direction as the pressure existing when the valves are required to perform their safety function.

022.26 (6.2.6) Table 6.2-36 identifies those systems that will not be vented and drained for the Type A test. It is our position that the containment isolation valves in these lines should be locally leak tested, unless hydrostatic leak testing can be justified, and the measured leakage added to the Type A test result. Discuss your plans for complying with this position.

022.27 (6.2.6) Identify all penetrations which will not be subjected to Type B leak rate testing and provide justification exempting the penetrations from Type B testing.

022.28 (6.2.6) Appendix J requires that containment piping penetrations fitted with expansion bellows be periodically leak tested at Pa. Verify that penetrations fitted with expansion bellows have the design capability for Type B testing. Where more than one bellows is utilized on a penetration, provide assurance that each bellows be subjected to a Type B test.

112.0 MECHANICAL ENGINEERING BRANCH

112.1 (3.6.2.4) The FSAR states that if guard pipes are to be used for containment penetrations, the design criteria will be provided in an amendment to the FSAR by approximately November, 1978. The staff will require that guard pipe design criteria utilized for the SONGS 2, 3 will provide a level of design conservatism and safety equivalent to the criteria described in Section 3.6.2 of the NRC Standard Review Plan. Provide your commitment to conform with this position.

112.2 (3.6.2.5) The staff cannot complete its review of the FSAR until results of analyses to be provided in Appendix 3.6A are reviewed and approved. In addition, we will require that the pipe break sizes calculated for SONGS 2,3 be based specifically on S.O. support and restraint configuration or that the S.O. parameters are within the range of parameters specified in Table 4-1 of Reference 1 (CENPD-168).

112.3 (3.9.1.4) Paragraph 3.9.1.4.1.1 states that inelastic analysis of components and their supports and pipe restraints is permitted only during pipe rupture conditions and is limited to elements in the reactor coolant cold or hot leg. It is further stated that calculated inelastic strains produced by the faulted condition are limited to 50% of the ultimate strain.

The current NRC position is that components and their supports subjected to faulted condition loads be designed to the stress limits specified in Appendix F and Subsection NF of ASME Section III, respectively. Furthermore the criteria of 50% of uniform ultimate strain applies only to pipe whip restraints. Revise the referenced paragraph to conform with the acceptable criteria.

112.4 (3.9.1.4.2) Specify those safety related non-NSSS components where simplified (elastic) methods do not adequately characterize the component dynamic response and elastic-inelastic time history methods used to determine dynamic response.

112.5 (3.9.2.1) The proposed preoperational vibration, thermal expansion and dynamic effects test program should include: (1) all high energy and Category I moderate energy lines outside containment in addition to Class 1, 2 and 3 piping, (2) provisions to verify operability of snubbers by recording hot and cold positions, (3) a commitment to present the detailed plan to the NRC at least 60 days prior to initiation of the test program. Revise Subsection 3.9.2.1 in the FSAR to be consistent with the above criteria.

112.6 (3.9.1.4) (3.9.2.5) (3.7.3.14)

The FSAR states that the CEDM's are seismically supported by the head lift rig assembly through a series of shock arrestors. In order to account for the interaction of 91 CEDM's with the supports, an indepth program was developed to study such effects. The program described contains the following elements: (1) a model dynamic analysis and test results to verify validity of the analytically derived characteristics, (2) a proof test of one or more CEDM's, seismically supported to verify component structural integrity and analytical results.

The description of the support system for reactor CEDM's is a unique feature of the SONGS 2, 3 facility. It is also evident that reliability of the shock arrestors (snubbers) is crucial to operability assurance of the CEDM's under faulted condition loads. Results of your program to be presented in August 1977 should include the following information:

(1) A graphical description of the SONGS 2, 3 structural arrangement showing the CEDM support system and components.

(2) The corresponding model utilized for structural analysis of the system and components.

(3) A description of the seismic arrestors and any unique features of the arrestors associated with the analysis, design, development, testing, construction or installation of the arrestors.

(4) A description of the tests and test results used to verify the analytical model.

(5) A description of the CEDM and support proof test configuration, proof test loads and test results.

112.7 (3.9.2.4)

Maine Yankee and Fort Calhoun are designated jointly as the prototype for the SONGS 2, 3 reactor internals and the design similarities noted. However, both Main Yankee and Fort Calhoun have thermal shields, whereas SONGS 2, 3 do not. The Arkansas One - Unit No. 2 reactor, like the San Onofre units has no thermal shield, is also a two loop plant and parameters cited in the FSAR as significant such as mass flow rate and pump characteristics are similar. The prototype designation is conditionally acceptable to the staff. The basis for the conditional acceptance is that results of the ANO-2 augmented internals inspection are expected to verify satisfactory performance of the ANO-2 internals as predicted by the emperical evaluation. However, should the ANO-2 inspection indicate the need for any corrective action, the staff will require the same corrective action for the SONGS 2, 3 reactor internals design.

Provide a commitment to the above requirement in the SONGS 2 and 3 FSAR.

112.8 (3.9.2.5) The dynamic system analyses of the SONGS 2, 3 reactor internals under faulted condition loads is not complete. Results of the non-linear response analysis due to SSE horizontal and vertical excitation described in Section 3.7.3.14 must be completed to demonstrate structural integrity of the reactor internals components under faulted condition loads. The FSAR states that results of the seismic analyses will be provided approximately in April 1978. This date is not compatible with the scheduled date for completion of the safety evaluation report (SER). The staff cannot complete its review prior to issuing the SER and will cite reactor internals structural integrity as an incomplete and unresolved issue in the SONGS 2, 3 application unless this information is submitted prior to the current staff SER schedule date of February 1978.

112.9 (3.9.2.7) (3.9.3.1) The FSAR states that reactor vessel support loads due to (i) pipe break thrust force, (ii) pressure drop across the core and (iii) asymmetric cavity pressure, will be completed in September 1977. The corresponding structural analysis to verify structural integrity of the vessel supports, according to paragraph 3.9.3.1, will not be completed until January 1979. The staff cannot complete its review until all the information has been submitted. In addition, the January 1979 date is incompatible with the current staff safety evaluation report (SER) schedule of February 1978. RPVS structural integrity will not have been demonstrated and will be cited as an incomplete and unresolved issue in the SER unless this information is reviewed prior to February 1978.

112.10 (3.9.3.1) (3.9.5.4) The FSAR states that a summary of maximum total stresses, usage factors and deformations and identification of items within 10% of the allowables would require increased staff review without improving plant safety. The staff does not agree with this contention. The results of analyses of Class 1, 2, 3 components and reactor internals alone do not assure plant safety, however these analyses are vital elements of the total information package which provides the basis for the staff's safety evaluation report for the SONGS Unit Nos. 2 and 3.

The staff will require that forthcoming results of analyses of Class 1, 2 and 3 safety related components, be summarized in the FSAR. In addition, for the primary reactor internals, a summary table should be provided which includes (1) alisting of the primary reactor internals components, (2) the maximum calculated stress and displacement, (3) the critical load combination producing the maximum stress and deformation of the component,

(4) corresponding stress and displacement limits, (5) the estimated natural frequency, forcing frequencies, maximum stress cycles and corresponding endurance limit.

112.11 (3.9.3.1) Amplify the discussion of loading combinations applicable to Class 2 and 3 NSSS components and supports and provide the basis for the statement that pipe break loads are not considered for Code Class 2 and 3 components.

112.12 (3.9.3.2) The FSAR states that the operability assurance verification results for some pumps will be submitted in June, 1977 (NSSS) and others in February 1978 (non-NSSS). The staff cannot complete its review of the SONGS 2, 3 operability assurance programs until the results of operability verification of all active pumps and valves have been submitted. Unless this information is submitted prior to the staff SER scheduled date of February 1978, this issue will be identified as unresolved in the SONGS 2 and 3 SER.

112.13 (3.9.3.3) The staff cannot complete its review until the analysis of closed discharge system relief valve piping has been completed. The February 1979 date projected for completion of the analyses is not compatible with the safety evaluation report (SER) completion date. This item will be cited as an incomplete and unresolved issue in the SER pending completion and acceptance of the analyses.

112.14 (3.9.4.4) The FSAR states that a qualification test was completed on the first C-E production magnetic jack control element drive mechanism (CEDM). State whether the first C-E production magnetic jack CEDM is identical to the SONGS 2, 3 CEDM and if not, whether similar qualification testing will be conducted for the SONGS 2, 3 CEDM design.

112.15 (3.9.3.4) (5.4.14) The FSAR does not discuss snubbers used as component supports in sufficient detail. The staff's primary concern is that the structural aspects of snubber utilization as component (and piping) supports in safety related systems be fully assessed for all load conditions under which the snubbers are actuated. If snubbers are used as component supports in your facility we require the following information to complete our review of the FSAR:

(1) Provide the basis for selecting the location, required load capacity, structural and mechanical performance parameters of safety related snubbers and achieving a high level of operability assurance including:

(a) A description of the analytical and design methodology utilized to develop the required snubber locations and characteristics.

(b) Criteria for construction of snubbers (e.g., ASME Section III Subsection NF).

(c) A discussion of design specification requirements to assure that required structural and mechanical performance characteristics and produce quality are achieved.

(d) Procedures, controls to assure correct installation of snubbers and checking the hot and cold settings during plant start-up tests.

(e) Provisions for accessibility for inspection, testing and repair or replacement of snubbers.

(2) A tabulation of snubbers utilized in your facility as supports for safety related systems and components including:

(a) Systems Identification and Location
(b) Type (hydraulic, mechanical)
(c) Fabricator and Rated Load Capacity
(d) Function (shock or vibration arrestor, dual purpose)

(3) Description of snubber suppliers performance qualification tests and load tests.

(4) System and component structural analysis showing:

(a) Structural analysis model.

(b) Description of the characterization of snubber mechanical properties used in the structural analysis including considerations such as (i) differences in tension and compression spring rates, (ii) effect of entrapped air and temperature on fluid properties, (iii) other factors affecting snubber characteristics.

(c) List load conditions and transients analyzed.

(d) Maximum snubber loads, corresponding piping or component stresses.

(e) Comparison of computed loads with rated snubber load.

112.6 (3.9.2) (3.10.1) The seismic qualification program presented in Sections 3.9.2.2 and 3.10 of the FSAR is not entirely in accord with current NRC requirements for the seismic qualification of mechanical and electrical equipment and instrumentation. Since the review of the SONGS 2 and 3 PSAR, there have been many changes in seismic qualification criteria as is evidenced by the differences between industry seismic qualification standards IEEE-344-71, referenced in the PSAR, and IEEE-344-75, referenced in PSAR's currently under NRC review. The staff recognizes that Category I equipment for this plant will as a minimum be qualified in accordance with the criteria specified in the PSAR. However, due to the referenced changes in qualification criteria since the review of the SONGS 2 and 3 PSAR, the staff is concerned that some components and supports qualified in accordance with minimum PSAR criteria may not have adequate margin to perform their intended design functions during the postulated DBE event applicable for the plant. To provide assurance to the public that Category I components have adequate margins to perform their safety functions under DBE conditions, when qualified to criteria of SONGS 2 and 3 PSAR vintage, the staff has instituted a seismic qualification review program which includes a site visit and a more detailed review of the implementation of the seismic qualification criteria.

CE Topical Report CENPD-182 is referenced for seismic qualification of NSSS equipment. Appendix 3.10 describes qualification procedures for BOP equipment. Staff review of CENPD-182 has not yet been completed and the topical report is not at this time a complete acceptable reference for SONGS 2 and 3. Appendix A of the topical report does not contain a list of equipment supplied by CE for SONGS 2 and 3.

The review of CENPD-182 is included in the staff seismic qualification review program. This program is currently being conducted on a generic basis with Combustion Engineering and Bechtel. Site visits by members of the staff to all plants currently under review for an operating license is also part of this program.

(1) Provide a commitment in the SONGS 2 and 3 FSAR to participate in the staff seismic qualification review program.

(2) To assist the staff in evaluating the current status of the equipment in SONGS 2 and 3, provide in Appendix 3.10B of the FSAR a listing of all safety related electrical equipment supplied by the BOP. Include the primary Class 1E function of the equipment, the method of qualification and a summary of results of the qualification

procedure. An acceptable format for this information can be found in Table 3.10-1 of the Edwin I. Hatch Nuclear Plant, Unit No. 2 FSAR (Docket Number 50-366).

112.17 (5.4.2.3) The FSAR states in Subsection 5.4.2.3.1.3 that structural integrity of steam generator tubes and tube supports under combined LOCA and SSE and to establish minimum tube wall thickness will be submitted in August 1977. The staff cannot complete its review until this analysis is submitted.

112.18 (5.4.14.3) The statement in Section 5.4.14.3 that the analysis to verify the adequacy of reactor vessel supports under the subcompartment and reactor internals loads will be submitted in July 1977 contradicts the statement in paragraph 3.9.3.1 that the structural analyses will be submitted in January 1979. Provide a clarification of this issue and indicate when the structural analysis of the reactor vessel supports due to the combined loads discussed in Q 112.9 will be submitted for review.

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.3 (5.2.4.1) Considering your CP date of October 1973, your preservice inspection program will be conducted to the more current 1974 edition of Section XI, Summer 1975 Addendum and inservice inspection program will comply with the requirements in the editions of the ASME B & P V Code and Addenda in effect no more than six months prior to the starting date of commercial operation. However, you state that the ASME Code Class 1, Class 2 or Class 3 shall meet the requirements of the Section XI of editions of ASME B & P V Code and Addenda to the extent practical within the limitations of design, geometry, and materials of construction of the components.

The inservice inspection plan based on current requirements will be reviewed to support the safety evaluation report finding on ISI. Your response should define all the examination requirements that you determine are not practical within the limitations of the design, geometry, and materials of construction of the components. Particular attention should be directed to impractical examinations resulting from revised Section XI requirements in Section 50.55a, paragraph (b), such as examinations that will result in high radiation exposure to personnel without a commensurate increase in safety (1) known inaccesible regions due to component arrangement, (2) restricted access to welds in accepted ASME Code weld geometry designs, and (3) limitations in examination methods or procedures due to metallurgical properties in approved materials of construction.

Discuss the inservice inspections (or testing) that will be performed in lieu of the ASME Code Section XI requirements that you determine to be impractical. The technical justification to support your conclusions should contain as a minimum (1) the identification of the applicable ASME Code edition(s) and subsection(s), (2) the number of components, (3) the safety significance of postulated failure at the inspection location, (4) the Section XI examination category, (5) the examination method, (6) the degree of conformance, and (7) the system modifications, equipment or conditions that would be necessary for total compliance.

The updated inservice inspection plan should be submitted for review within six months of anticipated commercial operation to demonstrate compliance with 10 CFR Part 50, Section 50.55a, paragraph (g). This plan will be evaluated in a safety evaluation report supplement. The objective is to supplement the previously submitted inservice inspection plan to incorporate (1) Section XI requirements in effect six months prior to commercial

operation and (2) any augmented examination established by the Commission. Your response should define all examination requirements that you determine are not practical within the limitations of design, geometry, and materials of construction of the components.

121.4 (5.3) Your prediction of RT_{NDT} shift and Charpy upper shelf energy decrease with radiation are based on the curves shown in Figures 5.3-1 to 5.3-3. However, we will estimate the intial heatup and cooldown limit curves according to Regulatory Guide 1.99, Revision 1 (April 1977). Appropriate adjustments may be made when actual surveillance samples are withdrawn and tested.

121.5 (5.4.1.4) In your discussion of compliance with Regulatory Guide 1.14 and acceptance criteria, the inservice inspection is not addressed. Confirm that the inservice inspection of the pump flywheel will conform to Regulatory Guide 1.14.

121.6 (6.6.8) Augmented inservice inspection program will be implemented on "as practical basis." You should compare your "practical basis" with the requirements of SRP, Section 6.6, II.8 and furnish the staff with all details for our review.

122.0 MATERIALS ENGINEERING BRANCH - METALLURGY SECTION

122.1 Describe the methods used to monitor the secondary coolant purity and show that these are at least as conservative as the positions given in the Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in Pressurized Water Reactor Steam Generators," referenced in the NRC Standard Review Plan, NUREG 75/087, Section 5.4.2.1.

221.0 REACTOR ANALYSIS SECTION, ANALYSIS BRANCH

221.1 (4.4.4.2) Provide a summary of the test data from the hydraulic tests on a 1/5 scale reactor vessel model and 1/8 scale model. The summary should include the results of measurements of flow distribution to the reactor core inlet and outlet, the results of measurements of pressure loss from the inlet to the outlet of the vessel and between significant intermediate points, and the results of measurements of the fluid mixing that occurs between the vessel inlet nozzles and the core inlet and between the inlet and outlet of the core. Further information supplied should include a description of the test apparatus, a listing of the measured variables and the location of the sensors, a summary of the uncertainties in the test data and a description of how these data were applied in the San Onofre 2/3 design.

221.2 (4.4.4.2) Provide an explanation of how the effects on the core flow and pressure drop of possible crud deposits are included in the thermal-hydraulic design.

Provide a description of the instrumentation available and the surveillance requirements and procedures which would alert the reactor operator to an abnormal core flow or core pressure drop during steady-state operation.

221.3 (4.4.2) Combustion Engineering has submitted a topical report (CENPD-225) on fuel and poison rod bowing which presents results of tests performed on a 21-rod bundle of electrically heated rods and an unheated guide tube. Results were presented for rods in full contact and partially bowed rods. The data show that plant thermal margins may be less than intended. Discuss how this data will be applied to San Onofre 2/3, including the application of any anticipated penalties.

221.4 (4.4.4.2) Provide an evaluation of the effect of operation with one or more reactor coolant pump(s) out of service for all permissible operating modes.

221.5 (4.4.4.2) Provide the results of fuel assembly flow tests applicable to the San Onofre 2/3 design to verify the values of the loss coefficients for the upper and lower end fittings and spacer grids.

221.6
(4.4.6)

Describe your proposed procedures and equipment, both temporary and permanent, for vibration monitoring. If your total system is to be described with the standard vibration and loose parts monitoring system description to be supplied later, you may so indicate in response to this question.

222.0 Systems Analysis Section, Analysis Branch

222.1 (6.2.1) Provide a description of all changes made to the computer programs used to calculate mass and energy release to the containment for San Onofre 2 and 3 from the versions described in CESSAR which you reference. The mass and energy release codes are CEFLASH-4A, CEFLASH-4, FLOOD-MOD2, the post-reflood model and SGN-III.

222.2 (6.2.1) Page 6.2-143 indicates that a feedwater flashing model was added to the SGN-III code. Provide all equations and assumptions associated with this model.

222.3 (6.2.1) For the analysis of the main steam line break:

a. provide a description and justification for the method by which the steam in the unisolated steam piping is added to the containment assuming failure of the main steam isolation valve adjacent to the ruptured steam generator.

b. provide a description and justification for the method by which the feedwater in the unisolated feedwater piping is added to the containment assuming failure of the feedwater isolation valve adjacent to the ruptured steam generator.

222.4 (6.2.1) The main steam line break analysis for San Onofre 2 and 3 assumes constant feedwater flow until the isolation valve begins to close. Applicants for plants similar to San Onofre 2 and 3 have determined that the feedwater flow rate to the affected steam generator will increase by approximately a factor of 2 following a main steam line break. The increase in feedwater flow is a result of the decrease in feedwater pump discharge pressure.

Provide analyses of the main feedwater flow transient into the affected steam generator for the most severe break size at each power level. Analyses should be provided for cases when the main feedwater isolation valve closes and when it fails to close.

Provide all equations and assumptions used by the analytical method. Discuss how the additional feedwater flow will be added to the containment.

312.0 SECTION B, ACCIDENT ANALYSIS BRANCH

312.23 (3.5)

1. The strike probabilities are presented in Tables 3.5-5 and 3.5-6 as a function of turbine wheel number. As such, the strike probabilities do not appear to be dependent on wheel position. For example, inspection of the Unit No. 2 turbine-containment configuration, as in Figure 3.5-7, would indicate that an end wheel burst would yield a different probability for striking the containment depending on whether it was located on the northern or southern end of the No. 3 LP turbine. Discuss the effect of wheel position along the turbine on the strike probabilities of the safety related targets.

2. In an assumed four-piece wheel burst the probability of a missile having the "correct" elevation angle for striking a target which subtends an angle of $\Delta \emptyset$ is $\Delta \emptyset/(\pi/2)$ rather than $\Delta\emptyset/2\pi$ as shown in Question (4) of Section 3.5.1.3.3.3 of the FSAR. This would indicate that the strike probabilities in Tables 3.5-5 and 3.5-6 are four times too low. Revise your P_2 estimates to reflect this consideration.

3. Indicate the nominal thicknesses and material strengths for any barriers which would be encountered by the turbine missiles which are eligible for striking the safety related equipment indicated by the shaded areas in Figures 3.5-7 through 3.5-13.

312.24 (6.5)

1. You have indicated a possible low sump pH of 8 in Figure 6.5-5. This pH will give an elemental iodine partition coefficient of about 1570. Since the water volume to air volume ratio is 4 to 100, the decontamination factor (DF) will be limited to 70 or less. Higher DFs can be justified only if the sump pH were increased. We will use a DF of 70 in our LOCA dose analysis unless appropriate modifications are provided for increasing the sump pH. Provide a discussion describing the steps you will take regarding this item.

2. Provide a description of tests to be done to assure the quality of the NaOH solution in the chemical additive tank.

3. Describe the method for finding nozzle blockage by means of passing air through the headers.

312.25 (6.1.2)

Table 6.1-4 lists "other organic materials in containment." It is not clear whether these chemicals meet the requirements of Regulatory Guide 1.54, or if these are considered to be insignificant in quantity. Please clarify this, bearing in mind that we only consider any quantities less than about 100kg to be insignificant.

312.26 (15.6.3) For the letdown line rupture accident, you indicated that the secondary side pressure will reach safety relief valve set points. The valves may open, releasing secondary coolant activity to the atmosphere. If you have not included this contribution to the doses, provide an analysis including this source term.

312.27 (15.6.5) It is noted from Page 15.6-46 of the FSAR that the post-LOCA ESF leakage is estimated to be approximately 1.2 liters per hour and that the resultant doses at the site boundary without filters is a low fraction of the 10 CFR 100 value. Provide an analysis of a gross failure of a pump seal and indicate the radiological consequences. Describe the instrumentation available to the reactor operator which would indicate a gross seal failure of an ESF train. Describe the ventilation system used in the ESF areas following a loss of coolant accident. Indicate if ESF grade filters are used in the exhaust from the ESF areas.

312.28 (15.7.4) Provide the following information with respect to a postulated fuel handling accident:

a. Describe the radiation monitoring instrumentation which will detect a fuel handling accident inside the containment structure and in the spent fuel storage building. Provide drawings of the containment and spent fuel pool area exhaust systems which show the location of the radiation detectors relative to the exhaust inlets and isolation valves.

b. Describe the response time of the containment isolation valves. Indicate closure times which will be included in your technical specifications.

c. Indicate the transient time from the radiation monitor detection to the isolation valve based on the maximum velocity of the air in the exhaust system.

d. Provide a refueling accident analysis to show that the containment building isolation valves will close before any significant release of activity can occur. Provide a similar analysis to show that any activity released from the spent fuel pool area will be diverted through safety grade filters prior to its release to the environment.

321.0 EFFLUENT TREATMENT SYSTEMS BRANCH

321.1 (6.5.1; 9.4.3.1) Your description of the engineered safety feature filter systems is not adequate. Provide your justification for not including demisters in the engineered safety features filter system for the fuel handling building.

321.2 (10.4.8 SRP) Describe your provisions for instrumentation and control of the steam generator blowdown system to protect temperature sensitive elements, such as demineralizers, and to control flushing, liquid levels, and process flows through individual components.

321.3 (11.2) For gaseous and liquid radioactive waste processing systems, provide in tabular form a comparison between the components of your proposed systems and the appropriate equipment codes presented in Table 1 of Branch Technical Position ETSB 11-1 (Rev. 1), a copy of which is attached. Your tabulation referenced in Chapter 3.2 of the FSAR is not satisfactory in that it is not complete with respect to Materials, Welder Qualification and Procedure, and Inspection and Testing.

321.4 (11.2) Provide a table listing tanks outside reactor containment which contain potentially radioactive liquid materials. The table should include tanks located both inside and outside of plant buildings. For each tank, indicate the provisions incorporated to monitor tank levels, to annunciate potential overflow conditions, and to collect and process liquids in the event of overflows. Acceptable provisions include dikes around tanks, retention basins, and elevated thresholds to contain liquids in bays containing the tanks.

321.5 (11.3; 9.3.2) Your response to question 320.2 is not satisfactory. It is our position that at least one additional gas analyzer, which is continuously on stream at a point common to streams measured sequentially, should be added to your system. It is also our position that the gas analyzers should, upon high-high alarm, initiate automatic control features to reduce the potential for explosion; acceptable automatic control features which should be considered are automatic isolation of either the source of oxygen or hydrogen or the injection of diluents to reduce concentrations to limits outside of the explosive envelope.

321.6 (11.3.1.7) Your cost-benefit analysis, as provided in Appendix 5A of the Environmental Report and referenced in Section 11.3, does not present sufficient information to permit us to evaluate your results. You should provide the bases for your cost-benefit analysis in the form of Cost Estimate Sheets, as shown in Appendix B of Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," March 1976, for each augment considered in your analysis.

321.7 (10.4.3) Section 10.4.3.2.2, seventh paragraph, states: "A full discussion of the radiological aspects of primary-to-secondary system leakage, including anticipated releases from the turbine gland sealing system...is included in Chapter 11." We did not find the referenced discussion in Chapter 11. You should provide the referenced discussion.

321.8 (11.4) Your waste solidification system provides for mixing paddles to be used only when solidifying resins. Describe your procedures for verifying that adequate mixing can be achieved when solidifying materials other than resins without mechanical mixing in the solidification container and for assuring the absence of free water in the completed product for all types of materials to be solidified.

331.0 — RADIOLOGICAL ASSESSMENT BRANCH

331.8 (Table 12.2-4)

The units of activity in Table 12.2-4 should be disintegrations/Cm^3 - S instead of disintegrations/cm^2 - S.

331.9 (12.3)

Provide a detailed layout of the solid radwaste area showing: low and high radioactive waste storage areas; fill and cap area; radwaste control room; truck loadout area; and waste baler.

Indicate on the layout drawing the path of a waste liner from the time it is removed from the new liner storage area to the time it is placed in the high radioactive storage area.

331.10 (Figure 12.3-1)

Provide the justification for the zone V classification of the passageway located between the boric acid concentrator pac and the radwaste control panels in Figure 12.3-1. Describe how personnel access is regulated through the door connecting this passageway with the zone I area opposite the chemical storage room.

331.11 (Fig. 12.3-4 and Fig 12.3-8)

In Figure 12.3-4, the room containing the fuel handling building pumps and sumps (P-328 and P-329), the fuel pool purification pump (P-014), the spent fuel pool pumps (P-009 and P-010), and the leak detection sump is designated as a zone I area. The room containing the same equipment for unit 3 in Figure 12.3-8 has a zone III designation. Clarify this apparent discrepancy in zoning. If these rooms are zone III areas, justify your use of a zone III radiation designation considering the location of the six above listed components in the same room.

331.12 (Figure 12.3-23)

Figure 12.3-23 indicates that the local sample lab in the radwaste area of the auxiliary building will be designated a zone V area during reactor shutdown. Describe your plans to restrict personnel access to this lab during this period.

331.13 (Table 12.4-10)

The estimated tritium concentration in the containment four hours after purge initiation as given in Table 12.4-10 is 3.8 times greater than the estimated tritium concentration in the containment during normal operation (with no purge) as listed in Table 12.2-8. Explain why the tritium concentration after four hours of purging is greater than the normal tritium concentration.

331.14 (12.5.2)

Discuss provisions for laundering contaminated protective clothing and equipment.

331.15
(12.5.2.1)

Discuss provisions for local exhaust systems to be employed in the hot machine shop for work on contaminated items.

362.0 Geotechnical Engineering Section, Geology-Seismology Branch

362.1 (2.5.4) A presentation of the construction control data pertinent to in situ soil or structural fills and backfills is necessary for a complete review. The data should indicate type of tests, locations, elevations, gradations, density, percent compaction, etc.

362.2 (2.5.4.2 Correct discrepancies in the terrace deposit description between Section 2.5.4.2, paragraph 2.5.4.2.1.2, Terrace Deposits and Appendix 2.5A.

362.3 (2.5.4.2) Paragraph 2.5.4.2.2.4, Dynamic Stiffness and Damping. Clearly label Table 2.5-12 as terrace deposit soils. The percent axial strain, in column 12, ranges from 0.2 to 10.0 percent; define the failure criteria used and present test results pertinent to soil strength evaluation. Strength, damping and stiffness test results should be presented separately. Clearly indicate when double or single amplitude strain is being discussed or used in all figures and tables.

362.4 (2.5.4.2) Paragraph 2.5.4.2.2.5, Dynamic Strength. Expand Table 2.5-13 or provide a new table to present the results of tests on the more predominant San Mateo Formation samples.

362.5 (2.5.4.2) Paragraphs 2.5.4.2.2.4, Dynamic Stiffness and Damping and 2.5.4.2.2.5, Dynamic Strength. Discuss and justify the predominant use of $k_c = 1.0$ on test samples.

362.6 (2.5.4.2) Paragraph 2.5.4.2.2.5, Dynamic Strength. A more detailed discussion is required for Figures 2.5-34, 35, 36 and 37. Indicate percent strain used as failure in the strength curves shown and justify the use of remolded samples in obtaining the in situ strengths of the San Mateo Formation and terrace deposit soils. Discuss how the samples were remolded. Present the failure curve used in Figure 2.5-35. Discuss and justify the use of different C_f values in Figures 36 and 37. Present the dynamic strength curves of the fine grained San Mateo Formation on Figures 36 and 37 for comparison.

362.7
(2.5.4.5) Section 2.5.4.5, paragraph 2.5.4.5.1, Excavation Plan and Sections. Present the "as constructed" excavation plan and section on Figures 2.5-49 and 50. If these are "as constructed" details, indicate this on the figure.

362.8
(2.5.4.8) Section 2.5.4.8, Liquefaction Potential Evaluation. Rewrite this section to present a straight-forward approach to liquefaction potential evaluation. "Recent changes," in the state-of-the-art should be presented by showing the new strength curves along with those developed in the PSAR for comparative purposes, rather than by the use of a correction factor.

362.9
(2.5.4.10) Paragraph 2.5.4.10.4, Settlement and Heave. Results of the settlement analysis are to be confirmed with as-built data. Update this paragraph to present recent settlement or heave readings.

362.10
(Table 2.5.C.1)
(2.5.D.2) Provide two sets of the reports listed in Tables 2.5.C.1 and 2.5.D.2.

362.11
(2.5.4.5.2) Paragraph 2.5.4.5.3, Backfill. Reference is made to an incorrect paragraph 3.5.4.2; correct this paragraph to indicate correct section or paragraph.

362.12
(2.5.4) Correct Figure 2.5-70 soil parameters to be consistent with the main text.

362.13
(2.5.4) Paragraph 2.5.4.14.2, Support of Structures on Backfill. Clarify Figures 2.5-64 and 65 to indicate where the +5 feet elevation control was used.

421.0 QUALITY ASSURANCE BRANCH

421.1 Describe in the FSAR your QA program for fire protection in accordance with the information and guidance previously transmitted with the Boyd/Moore letters dated May 3, 1976 and September 30, 1976.

422.0 CONDUCT OF OPERATIONS (QUALITY ASSURANCE BRANCH)

422.1 (13.1.1) Provide the number of professional persons assigned to the Mechanical, Civil/Structural, Controls and Electrical, Apparatus and Materials, and Nuclear Engineering Sections of the Engineering and Construction Department.

422.2 (13.1.1) Provide the personal resume of the person assigned to the position of Supervisor of Plant Operation and Maintenance (superior of Superintendent of San Onofre Station).

422.3 (13.1.2.2) Describe the responsibilities and authority of the Operating Foreman and Fuel Handling Foreman shown in Figure 13.1-3.

422.4 (RSP) (13.1.3.1) You state in Section 13.1.3.3 that the Supervisor of Plant Operations should have a "total of five years experience ..." It is our position that the Supervisor of Plant Operations' position is comparable to that of Operations Manager described in Section 4.2.2 of ANSI N18.1-1971 and should have a "minimum of eight years of responsible power plant experience ..." State your intent to conform to this position.

422.5 (13.1.3.1) Your description of the minimum requirements for the position of Chemical and Radiation Protection Engineer are not clear in regard to the requirements for a "graduate in engineering or physical science" and the credit toward the number of years experience being fulfilled by related technical or academic training; or the specific experience requirements in chemistry and in radiation protection. Please clarify this item. Note that our position in regard to the minimum requirements for the individual in charge of radiation protection at the site is described in Revision 1 to Regulatory Guide 1.8, September 1975.

422.6 (RSP) (13.1.3.1) You state that the minimum requirements for the positions of Chemical Radiation Foreman, Instrument Foreman and Maintenance Foreman require a high school education or equivalent, two years of appropriate experience and one or two years of training. It is our position that the minimum requirements for these three positions should be as described in Section 4.3.2 of ANSI N18.1-1971; a high school diploma or equivalent and a minimum of four years experience in the craft or discipline he supervises. State your intent to conform to this position.

422.7 (RSP) (13.1.3.1)	It is not clear from your description in Section 13.1.3.1 whether or not technicians and repairmen will meet the requirements specified in ANSI N18.1-1971. It is our position that your technicians meet the minimum requirements described in Section 4.5.2 of ANSI N18.1-1971 and that your repairmen meet the minimum requirements described in Section 4.5.3 of ANSI N18.1-1971. State your intent to conform to these positions.
422.8 (13.1.3.1)	Describe your minimum qualification requirements for the position of Fuel Handling Foreman.

423.0 INITIAL TEST PROGRAM (QUALITY ASSURANCE BRANCH)

423.1 (14.2)

(1) Regulatory Guide 1.68 states that the objectives of the initial test program are to provide assurance that (a) the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, (b) the plant procedures have been evaluated and demonstrated, and (c) the operating organization is knowledgeable about the plant and procedures and is prepared to operate the facility in a safe manner. The objectives of your test program as stated in Section 14.2.1 do not include Item 3. Expand this section to include this objective.

(2) Section 14.2.2 does not state how the station engineering staff (nuclear engineers and assistants, plant engineer, supervising nuclear engineer, and engineering aides) will be utilized during the initial test program. Expand this section to include this information.

423.2 (14.2.2)

For the staff to complete its review of the organization and staffing of the test program, the following additional or clarifying information will be required:

(1) The responsibilities of the Test Operations Supervisor and the Technical Supervisor of the SCE startup organization.

(2) The minimum qualifications requirements (educational, experience , and nuclear experience) for the following categories of personnel at the time they are assigned to the task. Your response should address all personnel performing the tasks listed and should not be limited to only SCE personnel (e.g., Test Working Group members and augmenting personnel). Note that ANSI N45.2.6, although applicable to some categories of personnel during the construction, preoperational, and startup phases, was not intended to cover personnel in the listed categories.

(a) Personnel that prepare individual preoperational test procedures.

(b) Personnel that supervise or direct the conduct of individual preoperational tests.

(c) Personnel that review and/or approve preoperational test procedures.

(d) Personnel that approve preoperational test results.

(e) Personnel that prepare individual startup test procedures.

(f) Personnel that supervise or direct the conduct of individual startup tests.

(g) Personnel that review and/or approve startup test procedure

(h) Personnel that approve startup test results.

423.3 (14.2.3) Section 14.2.3 implies that the acceptance review of test procedures will be performed by personnel other than the individual Test Working Group members. Either modify this section to require minimum qualifications for these reviewers that are commensurate with this responsibility or clarify the information presented to indicate that a technical review of test procedures will be performed by the individuals who are members of the TWG.

423.4 (14.2.3) Expand Section 14.2.3 to state which staff positions (functional title will write test procedures and review these procedures before they are submitted to the TWG.

423.5 (14.2.4) Section 14.2.4 describes the methods for changing test procedures after they have been approved by the TWG. Define scope/intent and nonscope/intent changes as used in this section.

423.6 (14.2.7) Revise Section 14.2.7 to state where in the test program that the operability of the safety injection tank discharge isolation valves is demonstrated using the emergency power source.

423.7 (14.2.11) For the staff to complete its safety evaluation of your test program schedule, it is necessary that your application provide a sequence of performing tests during the power ascension phase. This sequential schedule should establish that the commitments of Section 14.2.11 will be met. Modify this section to provide an accurate sequence of conducting tests or present this information in a table or figure.

423.8 (14.2.12) The staff's review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68, Appendix A will not be demonstrated by the initial test program (although Section 14.2.7 does not list these items as exceptions). Expand your FSAR to include appropriate test descriptions for the following items from Appendix A of the guide:

A.1.c	vibration monitoring of reactor internals
A.5.d	vent and drain systems
A.5.o	shield cooling system (include reactor cavity cooling system)
A.5.p	leak detection system
A.5.r	seismic instrumentation
A.6.a	normal distribution system
A.7.b	reactor building tests (include normal containment ventilation system)
A.8	gaseous radioactivity removal systems
A.9.a	ECCS expansion and restraint tests
A.10.b	refueling equipment (hand tools and power equipment)
A.11	reactor components handling system (include containment polar crane)
A.12.b	personnel monitor and survey instruments
A.12.c	laboratory equipment
A.13	radioactive waste systems
B.1.d	post fuel loading RCS leak test
B.1.j	vibration monitoring of reactor internals
C.1.i	chemical tests to demonstrate ability to analyze and control water quality
D.1.f	effluent radiation monitoring systems
D.1.n	capability of instrumentation to detect a dropped CEA and initiate associated automatic actions
D.1.p	vibration monitoring of reactor internals

423.9 (12.2.12) Our review of the listing of test description disclosed that several plant features which are assumed in Chapter 15 to limit or mitigate the results of postulated accidents may not be tested preoperationally. Provide test descriptions (or reference or revise existing test descriptions) which verify the operability of the following:

(1) atmospheric steam dump valves

(2) reactivity computer

(3) de-energization of pressurizer heaters on low level

(4) containment sump and associated instrumentation and alarms

(5) radwaste area sump and associated instrumentation and alarms

423.10 (14.2.12) The staff's review of your individual test descriptions disclosed that the information provided in several of these descriptions is not sufficient for the staff to conclude that adequate testing will be performed on the systems and components covered. Expand and/or modify the test abstracts to provide the following:

(1) Test No. 1: Modify the description to provide assurance that the test will verify that emergency loads will not exceed battery sizing assumptions and demonstrate that supplied loads will operate in accordance with design requirements at minimum design voltage level (at the battery terminals) for the system.

(2) Test No. 8: Modify the test description to provide assurance that the test will verify components during the low pressure operation on the nitrogen supply system.

(3) Test No. 14: Modify the description to provide assurance that the test will verify the capability of the turbine driven feedwater pump to start and operate under the full design range of steam pressures (1210-65 psia).

(4) Test No. 20: The acceptance criteria for this test should include conformance with the plant's technical specifications limits.

(5) Test No. 22: Verify that this test or other tests, as appropriate includes the station effluent radiation monitors.

(6) Test No. 32: The acceptance criteria for this test should be expanded to include conformance with the limits that will be included in the technical specifications for the facility.

(7) Test No. 45: Modify the description to provide assurance that the test will verify that injection valves are set to prevent pump runout.

(8) Test No. 52: State whether this test is performed with steam.

(9) Test No. 54: Modify the description to provide assurance that the test will demonstrate that low pressurizer level de-energizes the pressurizer heaters.

(10) Test No. 56: Modify the description to provide assurance that the test will demonstrate, by sample analysis, the capability to purge the quench tank and establish a nitrogen blanket.

(11) Test No. 58: State how corrections will be made to account for setting pressurizer safety valve lift-points with nitrogen at ambient temperature rather than with steam at normal operating temperature. Provide supporting technical justification for the correlations.

(12) Test No. 67: Modify the description to provide assurance that the test will verify proper electrical operation of the moveable incore detector system.

(13) Test No. 75: Modify the description to provide assurance that the test will verify that the travel time of each CEA in the dashpot region satisfies mechanical design requirements and satisfies reactivity assumptions.

(14) Test Nos.76,77: Modify the description to provide assurance that the test will demonstrate that sampling procedures are adequate.

(15) Test Nos. 82, 83, 84, 85, 86, 87, & 89: For each test, state quantitative acceptance criteria and the basis for the criteria.

(16) Test No. 90: State what transients will be performed or analyzed as a part of this test, at what power level each of the transients will be conducted, the mode of operation (automatic or manual) of control systems, and provide acceptance criteria based on the predicted results of each transient.

(17) Test No. 91: State what transients and trips will be performed or analyzed to determine proper operation of control systems; state which parameters are monitored during normal operations, trips, and transients that are used to make this determination; state the mode of operation of control systems for each transient and trip; and clarify "acceptable range" of the monitored parameters in the acceptance criteria.

(18) Test Nos. 93, 94: Identify the variables or parameters to be monitored for each test; provide assurance that the test results will be compared with predicted results for the actual tests to be run (for each trip); establish quantitative acceptance criteria and the basis for the required degree of convergence of actual test results with predicted results for the monitored variables or parameters for each trip, and establish acceptance criteria for grid stability, voltage, and frequency following the generator load rejection trip.

(19) Test No. 95: Modify the description to provide assurance that the test will verify that the reactor scram will be initiated from outside the control room and that offsite power will be available.

(20) Test Nos. 99, 100: For each procedure state quantitative acceptance criteria and the bases for the criteria.

423.11 (14.2) Provide a description of testing planned to demonstrate (1) the operability and adequacy of the CEDM cooling units and the ventilation systems for the auxiliary feedwater pump rooms; (2) the capability of the reactor cavity cooling system to maintain concrete temperature below 150° F; and (3) that the insulation provided for the high temperature containment penetrations prevents excessive heating of the concrete surrounding the penetrations.

423.12 (14.2) Tests of normal and emergency ventilation systems for areas housing engineered safety features components should demonstrate that the temperature of each compartment can be maintained below the design temperature limits of the components within the compartment with the maximum expected heat load being produced in the compartment. Modify the necessary test descriptions to include this demonstration.

423.13 (14.2) Provide a description of the testing planned to demonstrate that the response time of each protection channel and ESF channel from the measured variable to the final actuating device is within the assumptions used in the accident analysis.

423.14 (14.2) Identify any of the post-fuel loading tests described in Section 14.2.12 which are not essential towards the demonstration of conformance with design requirements for structures, systems, components, and design features that:

(a) will be relied upon for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period;

(b) will be relied upon for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions;

(c) will be relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications;

(d) are classified as engineered safety features or will be relied upon to support or assure the operations of engineered safety features within design limits;

(e) are assumed to function or for which credit is taken in the accident analysis for the facility (as described in the Final Safety Analysis Report); and

(f) will be utilized to process, store, control, or limit the release of radioactive materials

423.15 (14.2) Acceptable justification is not presented for performing the turbine trip test at 80% power instead of at 100% power as stated in Regulatory Guide 1.68. It is also not clear that the generator trip test meets the intent of the guide. The reasons for including the generator trip test in Regulatory Guide 1.68 were to assure that the turbine generator would not exceed its design speed and to establish that the plant's electrical system would perform as designed for this transient test during which the system may be subjected to frequencies in excess of 60 Hz. To accomplish the test objectives, the generator should be disconnected from the transmission system in a manner that will result in the calculated maximum overspeed condition. Normally, this is accomplished by opening of the generator output breaker in a manner that will require a turbine generator overspeed condition to initiate closure of the steam admission or stop valves.

It is our understanding that typical designs of the trip logic for the generator output breakers will, for certain sensed plant conditions, result in a direct and simultaneous trip of the turbine stop valves. There usually are additional trips that will also open the generator output breakers without directly tripping the turbine stop valves. Therefore, the latter type of trip should be simulated to initiate the transient.

Modify Section 14.2.7 and the test descriptions as necessary to clarify that the generator trip test will be performed as intended by Regulatory Guide 1.68 and to either state that the turbine trip test will be performed at 100% power or provide technical justification for conducting the test at a different power level.

423.16 Provide more detailed test descriptions for testing the core protection calculator, CEA calculator, and core operating limits supervisory system both preoperationally and following fuel loading.

423.17 (14.2.12) Provide a test description for vibration monitoring performed at 20, 50, 80, and 100% power as listed in Table 14.2-2.

423.18 (14.2.7) Provide additional justification for omitting or reducing in scope low power physics tests and power ascension tests on Unit No. 3. Also, for the tests that will be reduced in scope, describe what will be modified or omitted from these tests.

423.19 (15.2) Provide a description of the testing planned to demonstrate the operability of any structures, design features, or systems to protect the facility from external and internal flooding (include leak tightness tests of compartments, doors, and waterproof hatches of safety equipment areas).

432.0 EMERGENCY PLANNING BRANCH

432.1 Your FSAR submittal did not include the emergency response plans of offsite agencies as indicated in Appendix F of your emergency plan. Please provide copies of these plans or a schedule for their submittal.

432.2 It appears that a void in the communications with offsite agencies may exist in the potentially critical time period between initial notification via the SCE System Dispatcher and the establishment of the SCE Radiological Emergency Team I for continuing liaison with these agencies. Since Emergency Team I may not be available for two hours (Section 5.3), during which time emergency conditions may change, discuss what provisions exist for communicating significant developments from the Emergency Coordinator to the local supporting agencies.

432.3 In order to ensure timely activation of the San Diego County Office of Emergency Services it appears that a more positive notification be provided other than that indicated in Table 5.4 as a "professional answering service." For example, we would consider that a call list of the San Diego County OES staff personnel could be used for providing reasonable assurance of initiating the necessary emergency support activities.

Revise your plan accordingly or furnish justification that your present notification scheme provides such assurance.

432.4 Emergency support activities provided by certain Federal agencies appear to have been omitted from your plan although such services, in part, are suggested in Table 6.6, item 4.b, and in Section 9.2. Amend your plan to more explicitly identify the provisions for utilizing the supporting services from the U.S. Coast Guard and the ERDA Radiological Assistance Program.

432.5 As set forth in Section 6.1 of Annex A to Regulatory Guide 1.101, Revision 1 (March 1977), amend your plan to acknowledge the existence of a "message authentication scheme" with each of the supporting service organizations.

432.6 As set forth in Section 6.4 of Annex A to Regulatory Guide 1.101, Revision 1 (March 1977), describe the provisions to make available on request to occupants in the low population zone, information concerning how the emergency plans provide for notification to them and how they can expect to be advised what to do.

432.7 Item 4 in Table 6.6 of your plan indicates that control of public access via highways, water, and railroad to potentially affected portions of the exclusion area is the responsibility of the California State Highway Patrol, the U.S. Coast Guard, and the Atchison, Topeka, and Santa Fe Railroad. Because of the proximity of these access points it would appear desirable in the event of a serious accident that the response of the aforementioned organizations be accomplished in a timely fashion. However, it is not clear that your plan provides for prompt notification to

432.7 Cont- those directly responsible, nor does the plan describe the arrangements made for their response to emergencies. Please identify the provisions for such notification and, if appropriate, amend the applicable portions of your plan (e.g., Table 5.3, Figure 6.1) including letters of agreement which document the arrangements and expected response measures to be provided.

432.8 As set forth in Section 7.5 of Annex A to Regulatory Guide 1.101, Revision 1 (March 1977), provide a summary description of onsite damage control equipment and supplies.

432.9 As set forth in Section 8.1.2 of Annex A to Regulatory Guide 1.101, Revision 1 (March 1977), amend your plan to provide for quarterly drills for fire brigade members, annual fire drills including participation by an offsite fire department, and annual drills of repair and damage control teams.

432.10 Amend Section 8.2 of your plan to provide the following:

(1) Annual review of emergency plan and procedures.

(2) Dissemination of plan revisions and/or applicable procedure changes to emergency support organizations.

(3) Review and update of all written agreements at least every two years.

432.11 The last sentence in Section 9.2 of your plan appears to imply some kind of shared jurisdiction between the State Radiologic Health Section and the U.S. Energy Research and Development Administration with respect to decontamination of offsite areas.

432.11 Cont. We do not understand the meaning of this statement and consider it potentially misleading. Please clarify and reword to more clearly express what is intended.

432.12 Include as an appendix to your plan the information requested in item 3 of Section 10, Annex A to Regulatory Guide 1.101, Revision 1, (March 1977).

440.0 OPERATOR LICENSING BRANCH

441.1 (13.2.2) Include in Section 13.2.2.1.4 the commitment that: "Only two licensed individuals will be exempt from taking the annual exam, provided these individuals are directly involved in the preparation and grading of the exam.

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RMattu
RBosnak
VAMoore

Docket Nos. 50-361
and 50-362

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Gentlemen:

SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE

By your affidavit and application dated March 9, 1977 and March 30, 1977, respectively, you submitted the following documents and requested that they be withheld from public disclosure pursuant to 10 CFR 2.790:

1. Drawings of the San Onofre Units 2 and 3 Reactor Coolant System, Internals and Supports listed in Enclosure 1.

2. "Structural Properties of the CE 16 x 16 Fuel Assembly and Reactor Internals."

This information was requested by the NRC via reference (1) and transmitted to us at the February 10, 1977 meeting at Windsor, Connecticut, in which NRC, its subcontractor, the Idaho National Engineering Laboratory, Southern California Edison and Combustion Engineering participated. The information transmitted is required by the NRC for the North Anna Audit Analysis.

We have reviewed your application and the referenced materials based on the requirements of 10 CFR 2.790 and have determined that the documents sought to be withheld contain trade secrets or confidential or privileged commercial information.

We have also found at this time that the right of the public to be fully apprised as to the bases for and effects of the proposed licensing action does not outweigh the demonstrated concerns for protection of your competitive position. Accordingly, we have determined that the information should be withheld from public disclosure.

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We therefore approve your request for withholding pursuant to 10 CFR 2.790 and are withholding the materials referenced above from public inspection as proprietary.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents. We have furnished copies of this information to our consultant, the Idaho National Engineering Laboratory, who has signed the appropriate agreements for handling proprietary data.

Sincerely,
Original Signed by

Karl Kniel, Chief
Light Water Reactors
Branch No. 2
Division of Project Management

cc: See page 3

References:
(1) NRC letter, Karl Kniel to J. B. Moore, December 17, 1976

(2) Meeting summary, NRC/INEL/SCE/CE meeting at Windsor, Connecticut, February 10, 1977

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SURNAME➤	MMlynczak:mt	HRood	JMaynard	RMattu/RBosnak	KKniel	LChandler
DATE➤	4/ /77	4/ /77	4/ /77	4/ /77 4/ /77	4/ /77	4/ /77

APR 1 8 1977

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San Onofre Units 2 and 3

Drawings Presented to the NRC for North Anna Audit Analysis

	Drawing No.	Rev.	Component
1.	E-234-590	4	Steam Generator
2.	E-234-592	9	Steam Generator
3.	E-234-586	3	R.V. Column
4.	E-234-550	3	Reactor Vessel
5.	E-234-551	2	Reactor Vessel
6.	E-234-587	4	R.V. Column
7.	E-235-176	2	Pipe
8.	2F-1565, Sheet 1	D	Reactor Coolant Pump
9.	2F-1565, Sheet 2	D	Reactor Coolant Pump
10.	E-1370-320-007	04	S.G. Supports
11.	E-1370-320-022	04	S.G. Supports
12.	E-1370-320-035	01	R.V. Supports
13.	D-1370-320-052	03	RC Pump Skirt
14.	D-1370-320-053	03	RC Pump Supports
15.	D-1370-320-055, Sheet 1	05	RC Pump Supports
16.	D-1370-320-056	05	RC Pump Supports
17.	D-1370-320-054	04	RC Pump Supports
18.	D-1370-320-055, Sheet 2	05	RC Pump Supports
19.	D-1370-320-057	01	RC Pump Supports
20.	E-1370-320-058, Sheet 1	01	RC Pump Installation
21.	E-1370-320-058, Sheet 2	01	RC Pump Installation
22.	E-1370-320-059	00	RC Pump Installation
23.	E-234-981	7	Pressurizer
24.	E-234-982	6	Pressurizer
25.	C-E Calculation 3400-16-40-30	6/24/74	Internals Weights
26.	E-1370-164-311	02	Core Support Barrel
27.	E-STD-164-313	05	Core Support Barrel
28.	E-STD-164-314	04	Core Support Barrel
29.	E-STD-164-315	04	Core Support Barrel
30.	E-STD-164-316	06	Core Support Barrel

	Drawing No.	Rev.	Component
31.	E-1370-164-303	02	Reactor Internals
32.	E-STD-164-326, Sheet 1	04	Core Shroud
33.	E-STD-164-326, Sheet 2	04	Core Shroud
34.	E-STD-164-326, Sheet 3	04	Core Shroud
35.	E-STD-164-312, Sheet 1	07	Core Plate
36.	E-STD-164-312, Sheet 2	07	Core Plate
37.	E-STD-164-332, Sheet 1	08	Upper Guide Structure
38.	E-STD-164-332, Sheet 2	08	Upper Guide Structure
39.	E-STD-164-335	03	CEA Shroud
40.	E-STD-164-333, Sheet 1	06	Alignment Plate
41.	E-STD-164-333, Sheet 2	06	Alignment Plate
42.	E-3072-164-331, Sheet 1	01	Upper Guide Structure
43.	E-3072-164-331, Sheet 2	01	Upper Guide Structure
44.	E-STD-161-017	03	Fuel
45.	E-STD-164-001, Sheet 1	01	Internals
46.	E-STD-164-001, Sheet 2	01	Internals
47.	E-1370-320-036	01	RV Supports
48.	E-235-177	2	Pipe
49.	E-235-178	3	Pipe
50.	E-235-179	3	Pipe
51.	E-235-180	3	Pipe
52.	E-1370-320-001, Sheet 1	08	RCS Load Table
53.	E-1370-320-021	03	Steam Generator Support
54.	E-STD-220-031	05	Steam Generator Support
55.	E-STD-220-032	06	Steam Generator Support

FEB 25 1977

Docket Nos: 50-361
and ~~50-~~362

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Gentlemen:

RE: SAN ONOFRE 2 AND 3

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate a power reactor to advise you that the Nuclear Regulatory Commission has forwarded to the FEDERAL REGISTER, amendments to its regulations 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and 10 CFR Part 73, "Physical Protection of Plants and Materials." These new regulations identify measures to be taken for the protection of nuclear power reactors against industrial sabotage. Copies of these new requirements are enclosed. Of particular interest is the adoption of a new section 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against industrial sabotage." The new regulations require that you submit an amended physical security plan within 90 days of the publication of the rule in the FEDERAL REGISTER describing how you plan to comply with the requirements of 10 CFR 73.55, including schedules of implementation.

To provide additional detailed guidance on implementing the new rule, we are scheduling regional meetings to discuss the requirements of 10 CFR 73.55, to present an acceptable format and content for the required amended physical security plan and to provide preliminary acceptance criteria which the NRC staff will use to determine the acceptability of submittals. An agenda for these meetings is enclosed, including the dates and location of the meeting for each NRC Region and supplemental information related to some of the topics listed on the agenda. In order to provide a forum for effective discussion, you are requested to send no more than four representatives to the meeting. You may wish to include your A/E or security consultant within this number. Please complete the enclosed Registration Form and return it in the envelope provided.

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DATE➤						

The Commission has under active consideration a requirement that security clearances be obtained for certain licensee employees. We will present an overview of this proposal at the meeting and will consider any comments that you wish to give.

The Commission also has under development, amendments to its regulations that would require nuclear power plant licensees to develop and follow safeguards contingency plans for dealing with threats, thefts and sabotage relating to special nuclear material and nuclear facilities. A presentation and discussion on this subject is on the meeting agenda and background information on this subject is also enclosed.

If you have any particular related topics or generic safeguards problems that you would like discussed at the meeting, please let us know. For any further information or comments, please contact James R. Miller of my staff (301/492-7014).

Sincerely,

Original Signed by

Ben C. Rusche, Director
Office of Nuclear Reactor
Regulation

Enclosures:
1. Copy of Amended Regulations
2. Meeting Agenda
3. Registration Form and Return Envelope
4. Draft Standard Format and Content Document
5. Contingency Planning Information

cc: w/enclosures 1, 4 and 5 (see next page)

No concurrences required; approved for issuance by Mr. Rusche per C. Van Niel.
H. Smith 2/17/77

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SURNAME➤						
DATE➤						

FEB 25 1977

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Docket Nos. 50-361
and ~~50~~-362

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Gentlemen:

SUBJECT: ACCEPTANCE REVIEW OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

On December 1, 1976, you tendered an application for operating licenses for the San Onofre Nuclear Generating Station, Units 2 and 3. Your application included the Final Safety Analysis Report, General Information, the Environmental Report and Antitrust Information.

We have completed our review of your tendered application and have concluded that it is acceptable for docketing based on your submittal of outstanding information consistent with the schedule detailed in your letter of February 18, 1977.

Your application should be provided to us as soon as possible. Your filing of the application and any amendments thereto should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of that portion of the application containing the general information and forty (40) copies of the safety analysis report. As required by 10 CFR 50.30, you should retain an additional ten (10) copies of the general information and thirty (30) copies of the safety analysis report for direct distribution in accordance with the Enclosure 1 to this letter and further instructions which might be provided later. In

OFFICE ➤						
SURNAME ➤						
DATE ➤						

addition, forty one (41) copies of the Environmental Report must be submitted and an additional one hundred nine (109) copies retained for direct distribution in accordance with Enclosure 1. Within ten days after docketing, you must provide an affidavit that distribution in accordance with the enclosure has been completed. These requirements also apply to all subsequent amendments to your application.

You will be advised of key milestones of the review as soon as the schedule is developed. During the course of our preliminary review of your Final Safety Analysis Report and Environmental Report, Enclosures 2 & 3, Requests for Additional Information, were generated. These requests require an early response for our mutual benefit during the ensuing detailed technical review period. We will prepare the review schedule based on the assumption that your responses to all of our acceptance review questions will be received within five weeks of the docketing date. If this milestone cannot be met, it may be necessary for us to revise your review schedule.

If, during the course of our review, you believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that the appropriate meeting can be arranged on a timely basis. A written request is not necessary and all such requests should be initiated through our staff project manager assigned to the review of your application. This procedure is an informal one, designed to allow opportunity for applicants to discuss with management areas of disagreement in the case review.

Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Service List for Direct Distribution
2. Request for Additional Information Resulting from the Acceptance Review of the San Onofre Nuclear Generating Station Units 2 & 3 FSAR
3. Request for Additional Information Resulting from the Acceptance Review of the San Onofre Nuclear Generating Station Units 2 & 3 ER

cc:	See page 3					
OFFICE ➤		DPM:LWR #2	DPM:LWR #2	DPM:LWR #2	DPM:AD/LWR	DPM
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ENCLOSURE 1

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Enclosure 2

Request for Additional Information Resulting from Acceptance Review of San Onofre Nuclear Generating Station Units 2 and 3, Final Safety Analysis Report

This request for additional information was developed during the acceptance review of the San Onofre Nuclear Generating Station Units 2 and 3 Final Safety Analysis Report. The requests are numbered such that the three digits to the left of the decimal identify the technical review branch and the number to the right of the decimal is the sequential request number. The number in parentheses indicates the relevant section of the Final Safety Analysis Report. Branch technical positions referenced in these requests can be found in NUREG-75/087, "Standard Review Plans for the Review of Safety Analysis Reports for Nuclear Power Plants," dated September 1975.

010.0 AUXILIARY AND POWER CONVERSION SYSTEMS BRANCH

010.1 (1.1) Some symbols used on the P&IDs are not defined in Section 1.1 of the FSAR. Revise Figures 1.1-2 through 1.1-5 to include definitions of all the symbols used.

010.2 (3.4) Provide the results of an analysis to demonstrate that the safety-related systems are protected from flooding due to ground water seepage through seismic Category I building walls.

010.3 (3.5) Provide a tabulation of all safety-related components which are located outdoors and describe the protection to be afforded these components to prevent their being damaged by tornado generated missiles or a seismic event. Include in this tabulation all safety-related HVAC system air intakes and exhausts, and the diesel generator combustion air intake and exhaust. Identify the locations of these components, air intakes and exhausts on the plant arrangement drawings.

010.4 (RSP) (3.6) San Onofre Units No. 2 & 3 must meet the guidelines set forth in the Branch Technical Position APCSB 3-1 or follow the guidance provided in the December, 1972 letter from A. Giambusso given in Appendix B to the Branch Technical Position APCSB 3-1. In addition, an analysis made in conformance with B.3 of this position must be presented to demonstrate that acceptable protection against the effects of piping failures outside containment has been provided.

010.5 (3.6) Provide layout drawings of the safety-related areas outside containment showing the high energy piping systems and their relation to the safety-related equipment. Indicate the method of protection from a high energy piping system failure for each system listed in Table 3.6-2; also provide a table listing moderate energy piping systems and their method of protection per the analyses performed in request 010.4.

010.6 (3.6) As an example of your analysis of the effects on the safety-related system from the high and moderate energy piping failure, provide the details of your evaluation on the shutdown cooling system (SCS) to demonstrate that adjacent safety-related systems and components are protected from the consequences of SCS piping failure, and the SCS is protected from the effects of failure of adjacent high and moderate energy systems.

010.7 (9.0) Identify the seismic design and quality group classification of the piping systems and the points of change of classification in the system for all P&IDs.

010.8 (9.0) With regard to potential failures or malfunctions caused by freezing, icing, and other adverse environmental conditions, discuss the protective measures that are provided to assure the proper function of those components not housed within temperature controlled areas, and that are essential in attaining and maintaining a safe reactor shutdown.

010.9 (RSP) (9.1.3) Discuss the interfaces between the shutdown cooling heat exchangers and the spent fuel pool cooling systems. It is our position that the shutdown cooling system not be used to back up the spent fuel pool cooling system unless the reactor core is unloaded.

010.10 (9.1.4) (RSP) Provide results of an analysis which demonstrates that a postulated cask drop will not cause damage of any safety-related system or component which may be located under the travel path of the cask handling crane. Otherwise, the crane must be designed to meet the guidelines set forth in Branch Technical Position APCSB 9-1.

010.11 (9.2.1) Provide detailed drawings to show physical arrangement of the salt water system components inside the intake structure and the traveling water screens and screen wash equipment at the intake structure. Provide the seismic Category classification of the traveling water screen systems and discuss the consequences of a failure of the traveling water screen system in light of the service water system operability.

010.12 (RSP) (9.2.2) Section 9.2.2 of the FSAR states that the spent fuel pool heat exchangers are served by the non-seismic Category I portion of the component cooling water system. It is our position that the piping of the spent fuel pool cooling system and the portion of the piping supplying component cooling water to the spent fuel cooling heat exchangers must be analyzed for SSE loading. Seismic Category I supports to these piping and the components must be provided.

010.13 (RSP) (9.2.2) The design of the component cooling water system provides a single line supplying cooling water to the four reactor coolant pumps, and a single return line for all pumps. These lines are not designed to seismic Category I requirements and contain motor-operated valves for containment isolation. The seals and bearings of the reactor coolant pumps require continuous cooling by the component cooling water system during all modes of operation. Inadvertent closure of any one of the above motor-operated valves would terminate the coolant flow to all of the pumps. This may lead to fuel damage, due to a locked rotor. Therefore, it is our position that this portion of the component cooling water system must be designed so that the following criteria are met:

010.13 (RSP) (Cont'd)

(a) A single failure in the component cooling water system shall not result in fuel damage or damage to the reactor coolant system pressure boundary caused by an extended loss of cooling to the reactor coolant pumps. Single failure includes operator error, spurious actuation of motor-operated valves, and loss of component cooling water pumps.

(b) A leakage crack in a moderate energy system or an accident that is initiated by a failure in the component cooling water system piping shall not result in fuel damage or a breach of the reactor coolant system pressure boundary when an extended loss of cooling to the reactor coolant pumps occurs. A single active failure shall be considered when evaluating the consequences of the accident. Moderate energy system leakage cracks should be determined in accordance with the guidelines of Branch Technical Position APCSB 3-1, "Protection Aginst Postulated Failures in Fluid Systems Outside Containment."

To meet the two criteria above, that portion of the component cooling water system which supplies cooling water to the reactor coolant pumps can be designed to non-seismic Category I requirements and Quality Group D if it can be demonstrated that the reactor coolant pumps are capable of operating with loss of cooling for longer than 30 minutes without loss of function and without the need for operator protective action. Also, in this case, safety grade instrumentation to detect the loss of component cooling water to the reactor coolant pumps and to alarm the operator in the control room must be provided. The entire instrumentation system, including audible and visible status indicators for loss of component cooling water must meet the requirements of IEEE Std 279-1971. Alternatively, if it cannot be demonstrated that the reactor coolant pumps will operate longer than 30 minutes without loss of function or operator corrective action, the design must meet one of the following requirements for the entire component cooling water system:

(c) Safety grade instrumentation consistent with the criteria for the protection system shall be provided to initiate automatic protection of the plant. In this case, the component cooling water supply to the seal and bearing of the pumps may be designed to non-seismic Category I requirements and Quality Group D; or

010.13 (RSP) (Cont'd)

(d) The component cooling water supply to the pumps shall be capable of withstanding a single active failure or a moderate energy line crack as defined in our Branch Technical Position APCSB 3-1 and shall be designed to seismic Category I, Quality Group C and ASME Section III, Class 3 requirements.

010.14 (RSP) (9.3.1)

Section 9.3.1 indicates that the compressed air system is not designed to seismic Category I requirements and the atmospheric steam dump valves are designed to fail-shut in the event of loss of air supply. It is our position that the atmospheric steam dump valves must be able to be operated from the control room for cold shutdown of the plant, and a seismic Category I air supply to the steam dump valves must be provided.

010.15 (RSP) (9.5.1)

The description of the fire protection system provided in the FSAR does not provide all of the information requested by Regulatory Guide 1.70.4. In addition, the design should meet the guidelines of Appendix A to Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." Revise the design as necessary to meet these guidelines.

010.16 (9.5)

Provide a detailed description, safety evaluation, and P&IDs of the diesel generator supporting system (i.e., diesel generator fuel oil storage and transfer system, starting system, diesel generator lubrication system and diesel generator combustion air intake and exhaust system).

010.17 (10.4.5)

Provide an evaluation regarding the effects of possible circulating water system failure inside the turbine building. Include the following:

(a) The maximum flow rate through a completely failed expansion joint.

(b) The potential for and the means provided to detect a failure in the circulating water transport system barrier such as the rubber expansion joints. Include the design and operating pressures of the various portions of the transport system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (rapid valve closure).

010.17 (Cont'd) (c) The time required to stop the circulating water flow (time zero being the instant of failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and flow coastdown time.

(d) For each postulated failure in the circulating water transport system barrier give the rate of rise of water in the associated spaces and maximum height of the water when the circulating water flow has not been stopped or overflows to site grade.

(e) For each flooded space provide a discussion, with the aid of drawings, of the protective barrier provided for all essential systems that could be affected as a result of flooding. Include a discussion of the consideration given to passageways, pipe chases, and/or the cableways joining the flooded space to the spaces containing safety-related system components. Discuss the effect of the flood on all submerged essential electrical systems and components.

010.18 (10.4.7) Events such as damage to the feedwater system piping at Indian Point Unit No. 2 on November 13, 1973, and at other plants, could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater inlet nozzles. Subsequent events in turn lead to the generation of a pressure wave (water hammer) that is propagated through the pipes and could result in damage.

(a) Describe normal operating transients that could cause the water level in the steam generator to drop below the sparger or nozzles and allow steam to enter the sparger and feedwater piping.

(b) Describe the routing criteria or show by isometric diagrams the routing of the feedwater piping from the from the steam generators through the containment to the outer isolation valve and restraint.

(c) Describe any analysis of the piping system, including any forcing functions, that will be performed or give the results of test programs to verify that uncovering of feedwater lines could not occur, or that if it did occur damage such as that experience at the Indian Point Unit No. 2 facility would not result at this facility.

010.19
(RSP)
(10.4.9)

It is our position that the power sources for all controls, valve operators and other supporting systems (e.g., pump lube oil cooling system) associated with the turbine driven auxiliary feedwater pump be independent from A/C power. This is necessary to comply with the diversity requirement in Branch Technical Position APCSB 10-1. Modify the system design to comply with this position and confirm that the turbine driven pump lube oil cooler will receive cooling water from the pump recirculation line.

020.0 CONTAINMENT SYSTEMS BRANCH

022.1 (Table 6.2-9) Expand Table 6.2-9 to include the following information:

(a) The energy removed by sprays, fan coolers and structural heat sinks up to the time of peak pressure.

(b) The actuation time of the sprays and fan coolers.

(c) The calculated containment pressure at 24 hours following the accident.

022.2 (6.2.1) For the postulated pipe break accident which results in the maximum containment pressure for a LOCA, provide the following information at the beginning of the accident, at the end of the blowdown phase, at the end of the core reflood phase, at the time of the peak containment pressure and at the end of one day:

(a) The energy in the containment atmosphere vapor and air and the energy absorbed by the containment structures;

(b) The energy removed by the containment spray water;

(c) Graphically show the energy distribution in the containment vapor region, sump region, structures, and the total energy, as a function of time.

022.3 (6.2.1) Provide and justify the analyses which demonstrate that the maximum safety injection assumption for LOCA containment pressure analysis yields the worst case containment pressure transient. Include the results of analyses assuming minimum safety injection for the spectrum of reactor coolant system pipe ruptures. Specify the maximum calculated pressure and the containment pressure at 24 hours.

022.4 (6.2.2) Provide the actuation signal(s) and setpoint(s) for the initiation of the containment spray system. Justify the adequacy of the delay time assumed in the containment analysis, including the effect of instrument error and dead band and the delays in instrument response, equipment startup, valve operations, and time to fill piping systems.

022.5 (6.2.1) For the containment response analysis to inadvertent containment spray system actuation, provide a curve of containment atmosphere temperature versus time. Discuss the provisions which will preclude simultaneous operation of other active containment heat removal equipment.

022.6 (6.2.1) With regard to the main steam line break accident analysis, provide the following additional information:

(a) Expand the spectrum of MSLB cases considered to include various break sizes at the different power levels;

(b) For the case which results in the maximum containment atmosphere temperature, graphically show the containment atmosphere temperature, the containment shell temperature, and the containment concrete temperature as a function of time;

(c) For the case which results in the maximum containment atmosphere pressure, graphically show the containment pressure as a function of time; and

(d) Specify and justify the design temperature of the containment structure shell and concrete, the design temperature of the internal structures, and the temperature used to qualify the safety-related instrumentation located within the containment.

022.7 (6.2.1) Quality assurance and operability testing are not sufficient bases for assuming no single failure of either a main steam isolation valve or main feedwater isolation valve. Therefore, provide MSLB analyses considering each of those single failures.

022.8 (6.2.1) In the unlikely event of a pipe rupture inside a major component subcompartment, the initial blowdown transient would lead to nonuniform pressure loadings on both the structure and the enclosed component(s). To assure the integrity of these design features, we require that you perform a subcompartment, multi-node pressure response analysis, and provide the following information:

022.8
(6.2.1)
(Cont'd)

(a) Provide the results of analyses of the pressure transient resulting from postulated hot-leg and cold-leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity, pipe penetrations, and steam generator compartments. Provide the results of similar analyses for the pressurizer surge and spray lines, and other high energy lines located in containment compartments that may be subject to pressurization.

(b) Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design or both.

(c) For each compartment provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.

(d) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

(e) Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

022.8 (6.2.1) (Cont'd)

(f) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially and radially within the compartment. Describe and justify the nodalization sensitivity study performed for the major component supports evaluation, where transient forces and moments acting on the components are of concern.

(g) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

(h) Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressure on structures and components.

(i) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed.

022.8 (6.2.1) (Cont'd)

(j) Provide the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$) and transient moments (e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$) as resolved about a specific, identified coordinate system.

(k) Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

022.9 (6.2.2)

Discuss the extent to which insulation in the vicinity of a postulated pipe break could be stripped from piping and components and identify the insulation involved. Discuss the potential for loose insulation and other debris to clog drains leading to the sump (e.g., within the refueling canal) and the sump screening.

022.10 (Table 6.2-30)

Expand Table 6.2-30 to include the number of actual pentrations corresponding to each penetration number and the flow direction for each penetration.

022.11 (RSP) (6.2.6)

It is stated in 6.2.6.1 that integrated leakage rate retest after repair (of a local leak) will not be required provided that the calculated integrated leakage rate with the difference in leakage rates for the affected components (before and after repair) meets with the acceptability criteria.

It is our position that the difference in Type B and C test results before and after the repair of local leaks may not be deducted from the Type A test result in order to achieve an acceptable containment integrated leak rate.

Conversely, it is our position that local leakage rates measured before and after repair must be reported, and the sum of the post-repair leakage rate and the Type A test results must meet the Appendix J allowable leakage rate (0.75 La).

022.12
(RSP)
(6.2.4)

Discuss how the containment purge system meets the recommendations in Branch Technical Position CSB 6-4. Identify also those areas which are not in compliance with our position.

022.13
(6.2.5)

Combustible gas control systems and the provisions for mixing, measuring and sampling should meet the design, quality assurance, redundancy, energy source, and instrumentation requirements for an engineered safety feature. Discuss how the proposed system meets these provisions and justify any deviations from this policy.

030.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

030.1 (7.2) It is not clear whether the failure modes and effects analysis (FMEA) in Table 7.2-5 is based on four operable plant protection system channels or on three operable channels with one channel in the bypass mode. Clarify the basis for this table.

030.2 (7.2) If Table 7.2-5 is based on four operable channels, provide a similar FMEA based on three operable channels and one bypassed channel. This analysis is necessary to demonstrate that the facility can be allowed to operate continuously and indefinitely with one plant protection system channel in the bypass mode.

030.3 (7.2) Describe in detail all the differences between the San Onofre Core Protection Calculator System (CPCS) and the reference CPCS (ANO-2). Include in the description all differences in design, qualification and criteria.

130.0 STRUCTURAL ENGINEERING BRANCH

130.1 (3.3.2) State the tornado load combinations such as wind, pressure and tornado generated effects and the corresponding load combination equations.

130.2 (3.4.1) Provide the flood and/or the highest ground water design level for Seismic Category I structures and components. The discussion in this section should be consistent with Section 2.4.1.1, 2.4.2.2 and 2.4.10.

130.3 (3.4.2) Describe the methods and procedures by which the static and dynamic effects of the design loads identified in Section 2.4 are applied to safety related structures, systems and components. Also, consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loading, and the static and dynamic effects on foundation properties (Section 2.5) should be provided.

130.4 (3.5.3) Procedures utilized for the prediction of the overall response should be described. This includes assumptions on acceptable ductility ratios and estimates of forces, moments and shears induced in the barrier by the impact force of the missile. Section 2 of BC-TOP-92 referred to in the FSAR describes only the methods used to assess the local damage due to missile impact.

130.5 (3.7.1) Section 3.7.1 shows only one horizontal earthquake record. When a time history analysis method is used with simultaneous input of three component earthquakes, the earthquakes specified in the three mutually orthogonal directions should be shown to be statistically independent. Provide an additional horizontal and a vertical time history and demonstrate that the three earthquake components are statistically independent.

130.6 (3.7.2) Provide the details of the analytical procedures used to account for significant effects such as hydrodynamic effects and nonlinear response.

130.7 (3.8.2) (a) Describe and give the pertinent design criteria of the polar crane supporting element.

(b) Describe the extent to which the steel structures, such as the linear supports for the reactor coolant system, comply with Subsection NF of the ASME Code, Section III Division 1.

130.8 (3.8.4) (a) The FSAR shoud contain a description of the miscellaneous Category I structures such as the Category I electrical manholes, pipes and tunnels leading from the intake structure to the plant.

(b) Cross deformations and strains for each structure should be addressed in the FSAR.

130.9 (3.8.5) The minimum factor of safety against sliding and buoyant forces (as applicable) should be stated in the FSAR.

130.10 (3.7, 3.8) Indicate which of the topical and project reports referenced in Sections 3.7 and 3.8 have been approved by the staff. If the referenced document has been approved by the staff, but in a different version, identify the differences between the referenced version and the staff approved version.

130.11 (3.8.1) The major document used in design of the containment is the Bechtel topical report BC-TOP-5 Rev. 1, December 1972. The staff reviewed and approved Revision 3 of this topical report which reflects the main provisions of the Article CC-3000 of the "Code for Concrete Reactor Vessels and Containments" ASME, Division 2 (ACI-359). In view of the above, the differences between the criteria used in design of the containment and the approved version of BC-TOP-5 should be itemized and compared in sufficient detail to enable the staff to make the determination that the criteria used in design of the plant are acceptable and comparable regarding current level of conservatism.

130.12 (3.8.1) The FSAR should contain description of the main provisions of the design and analysis procedures, particularly with respect to the following:

1) Assumptions on boundary conditions.
2) Treatment of transient and localized loads.
3) The treatment of the effects of seismically induced tangential (membrane) shears.
4) The evaluation of the effects of variations is specified physical properties of materials on analytical results.
5) Treatment of creep, shrinkage and cracking of concrete.

310.0 ACCIDENT ANALYSIS BRANCH

312.1 (2.1.1) Clarify the relationship between the site boundary or plant property line, as shown on Figure 2.1-1, and the exclusion area boundary, as shown on Figure 2.1-2.

312.2 (2.1.2) Discuss how the plant operator will become aware of any military operations or the presence of military personnel within the exclusion area in the event of emergency.

312.3 (2.2.2) Amplify the discussion on pages 2.1-1 and 2.2-2 by specifically discussing what types of military operations or movements of personnel are permissible or could occur within the LPZ and exclusion area.

312.4 (2.2.2) Indicate the closest distance of approach of the 12-inch diameter Southern California Gas Company Pipeline. Indicate whether liquified petroleum products are presently carried or contemplated being carried in this line.

312.5 (2.2.3) Provide an analysis of an accident involving the 12-inch Southern California Gas Company pipeline. Discuss the effects of such an accident both with and without giving credit for plume rise. State all your assumptions.

312.6 (RSP) (3.5) It is the staff's position that plants whose CP review was conducted prior to 1973 must provide adequate protection of structures housing safe shutdown equipment at least against tornado missiles "C" and "F" in SRP 3.5.1.4. Describe the degree of protection provided by the barriers listed in Table 3.5-14 of the FSAR against the above tornado missiles.

312.7 (3.5) Note "i" on p. 3.5-12, in reference to turbine missiles in Table 3.5-1, p. 3.5-11 needs clarification, since a turbine missile analysis is discussed in Section 3.5 of the FSAR.

312.8 (3.5) (10.2) Section 3.5.1.3.5 refers to Section 10.2 for a discussion of turbine valve testing frequency. However, Section 10.2 does not appear to include a discussion of valve testing frequency. Unless this information is included elsewhere in the FSAR, describe the valve testing frequency that will be used and include a discussion of the load shedding effects, if any, associated with the valve testing.

312.9 (3.5) Clarify the difference between the turbine vendor names English Electric (PSAR) and General Electric Company, Ltd. (FSAR), and discuss the changes (if any) in the turbogenerator design, fabrication, operation, testing, and maintenance that may have occurred since issuance of the CP for Unit Nos. 2 and 3, and that are relevant to turbine missile risks.

312.10 (3.5) Indicate the direction of rotation of each turbine (e.g., clockwise when viewing turbine axis from north to south).

312.11 (3.5) Provide a basis for the concrete strengths of 4000 and 6000 lb/in^2 used in evaluating P_3 in Section 3.5.1.3.3.4 (e.g., technical specifications, as measured, etc.).

312.12 (3.5) With respect to the target structures within the low trajectory turbine missile strike zones (as indicated in Figure 3.5-2), provide a sketch indicating the relative location of safety related areas, systems, or equipment within each target structure. Provide redundancy (e.g., "Train A, Train B") and separation (e.g., System A separated X feet from System B).

312.13 (3.5) (1.2) Describe quantitatively the reactor missile shield indicated in Figure 1.2-11.

312.14 (6.2.2)

(a) Identify all sprayed and unsprayed regions and estimate their respective volumes. Suitable diagrams and tables should be used where appropriate.

(b) Provide the lengths of time of injection under conditions of maximum and minimum ECCS flow. Indicate if there may be times when there is no NaOH addition to the spray water.

(c) Provide curves of spray nozzle water pH vs. time under the ECCS flow rates as mentioned above, until the time when NaOH addition is terminated.

(d) Page 6.5-12 mentions two proportionality constants. State the magnitudes of these constants.

(e) Provide plots of sump pH vs. time under the different conditions mentioned in b., until the time when sump pH is stabilized (after NaOH addition is terminated).

312.15 (6.4) (7.1) Section 6.4.3.2 indicates that the control room emergency ventilation mode is initiated upon receipt of a control room outside air intake high radiation signal. Section 7.1.1.7 also includes the safety injection actuation signal as causing control room isolation. Clarify within Section 6.4.3.2 whether both types of signals are to be used for control room emergency mode initiation.

312.16 (6.4) With respect to item G of Section 6.4.4.3, specify quantitatively the "adequate supply of protective clothing, respirators, and self-contained breathing apparatus..."

312.17 (6.4) In reference to item C of Section 6.4.2.2.2, it is not clear if the charcoal filter description (e.g., iodine removal efficiency of 95%) applies to just the outside air emergency filtration units or is meant to include also the emergency recirculation units. Clarify this aspect insofar as it relates to the iodine removal credit that can be given for the system when evaluating potential control room operator accident doses.

312.18 (6.4) The term "slightly positive" is insufficient in describing the control room pressurization under accident conditions. Provide a quantitative estimate of the pressurization (e.g., 1/8-inch water-gauge pressure above air zones adjacent to the control room habitability envelope), and give a basis for the estimate (e.g., exfiltration analysis, tests).

312.19 (6.4) In reference to item F of Section 6.4.1, provide a more detailed description of the habitability system capability for detecting noxious gases.

312.20 (6.4) List all toxic gases which may be stored on site and provide quantitative information relevant to Regulatory Guides 1.78 and 1.95.

312.21 (15.1.3.1, 15.4.3.2, & 15.6.3.2) Provide the following information necessary for dose calculations:

(a) Approximate mass of metal in contact with the RCS water.

(b) Steam generator secondary side volume.

(c) Air ejector flow rate for normal operation.

(d) Letdown rate during normal operation.

(e) Amount of water in RCS during normal operation.

(f) Volume fraction of water in the steam generators under normal operating conditions.

Additionally, for the case with the most severe radiological consequences for each of the three accidents, the following information is needed:

(g) Auxiliary feedwater system - initiation time and flow rate to each SG. In light of this, also provide time-dependent liquid volume fractions in each SG after the accident for a duration of two hours.

(h) Curves showing pressure (and temperature if there is super-heating) changes inside the reactor vessel, the intact SG and the failed SG, for a duration of two hours after the accident.

312.22 (15.6.3.1) State if flow restricting orifices are present in instrument lines to limit leak rates should a rupture occur.

320.0 EFFLUENT TREATMENT SYSTEMS BRANCH

320.1 (9.2.1.5, 11.5) Section 9.2.1.5 states that the discharge from the saltwater cooling system is not monitored for radioactivity content. Continuous monitoring of gross radioactivity concentration for this potentially radioactive discharge release should be provided.

Demonstrate how radioactivity concentrations exceeding a predetermined level will be alarmed at the reactor control room.

320.2 (11.3.1.6, 9.3.2) In Subsection 11.3.1.6, "Hydrogen Control", states that hydrogen and oxygen analyzers in the gaseous radwaste system initiate alarms at predetermined setpoints prior to reaching a potentially explosive hydrogen-oxygen mixture and that manual action by the operate is required (to correct abnormal conditions). For systems not designed to withstand a hydrogen explosion, hydrogen or oyxgen analyzers which actuate automatic control functions to preclude the formation or buildup of explosive hydrogen-oxygen mixtures should be provided. Section 9.3.2 states that the waste gas surge tank is sampled and analyzed intermittently on a timed cycle and that other points are selected manually.

For systems not designed to withstand a hydrogen explosion, provide, in addition to the timed-cycle analyzers described in Subsection 9.3.2, a continuously operating hydrogen or oxygen analyzer on the liner between the compressor outlet and the waste gas decay tank inlet; this analyzer should also have the automatic control functions described above.

320.3 (11.5) In Table 11.5-1, "Continuous Process and Environmental Radiation Monitoring," the concentration values for range, expected concentrations, and alarm setpoint are shown in "Ci/cm^3"; these appear to be typographical errors and should be changed to read either "uCi/cm^3" or "Ci/m^3". Also in Table 11.5.1, the range of the Condenser Air Ejector High Range Monitor is given as $10^{-2}(u)Ci/cm^3$; this also appears to be a typographical error and should be read either $10^2 uCi/cm^3$ or $10^2 Ci/m^3$.

330.0 RADIOLOGICAL ASSESSMENT BRANCH

331.1 (12.2.2.1) The term ($V\lambda_{Ti}$) should appear in the numerator, not the denominator, of the first equation in section 12.2.2.1.

331.2 (Table 12.2-6) Give the units of the values in Table 12.2-6.

331.3 (Table 12.2-8) Include expected particulate isotopes in the table of normal airborne radioactivity concentrations (Table 12.2-8).

331.4 (12.3.1) Adequate and rapidly serviceable lighting should be provided for each room or cubicle containing zone III-V components. State how the above will be implemented.

331.5 (Table 12.4-9) Do the figures in the last four columns of Table 12.4-9 presented the doses received by individual plant employees working in the areas listed, or do they represent the total accumulated personnel dose commitment caused by airborne radioactivity in these areas? If these figures represent individual worker doses, list the total number of personnel expected to work in each of the areas listed and give the total estomated personnel dose caused by airborne radioactivity.

331.6 (12.5.2.1) On figures 12.3-1 through 12.3-25 indicate the major traffic patterns used by plant personnel during their daily activities. Also describe the route a plant employee would take in going from the main entrance to the controlled area.

331.7(a) (12.5.2.2.5) The airborne radioactivity monitoring system should be sensitive enough to indicate that an airborne radioactivity hazard exists in any compartment (or area) for which the monitor is applicable. Assume a 1 MPC concentration of the most representative particulate and gas is present in the compartment with the lowest flowrate in the area being monitored during normal operation. For each airborne radioactivity monitor in the plant, give the response time to detect this concentration.

331.7(b)
(12.5.2.2.5)

In order to adequately detect airborne radioactivity in areas which may be occupied by personnel, airborne radioactivity monitors should be located upstream of the air cleaning systems. It is not clear from studying Fig. 9.4-9 (sheet 2) whether this is being done for the exhaust air from the fuel handling building. Provide assurance that all airborne radioactivity monitors sampling air from the areas which may be occupied by personnel have sampling points upstream of the air cleaning systems. Provide HVAC drawing 40090.

360.0 SEISMOLOGY/GEOLOGY BRANCH

361.0 Geology Section

361.1 (2.5) The FSAR has not defined the Safe Shutdown Earthquake for the San Onofre site but has instead determined a "Design Basis Earthquake." The facility is required to meet the seismic and geologic siting criteria of Appendix A to 10 CFR 100. If the term "Design Basis Earthquake" is to be used as a synonym for "Safe Shutdown Earthquake," so indicate in the FSAR. If the terms are not synonymous, specify in detail how they differ, including the general guidelines used to establish a Design Basis Earthquake; provide supporting data to justify such a departure from the criteria of Appendix A to 10 CFR Part 100.

361.2 (2.5) Provide a detailed description of the seismic network, which is presently monitoring seismic activity in the site area, the resulting fault plane solutions, and epicentral locations not previously reported. A discussion of the relation between seismic activity and geologic structure in the site area should be provided.

370.0 HYDROLOGY/METEOROLOGY BRANCH

371.0 Hydrology Section

371.1 (2.4.2) (RSP) We note that HMR-36 was used to determine the precipitation for the Probable Maximum Flood. A "Preliminary Report, Probable Maximum Thunderstorm Precipitation estimates - Southwest States" prepared by the National Weather Service in August of 1972 describes criteria acceptable to the staff. It is our position that this criteria must be used to determine the Design Basis Flood Level.

371.2 (2.4.2) Provide detailed drawings and cross-sections of the site clearly showing site topography; site drainage features, such as ditches, culverts, etc.; the areas of impoundment; the location and size of drains; and an outline showing the maximum water surface elevation reached by the PMF.

371.3 (2.4.3) Provide design details for the diversion structure east of Interstate Highway 5. Include a topographic map of the area with sufficient detail to show the storage available behind the dike. Also, provide the routing of the PMF showing inflow, stage and outflow. Provide the bases for the storage set aside for debris.

371.4 (2.4.3) Provide an analysis of the floods (and the potential for debris production) after a fire in the watersheds above the plant.

371.5 (2.4.13) (3.4.1) The Design Basis Groundwater Level as stated in the FSAR is elevation + 5.0 Ft. MLLW. The Design Basis Flood Level is stated as elevation + 30.0 Ft. MLLW. Section 3.4 of the FSAR states that various perimeter walls are to be used for flood protection; however, this section does not provide the design hydrostatic level of the various structures. Accordingly, provide these design hydrostatic levels (and the bases for these levels) used in the design analysis for all safety-related systems and structures. Document that the structures are capable of resisting the above levels.

370.0 HYDROLOGY/METEOROLOGY BRANCH

372.0 Meteorology Section

372.1 (2.3.1.2) Section 2.3.1.2 states that "the dust storm potential at San Onofre is estimated utilizing data from the San Mateo Point". Describe the type of data used in the estimate.

372.2 (2.3.3.5) Provide a description of all major (greater than 24 hours) or recurring meteorological instrument outages, and the corrective actions taken, during the period of the onsite data record. (1/73-1/76).

372.3 (2.3.5) Provide an evaluation of long-term atmospheric diffusion estimates consistent with Appendix I to 10 CFR Part 50 and with Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors. This evaluation should include.

(a) estimates of relative concentrations (X/Q) and deposition (D/Q) as appropriate to distances up to 50 miles;

(b) a description of the atmospheric transport and diffusion model selected to obtain these X/Q and D/Q estimates. Describe the validity and accuracy of these estimates considering the model, input data, and site and regional characteristics. For example Regulatory Guide 1.111 suggests that use of a straight-line airflow model, such as the one discussed in FSAR Section 2.3.5, may underpredict concentrations at certain downwind receptors due to this model's inability to account for spatial and temporal variation in the air flow. At San Onofre such phenomena as the sea-land breeze circulation, mixing-layer depth variation with time and distances from the shoreline, the shoreline bluff effects, channeling and/or obstruction to the airflow by the coastal mountains may effect the model's predictions. Discuss any corrections necessary to adjust the model you selected.

430.0 INDUSTRIAL SECURITY AND EMERGENCY PLANNING BRANCH

432.1 The information requested below, required by Regulatory Guide 1.70 (Revision 2), was not included within the Emergency Plan tendered for San Onofre Units 2 and 3. Provide this information so that the adequacy of the San Onofre Units 2 and 3 Emergency Plan may be evaluated.

(a) Provide the detailed maps (or specific reference to the appropriate maps located elsewhere within the FSAR) described in paragraphs 6.a. and 6.b. of Section 13.3 of Reg. Guide 1.70 (Revision 2).

(b) Provide the agreements reached with offsite local, state and federal officials and agencies as required by 10 CFR 50 Appendix E, paragraph IV.D (Note: This material is referred to as being in Appendix F to the Emergency Plan; however, Appendix F was not included in the tendered application).

440.0 OPERATOR LICENSING BRANCH

442.1 (13.5) Provide a commitment to conduct all safety-related operations in accordance with detailed written and approved procedures.

442.2 (13.5) Provide the commitment that all administrative and operating procedures will be completed at least six months before fuel loading.

Enclosure 3

Request for Additional Information
Resulting from Acceptance Review of
San Onofre Nuclear Generating Station
Units 2 and 3, Environmental Report

REQUEST FOR ADDITIONAL INFORMATION
ENVIRONMENTAL REPORT

Section 2.2.2

1. Identify in the Environmental Report (ER) the aquatic species, appearing on either state of Federal lists of threatened or endangered species, which might be affected by operation of the San Onofre Station.

Section 2.2.2.5

2. Provide a copy of Figure 2.2-8 which was omitted from the ER.

Section 2.4.3.5

3. Provide the basis for the receiving water temperature fluctuations cited in this section.

4. Provide available data which describes the characteristics such as depth, thickness and gradient of the thermocline at various times of the year.

5. Provide the raw data or the summary reports which comprise the basis for the numerous statements on the dissolved oxygen, pH, turbidity and coliform characteristics of the waters near the SONGS site. Presently, these statements are not referenced adequately.

6. Provide a discussion of the existing environmental stresses on the aquatic environment near the San Onofre site.

Section 3.3

7. Tables 3.3-1 and 3.3-2 are missing. Provide these tables.

8. Indicate the plant power level referred to in Section 3.3.14.1 as a basis for Table 3.3-2 and Figure 3.3-1. Provide similar data as in these tables for the normally anticipated plant load and minimum power level anticipated for the plant under normal conditions.

Section 3.4.4

9. Indicate the frequency of the heat treatment of the plant cooling water system.

Section 3.4.5

10. Describe procedures, species and size classes of fish used in the model testing which resulted in the proposed design of the Fish Conservation system. If results have been presented in a report, provide reference or copies (if none have been submitted to the NRC).

Section 3.6.1

11. Clarify the meaning of "...the maximum concentration of free residual chlorine during any chlorination is less than 0.5 mg/liter in the immediate vicinity of the discharge."

12. The stated expected residual chlorine concentration in this section is not consistent with that discussed in Section 5.3. Clarify this apparent discrepancy.

Section 3.6.2

13. Indicate the water quality characteristics and quantity of the expected releases from the Overboard System.

Table 3.6-1

14. The waste discharge should also indicate the concentrations of expected waste products to be discharged under normal and worst case conditions.

Section 4.2.1

15. Provide the staff with a copy of all Southern California Edison's (SCE) standard specifications relating to siting, construction and operating procedures employed to avoid and/or mitigate transmission system environmental impacts.

Section 4.2.2.1

16. Provide the staff with a copy of "SDG&E Foreman's Guide for Improving the Appearance of Transmission Lines" and all other SDG&E Guidelines or Specifications relating to siting, construction and operation of transmission lines employed to avoid and/or mitigate environmental impacts.

Section 5.1.3.4.2

17. Provide information on the procedures used in the monitoring of ichthyoplankton entrainment for Unit 1. Provide data when results of this study are anticipated.

Section 5.3

18. Provide a copy of the recently issued NPDES Permit to San Onofre Units 2 and 3.

Section 5.5

19. Specify design characteristics and intended mitigative actions to be utilized to minimize any effects on radio and television reception due to transmission system operation.

20. Indicate the maximum design ground level field gradients for all lines.

21. State the exact types of right-of-way maintenance procedures to be used. If any herbicides are to be used, indicate this fact as well as formulation and use compliance with State and Federal registration requirements.

22. Indicate by topographic maps all habitats along the proposed transmission line ROWS classified by State and Federal Authorities as being critical in endangered, rare, threatened or protected wildlife and plant species.

23. Indicate how much prime or unique farmlands (land in capability Class I, most of Class II, and Class III that has an adequate water management system such as pivot irrigation - Refer to the Soil Conservation Service's Land Inventory and Monitoring Memorandum-3, October 15, 1975 will be affected by the proposed transmission siting.

Section 6.1.3.1

24. Provide a description of all major (greater than 24 hours) or recurring meteorological instrument outages, and the corrective actions taken, during the period of your onsite data record (1/73-1/76).

Section 6.1.3.2

25. Provide an evaluation of long-term atmospheric diffusion estimates consistent with Appendix I to 10 CFR Part 50 and with Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors. This evaluation should include:

    (a) estimates of relative concentrations (X/Q) and deposition (D/Q) as appropriate to distances up to 50 miles;

    (b) a description of the atmospheric transport and diffusion model you select to obtain these X/Q and D/Q estimates. (You should describe the validity and accuracy of your estimates considering your model, input data, and site and regional characteristics. For example, Regulatory Guide 1.111 suggests that use of a straightline airflow model, such as the one you discuss in ER Section 6.1.3.2.2 may underpredict concentrations at certain downwind receptors due to this model's inability to account for spatial and temporal variation in the air flow. At San Onofre such phenomena as the sea-land breeze circulation, mixing-layer depth variation with times and distance from the shoreline, the shoreline bluff effects, channeling and/or obstruction to the airflow by the coastal mountains may effect your model's predictions. Thus your discussion should provide any corrections necessary to adjust the model you select).

Section 6.2

26. Provide a description of the operational monitoring program for meteorology. If there are no differences between the preoperational and operational programs, please state this and commit to conduct the operational program.

Section 6.3.1

27. Provide a copy of the proposed Thermal Exception Studies.

Appendix 6B

28. The proposed ETS do not include Fish Entrainment and Impingement Monitoring Programs. Provide justification for omission of these programs or supplement the proposed ETS with proposed studies.

Section 10.9

29. Construction of the proposed transmission routes is scheduled to begin September 1977. Indicate all changes to siting and design of the proposed system since submittal of the Construction Permit Environmental Report. Provide the staff with a description of the selection method used to determine the proposed routes and provide a map showing all alternative routes considered in the selection process.

JAN 2 4 1977

Distribution
Docket File
NRC PDR
Local PDR
LWR #2 File
DBVassallo
KKniel
LChandler
STreby
HRood
MMlynczak
VAMoore

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

ACCEPTANCE REVIEW OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3, FSAR

On December 1, 1976, you tendered an application for operating licenses for the San Onofre Nuclear Generating Station, Units 2 and 3. Your application included the Final Safety Analysis Report, General Information, the Environmental Report and Antitrust Information. The acceptance review was initiated upon receipt of the application.

In order to complete our acceptance review, we must determine the completeness of the FSAR as tendered. Because all technical information necessary for the completion of an effective independent evaluation of the SONGS 2 and 3 design has not been included in the FSAR, we require definition of anticipated submittal dates prior to docketing the application and subsequent initiation of the detailed review.

We require submittal dates compatible with the anticipated review schedule for each of the following items:

(1) CPC (Core Protection Computer) detailed drawings (Section 1.7); CPC software submittal;

(2) Electrical Instrumentation and Control Drawings (Section 1.7);

(3) Instrument Location Layout Drawings (Section 1.7);

(4) Bechtel proprietary drawings (Section 1.7);

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

(5) Seismic Qualification data, BOP (Section 3.10B);
Seismic Qualification data, NSSS (Section 3.10);

(6) Environmental Qualification data, BOP (Section 3.11A);
Environmental Qualification data, NSSS (Section 3.11);

(7) Flood Protection Design (Section 2.4);

(8) High Energy Pipe Break Analysis (Section 3.6).

Your commitment to provide the above information within six months will allow docketing of the application and initiation of the detailed review of the FSAR. If you cannot make this commitment, please provide us with the earliest date that you can practically meet for each item, and we will use those dates as a basis for making a decision regarding docketing your application at this time.

Sincerely,

Original signed by
D. B. Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

cc: See page 3

OFFICE	DPM:LWR #2	DPM:LWR #2	DPM:AD/LWR			
SURNAME	MMlynczak:mt	KKniel	DBVassallo			
DATE	1/21/77	1/2/77	1/21/77			

cc: Rollin E. Woodbury, General Counsel
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Mr. B. W. Colston
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Distribution
Docket File
NRC PDR
Local PDR
LWR #2 File
RSBoyd
DBVassallo
HRood
MMMlynczak
LChandler
STreby
RWFroelich

DEC 0 9 1976

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

ACCEPTANCE REVIEW OF SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

On December 1, 1976, we received your letter dated November 30, 1976 tendering your application for operating licenses for San Onofre Nuclear Generating Station, Units 2 and 3. The tendered application includes the following documents:

1. Final Safety Analysis Report

2. General Information

3. Environmental Report

4. Antitrust Information

Project Nos. 50-361 and 50-362 previously assigned to your application will remain applicable. All future correspondence should reference these numbers.

We began an acceptance review on December 1, 1976 to determine the completeness of these documents and expect to complete our review by January 7, 1977. Upon completion of our review, we will notify you by letter as to the conclusions and arrange a meeting with you to discuss these conclusions.

appl 3

OFFICE ➤						
SURNAME ➤						
DATE ➤						

A copy of your application has been placed in the NRC Public Document Room in Washington, D. C., and a local public document room has been established at Mission Viejo Branch Library, 24851 Chrisanta Drive, Mission Viejo, California 92676. A copy of the application, and other relevant documents, as they become available, will be on file for public inspection. It is requested that you have one of your representatives make periodic checks of the material available and assure that any revised and supplemental information is properly incorporated into the Application, the Final Safety Analysis Report and the Environmental Report and that any amendments, reports, and letters which you file with us are available. We will send documents you file with us to the local public document rooms.

Sincerely,

Original signed by
K. Kniel

Karl Kniel, Chief
Light Water Reactors
Branch No. 2
Division of Project Management

cc: See page 3

OFFICE➤	DPM:LWR #2	DPM:LWR #2	DPM:LWR #2			
SURNAME➤	MMlynczak:mt	HRood	KKniel			
DATE➤	12/ /76	12/ /76	12/ /76			

DEC 0 9 1976

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
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Center for Law in the Public Interest
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Los Angeles, California 90067

Mr. Kenneth E. Carr
City Manager
City of San Clemente
100 Avenido Presidio
San Clemente, California 92672

Alan R. Watts, Esq.
Assistant City Attorney
City Hall
Anaheim, California 92805

Lawrence Q. Garcia, Esq.
California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

Mr. B. W. Colston
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

OFFICE ➤						
SURNAME ➤						
DATE ➤						

cc: Mr. W. D. Griffith
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

NOV 1976

50-361/362

Gentlemen:

In reviewing our requirements for copies of licensees' and applicants' Industrial Security Plans, we have determined that a change in the distribution and number of copies would provide greater efficiency in our reviews. Consequently, for all future submittals of security plans and requested changes to security plans (amendments) you should send five (5) copies to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. This change supersedes the current instructions in Regulatory Guide 10.1. Therefore, you should no longer send one copy directly to the NRC Regional Office. Regulatory Guide 10.1 will be modified to reflect this change in its next revision.

As in the past, the cover letter that transmits a security-related attachment that is to be withheld from public disclosure should be so identified (e.g., stamped "ATTACHMENT TO BE WITHHELD FROM PUBLIC DISCLOSURE").

Sincerely,

D B Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Form letter sent to the following companies:

Alabama Power Company
Arizona Public Service Company
Arkansas Power & Light Company
Babcock & Wilcox Company
Baltimore Gas & Electric Company
Boston Edison Company
C. F. Braun & Company
Carolina Power & Light Company
Central Maine Power Company
Cincinnati Gas & Electric Company
Cleveland Electric Illuminating Company
Combustion Engineering, Inc.
Commonwealth Edison Company
Consumers Power Company
Department of Water and Power, City of Los Angeles
Detroit Edison Company
Duquesne Light Company
Florida Power Corporation
Florida Power & Light Company
General Electric Company
Gibbs & Hill
Gulf States Utilities Company
Houston Lighting and Power Company
Illinois Power Company
Indiana & Michigan Electric Company and
Indiana & Michigan Power Company
Jersey Central Power & Light Company
Kansas Gas & Electric Company
Long Island Lighting Company
Louisiana Power & Light Company
Metropolitan Edison Company
Mississippi Power and Light Company
Niagara Mohawk Power Corporation
Northeast Nuclear Energy Company
Northern Indiana Public Service Company
Northern States Power Company
Offshore Power Systems
Omaha Public Power District
Pennsylvania Power and Light Company
Philadelphia Electric Company
Portland General Electric Company
Potomac Electric Power Company
Power Authority of the State of New York
Public Service of Indiana
Public Service Company of New Hampshire
Public Service Company of Oklahoma
Puerto Rico Water Resources Authority
Puget Sound Power & Light Company
Rochester Gas & Electric Corporation
San Diego Gas & Electric Company
South Carolina Electric and Gas Company
Southern California Edison Company

Stone & Webster Engineering Corporation
Tennessee Valley Authority
Texas Utilities Generating Company
Toledo Edison Company
Union Electric Company
Virginia Electric and Power Company
Washington Public Power Supply System
Westinghouse Electric Corporation
Wisconsin Electric Power Company
Duke Power Company
Georgia Power Company
PUblic Service Electric and Gas Company
Pacific Gas and Electric Company
Project Management Corporation
Public Service Company of Colorado

Distribution
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NRC PDR
Local PDR
LWR #3 File
JCollins
EIGoulbourne (2)
RFroelich, EP
JRBuchanan, NSIC
TBAbernathy, TIC
FJWilliams
ELD
IE (3)
ACRS
HRood
PKreutzer, EP

JUN 2 4 1976

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

This letter is to inform you of recent events and conclusions concerning our generic review of Anticipated Transients Without Scram (ATWS) as they relate to San Onofre Nuclear Generating Station Units 2 and 3.

As you know, upon the publication of our report on ATWS in September 1973, WASH 1270, we sent you a letter on October 10, 1973 requesting analysis of ATWS in compliance with WASH 1270 as it applied to your facility. Your response of December 26, 1973 indicated that the NSSS vendor for Units 2 and 3, Combustion Engineering, was developing methods for ATWS analysis, and following our approval of these methods, ATWS analyses would be performed for San Onofre Units 2 and 3. You also indicated that these analyses would be used to identify any design changes needed to make the consequences of ATWS acceptable.

You are probably aware that the NRC staff and Combustion Engineering personnel have been engaged in an extensive effort to resolve our differences concerning the Combustion Engineering ATWS analysis model. We published our Status Report (Enclosure 1) on December 9, 1975 wherein we stated that additional analyses and justification of the Combustion Engineering analysis model are needed and that changes in typical Combustion Engineering plant designs are indicated. At the 189th Meeting of the Advisory Committee on Reactor Safeguards (ACRS), the ATWS issue and our Status Reports were reviewed in conjunction with the staff and Combustion Engineering.

We now consider our review of the Combustion Engineering ATWS methods to be complete and our position clearly and adequately presented in the Status Report. We have written to Combustion Engineering (Enclosure 2) informing them that we expect by June 30, 1976 they can

provide us with acceptable additional analyses and justification of the Combustion Engineering analysis model as identified in our Status Report. Therefore, we request that you provide by December 30, 1976 the following information:

a. The additional analyses and justification of the Combustion Engineering analysis model identified in the Status Report (We anticipate that you will reference the generic June 30, 1976 Combustion Engineering analysis model).

b. Based on these analyses, identification of the design changes needed to assure that the limits specified in WASH-1270 will not be violated following an ATWS event.

c. A schedule for the submittal of a detailed description of the design changes identified both in the Status Report and in item "b", including a detailed description of the diverse means of interrupting power to the control rod drive mechanisms.

d. A schedule for the installation of the instruments, controls, and equipment described under item "c".

We have established the above schedules for the submittal of the required information to emphasize our determination to move forward to resolve the ATWS concern and conclude the lengthy deliberations that have taken place among all parties. We expect your efforts will be directed to this same end, and if you have any questions regarding this letter, please let me know.

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072). This clearance expires July 31, 1977.

Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Status Report
2. Letter to Combustion Engineering, Inc.

cc: See page 3

OFFICE ➤	DPM:LWR #3	DPM:LWR #3	DPM:AD/LWR	OELD	DPM	
SURNAME ➤	HRood:mt	ODParr	RCDeYoung		RSBoyd	
DATE ➤	5/ /76	5/ /76	5/ /76	5/ /76	5/ /76	

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
845 North Perry Avenue
Montebello, California 90640

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Center for Law in the Public Interest
10203 Santa Monica Boulevard
Los Angeles, California 90067

Mr. Kenneth E. Carr
City Manager
City of San Clemente
100 Avenido Presidio
San Clemente, California 92672

Alan R. Watts, Esq.
Assistant City Attorney
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Anaheim, California 92805

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California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

APR 7 1976

Mr. Edward Scherer
Licensing Manager
Combustion Engineering, Inc.
Windsor, Connecticut 06095

Dear Mr. Scherer:

In December we issued the NRC Status Report on Anticipated Transients Without Scram for Combustion Engineering Reactors. In meetings with us and the ACRS you indicated that Combustion Engineering believes that ATWS is not a concern to the health and safety of the public and the changes to the design of Combustion Engineering reactors specified in the Status Report are not required. Although our further evaluation indicates that the specific recommendations for Category A plants set forth in WASH-1270 are not necessary, we continue to believe, as we have often stated, that design improvements with respect to ATWS are appropriate to maintain and improve further the safety margins provided for the protection of the public.

We have discussed these matters with the Advisory Committee on Reactor Safeguards, which has also had the benefit of your views. We stated in the Status Report that additional analyses and justification of the Combustion Engineering analysis model are needed and that changes in typical Combustion Engineering plant designs are indicated. Since then we have completed our review of the latest version of your ATWS analysis methods and will shortly inform you of the additional justification of your analysis model that is required. Therefore, we now consider our review of the Combustion Engineering ATWS analysis methods to be complete.

Previously we have discussed with you the positions which are presented in the Status Report. We believe that our positions are the same as expressed in WASH-1270 and are clearly and amply presented in the Status Report. Now that our review of the latest version of your analysis model is complete, the remaining issues concerning satisfactory completion of your ATWS analysis methods can be quickly resolved. Therefore, we wish to work with you between now and the end of June to resolve these remaining items so that the appropriate modifications may be incorporated into reactors of your design which are under licensing review.

The resolution of these matters should be scheduled so that the following information for your CESSAR application can be submitted by June 30, 1976. We also will send a copy of this letter to each applicant for a construction permit or operating license for a reactor using a Combustion Engineering NSSS for which our regulatory review is scheduled to be completed after December 31, 1976, including those referencing the CESSAR standard design, and request that the same information for their plant be provided by September 30, 1976. This information should contain:

a. The results of additional analyses and the further justification of the Combustion Engineering analysis model identified in the Status Report and its supplement.

b. Based on these analyses, identification of the design changes needed to assure that the limits specified in WASH-1270 will not be violated following an ATWS event.

Applicants for operating licenses receiving copies of this letter will be requested to supply the following information in addition to items "a" and "b" by September 30, 1976.

c. A schedule for the submittal of a detailed description of the design changes identifed both in the Status Report and in item "b", including a detailed description of the diverse means of interrupting power to the control rod drive mechanisms.

d. A schedule for the installation of the instruments, controls, and equipment described under item "c".

We will establish appropriate schedules for submission of such information by other applicants and licensees with facilities of your design.

If you have any questions regarding this letter, please let me know.

Sincerely,

Original signed by
Robert E. Heineman

Robert E. Heineman, Director
Division of Systems Safety
Office of Nuclear Reactor Regulation

Distribution
Docket File
NRC PDR
Local PDR
LWR #3 File
RCDeYoung
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VAMoore
RHVollmer
MLErnst
WPGammill
ELD
IE (3)
ACRS (16)
JMiller
JRBuchanan, NSIC
TBAbernathy, TIC

MAY 14 1976

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

On November 28, 1975, we informed you of a potential safety question which has been raised regarding the design of reactor pressure vessel support systems. We requested that you review the design bases for the reactor vessel support system for San Onofre Units 2 and 3 to determine whether the transient loads described in the enclosure to our letter were appropriately taken into account in the design.

Your reply of December 30, 1975 indicates that the transient loads were not explicitly considered in the support design.

To our letter of November 28, 1975, we attached a preliminary listing of potential requests for additional information should we later determine, on the basis of your initial review, that a reassessment of the vessel support design is required. We have now made the determination that reassessment of the vessel support design is required.

As you are probably aware, we have been discussing with the PWR vendors and various architect/engineer firms the generic aspects of this problem. Should you contemplate utilizing organizations other than your PWR vendor for calculation of the sub-cooled internal loads, we suggest you contact us for the benefit of a brief review of our generic discussions to date. We will continue these generic discussions with the vendors and architect/engineers, but such discussions are not

OFFICE ➤						
SURNAME ➤						
DATE ➤						

intended to pace your evaluation of this concern nor to eliminate the possibility that we may have additional questions regarding your evaluation after submittal. While the emphasis given in this letter deals with the reactor vessel cavity, for your information and guidance our generic review may consider other areas in the nuclear steam supply system and further evaluation may be required.

Please inform us within 30 days after receipt of this letter, your schedule for providing us your evaluation of the adequacy of the pressure vessel supports when the sub-cooled loads are calculated and taken into account in a manner which you determine best represents these phenomena. Your evaluation should include the answers to the attached request for additional information.

This request for generic information was approved by GAO blanket clearance number B-180225 (R0072). This clearance expires July 31, 1977.

Sincerely,

Original Signed by
Olan Parr

Olan D. Parr, Chief
Light Water Reactors
Branch No. 3
Division of Project Management

Attachment:
Request for Additional
Information

cc: See page 3

OFFICE➤	DPM:LWR #3	DPM:LWR #3				
SURNAME➤	HRood:mt	ODParr				
DATE➤	5/ /76	5/ /76				

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

Mr. David Sakai
845 North Perry Avenue
Montebello, California 90640

Frederick P. Sutherland, Esq.
Center for Law in the Public Interest
10203 Santa Monica Boulevard
Los Angeles, California 90067

Mr. Kenneth E. Carr
City Manager
City of San Clemente
100 Avenido Presidio
San Clemente, California 92672

Alan R. Watts, Esq.
Assistant City Attorney
City Hall
Anaheim, California 92805

Lawrence Q. Garcia, Esq.
California Public Utilities Commission
5066 State Building
San Francisco, California 94102

George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that reactor pressure vessel supports may be subjected to previously underestimated lateral loads under the conditions that result from the postulation of design basis ruptures of the reactor coolant piping at the reactor vessel nozzles. It is therefore necessary to reassess the capability of the reactor coolant system supports to assure that the calculated motion of the reactor vessel under the most severe design basis pipe rupture condition will be within the bounds necessary to assure a high probability that the reactor can be brought safely to a cold shutdown condition.

The following information should be included in your reassessment of the reactor vessel supports and reactor cavity structure.

1. Provide engineering drawings of the reactor support system sufficient to show the geometry of all principle elements and materials of construction.
2. Specify the detail design loads used in the original design analyses of the reactor supports giving magnitude, direction of application and the basis for each load. Also provide the calculated maximun stress in each principle element of the support system and the corresponding allowable stresses.
3. Provide the information requested in 2 above considering a postulated break at the design basis location that results in the most severe loading condition for the reactor pressure vessel supports. Include

a summary of the analytical methods employed and specifically state the effects of asymmetric pressure differentials across the core barrel in combination with all external loadings including asymmetric cavity pressurization calculated to result from the required postulate. This analysis should consider:

(a) limited displacement break areas where applicable

(b) consideration of fluid structure interaction

(c) use of actual time dependent forcing function

(d) reactor support stiffness.

4. If the results of the analyses required by 3 above indicates loads leading to inelastic action in the reactor supports or displacements exceeding previous design limits provide an evaluation of the following:

   (a) Inelastic behavior (including strain hardening) of the material used in the reactor support design and the effect on the load transmitted to the reactor coolant system and the backup structures to which the reactor coolant system supports are attached.

5. Address the adequacy of the reactor coolant system piping, control rod drives, steam generator and pump supports, structures surrounding the reactor coolant system, [core support structures, fuel assemblies, other reactor internals ....] and ECCS piping for both the elastic and/or inelastic analyses to assure that the reactor can be safely brought to cold shutdown. For each item include the method of

analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

The compartment multi-node pressure response analysis should include the following information:

6. The results of analyses of the differential pressures resulting from hot leg and cold leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity and pipe penetrations.
7. Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure within the reactor cavity. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variations circumferentially, axially and radially within the reactor cavity.
8. Provide a schematic drawing showing the nodalization of the reactor cavity. Provide a tabulation of the nodal net free volumes and interconnecting flow path areas.
9. Provide sufficiently detailed plan and section drawings for several views showing the arrangement of the reactor cavity structure, reactor vessel, piping, and other major obstructions, and vent areas, to permit verification of the reactor cavity nodalization and vent locations.

10. Provide and justify the break type and area used in each analysis.

11. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

12. Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical justification for the removal of such items to obtain vent area. Provide justification that vent areas will not be partially or completely plugged by displaced objects.

13. Provide a table of blowdown mass flow rate and energy release rate as a function of time for the reactor cavity design basis accident.

14. Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressures.

15. Provide the peak calculated differential pressure and time of peak pressure for each node, and the design differential pressure(s) for the reactor cavity. Discuss whether the design differential pressure is uniformly applied to the reactor cavity or whether it is spatially varied.

In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during the subcooled portion of the blowdown analysis, the following information is requested:

16. A complete description of the hydraulic code(s) used including the

development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations.

17. In support of the hydraulic code(s) used provide comparisons with the code(s) to applicable experimental tests, including the following:

    (a). CSE tests B-63 and B-75

    (b). LOFT test L1-2

    (c). Semiscale tests S-02-6 and S-02-8

    The models developed should be based on the assumptions proposed for the analysis of a PWR.

18. Provide a detailed description of the model proposed for your plant and include a listing of the input data used and a time zero edit. Identify the assumptions used in developing the model, specifically the treatment of area, length and volume.

19. Typically the current generation of hydraulic subcooled blowdown analysis codes solve the one-dimensional conservation equations. However, they are used to model the multi-dimensional aspects of the reactor system (i.e. the downcomer annulus region). Provide justification for the use of the code(s) to model multi-dimensional regions, including the equivalent representation of the region as modelled by the code(s).

5/10/76

Docket No. 50-361 and 50-362

Distribution
Docket File
LWR #3 File
NRC PDR
Local PDR
RCDeYoung
EP LA
FJWilliams
ODParr
EIGoulbourne
HRood
EPM

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

The Nuclear Regulatory Commission (NRC) has adopted amendments to Parts 2, 50 and 51 of 10 CFR which were published in the Federal Register on April 15, 1976. These amendments are procedural changes pertaining to the initial treatment of an application for a construction permit or facility operating license and amendments thereto. They become effective on May 17, 1976.

Previously, when a tendered application was determined by the NRC staff to be complete and acceptable for detailed review, applicants would be so informed and requested to submit additional copies of the application and the environmental report to the NRC for distribution to appropriate Federal, State and local officials.

Effective May 17, 1976, when a tendered application is determined by the NRC staff to be complete and acceptable for docketing, applicants will be so informed and requested to: (a) submit some additional copies of the application and the environmental report to the NRC, and: (b) make direct distribution of other additional copies of the documents to appropriate Federal, State and local officials in accordance with written instructions furnished to the applicant by the Director of Nuclear Reactor Regulation. Copy requirements are summarized in the following table:

ITEM	TENDERING	DOCKETING	
	copies submitted to NRC	copies submitted to NRC	copies retained by applicant
Application and General Information	10	15*	10
Safety Analysis Report	15	40*	30
Environmental Report	20	41*	109

NOTE: These same copy requirements apply to amendments to applications

*Plus three notarized originals

OFFICE➤						
SURNAME➤					R	
DATE➤						

In accordance with the new requirements of 10 CFR Parts 2, 50 and 51, we are attaching lists of Federal, state and local officials to whom you should make direct distribution of amendments to your application and environmental report for your San Onofre Nuclear Generating Station, Unit 2 and Unit 3. We will keep you informed of changes to these lists.

Within 10 days after docketing an amendment, you should submit to the Director of Nuclear Reactor Regulation, an affidavit that distribution of the additional copies have been completed in accordance with the requirements of the regulations and our specific instructions.

A copy of the Federal Register notice is enclosed for your information.

Sincerely,

Original signed by
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1.-3. Lists of Federal, state and local officials
for application, SAR and ER

4. Federal Register Notice published April 15, 1976
(41 FR 15832) and correction

cc: see page 3

OFFICE ➤	DPM:LWR #3	DPM				
SURNAME ➤	EIGoulbourne mjf	RSBoyd				
DATE ➤	5/ /76	5/ /76				

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
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111 Sutter Street
San Francisco, California 94104

Mr. Larry E. Moss
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Pacific Palisades, California 90272

Mr. David Sakai
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George Spiegel, Esq.
2600 Virginia Avenue, N. W.
Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ENCLOSURE 1

DISTRIBUTION LIST

Application and General Information and Amendments

State Official

Mr. Irving Goldberg, Chief
Environmental Radiation Control Unit
Radiological Health Section
California Department of Health
714 P Street
Sacramento, California 95814

Local Official

Chairman, Board of Supervisors
San Diego County
San Diego, California 92412

Mayor
City of San Clemente
San Clemente, California 92672

ENCLOSURE 2

DISTRIBUTION LIST

Safety Analysis Report and Amendments

State Official

Mr. Irving Goldberg, Chief
Environmental Radiation Control Unit
Radiological Health Section
California Department of Health
714 P Street
Sacramento, California 95814

Local Official

Chairman, Board of Supervisors
San Diego County
San Diego, California 92412

Mayor
City of San Clemente
San Clemente, California 92672

EPA Regional Office

Mr. James Channell
Environmental Protection Agency
100 California Street
San Fransicso, California 94112

DISTRIBUTION LIST

ENVIRONMENTAL REPORT, AMENDMENTS, AND SUPPLEMENTS

(Number in parens indicates number of copies)

DEPARTMENT OF COMMERCE

Dr. Sidney R. Galler (6)*
Deputy Assistant Secretary for
Environmental Affairs
U. S. Department of Commerce
14th & Constitution, N. W., Rm. 3425
Washington, D. C. 20230

Mr. Robert Ochinero, Director (1)
National Oceanographic Data Center
Environmental Data Service
National Oceanic and Atmospheric
Administration
U. S. Department of Commerce
Washington, D. C. 20235

* Generic - 13 copies
Amendments - 5 copies

DEPARTMENT OF INTERIOR

Mr. Bruce Blanchard, Director (18)
Office of Environmental Projects
Review, Room 5321
U. S. Department of Interior
18th & C Streets, N. W.
Washington, D. C. 20240

cc: (transmittal letter only)
Chief
Division of Ecological Services
Bureau of Sport Fisheries & Wildlife
U. S. Department of Interior
18th & C Streets, N. W.
Washington, D. C. 20240

DEPARTMENT OF H.E.W.

Mr. Charles Custard, Director (2)
Office of Environmental Affairs
U. S. Department of Health, Education
and Welfare, Room 4740
330 Independence Avenue, S. W.
Washington, D. C. 20201

FEDERAL POWER COMMISSION

Mr. T. A. Phillips, Chief (1)
Bureau of Power
Federal Power Commission, Rm. 5100
825 North Capitol Street, N. E.
Washington, D. C. 20426

Dr. Richard Hill (1)
Federal Power Commission, Rm. 6100
825 North Capitol Street, N. E.
Washington, D. C. 20426

DEPARTMENT OF TRANSPORTATION

(transmittal letter only addressed
to:

Mrs. Judith T. Conner
Assistant Secretary for Environment,
Safety & Consumer Affairs
U. S. Department of Transportation
400 7th Street, S. W., Rm. 10101
Washington, D. C. 20590

cc of transmittal letter to:
Captain William R. Riedel
Water Resources Coordinator
W/S 73 U.S.C.G., Room 7306
U. S. Department of Transportation
400 7th Street, S. W.
Washington, D. C. 20590

(after DES is issued, send
4 copies of ER and amendments
to Riedel)

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Durham, North Carolina 27711

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(Coastal (salt water) only)
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Director
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Mr. Richard H. Broun
Environmental Clearance Officer
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ARMY ENGINEERING DISTRICT

U. S. Department of the Army (1)
Corps of Engineers
P. O. Box 2711
Los Angeles, California 90053

RIVER BASIN COMMISSION

Not applicable

ADJOINING STATES

Not applicable

LOCAL OFFICIAL

Chairman
Board of Supervisors
San Diego County
San Diego, California 92412

Mayor
City of San Clemente
San Clemente, California 92672

STATE OFFICIAL

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Radiological Health Section
California Department of Health
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Sacramento, California 95814

State of California Office of Intergovernmental Management (1)
1400 10th Street
Sacramento, California 95814

CLEARINGHOUSES

Office of the Governor (10)
Office of Planning and Research
1400 Tenth Street
Sacramento, California 95814

San Diego County Comprehensive Planning Organization (1)
Security Pacific Plaza
1200 Third Avenue
San Diego, California 92101

Librarian/Thermal Reactors Safety Group (1)
Building 130
Brookhaven National Laboratory
Upton, L.I., New York 11973

Atomic Industrial Forum (1)
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

If there are any questions regarding this distribution list, contact the Environmental Licensing Assistant, Division of Site Safety and Environmental Analysis, 301-443-6980.

Title 10—Energy

CHAPTER I—NUCLEAR REGULATORY COMMISSION

CONSTRUCTION PERMIT OR OPERATING LICENSE

Initial Treatment of Application

On September 25, 1974, the Atomic Energy Commission published in the FEDERAL REGISTER (39 FR 34423) proposed amendments of 10 CFR Parts 2, 50, and 51 which were procedural changes pertaining to the initial treatment of an application for a construction permit or facility operating license. Under the proposed procedure a tendered application would be initially reviewed by the staff for completeness. If the application is determined to be complete and acceptable for processing the applicant would be so informed and requested to (a) submit additional copies of the application and environmental report and (b) make direct distribution of additional copies of the documents to Federal, state and local officials.

In accordance with the Energy Reorganization Act of 1974, Pub. L. 93-438, the Nuclear Regulatory Commission which was established January 19, 1975, assumed the licensing and related regulatory functions of the former Atomic Energy Commission.

After consideration of the comments received and other factors involved, the Nuclear Regulatory Commission has adopted the proposed amendments. The text of the rule set forth below is the same as the text of the proposed rule except for the following:

(a) Proposed § 2.101(a)(3)(iii) would have required the applicant to make direct distribution of additional copies of the application and environmental report to Federal and State officials, and other interested persons in accordance with written instructions furnished to the applicant by the staff. A sentence has been added to § 2.101(a)(3)(iii) that "Such written instructions will be furnished as soon as practicable after all or any part of the application, or environmental report, is tendered".

(b) Paragraph 2.101(a)(3)(iii) would have required that the copies of the application and environmental report submitted to the staff and distributed by the applicant be completely assembled documents, identified by docket number. Language has been added that "Subsequently distributed amendments to applications, however, may include revised pages to previous submittals and, in such cases, the recipients will be responsible for inserting the revised pages".

(c) A sentence has been added also to § 2.101(a)(4) that "Distribution of the additional copies shall be deemed to be complete as of the time the copies are deposited in the mail or with a carrier prepaid for delivery to the designated addressees".

(d) Changes to § 2.101(a)(5) have been made to conform with the amendments of § 2.101(a) published on September 25, 1974 (39 FR 34394) to allow applicants to submit the information required by Part 50 in three parts.

(e) Paragraph 50.30(c)(1)(i) has been changed to specify that 30 copies of the safety analysis report and 10 copies of the general information shall be retained by the applicant for direct distribution, or submitted upon request, in accordance with instructions by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. The proposed rule did not indicate the number of copies to be retained for this purpose. Additional copies may be required for applications having a unique design or with unusual or multiple sites.

(f) Section 51.40 currently requires that applicants covered by § 51.5(a) submit a total of 200 copies of the environmental report. This number has been reduced to a total of 150 copies. Paragraph 51.40(b) requires that applicants for a license to construct and operate a production or utilization facility (including amendments to such applications) shall submit 41 copies of the environmental report and retain an additional 109 copies to be submitted upon request or distributed in accordance with written instructions issued by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. The number of copies of the environmental report to be submitted with a petition for rule making has been reduced from 80 to 50 copies. Conforming amendments have been made to §§ 51.20(f) and 51.21.

One commenter noted the significant cost, handling, and storage problems involved when dealing with page copy of safety analysis reports and environmental reports, and suggested that the Commission change its requirements to permit most of the required copies of reports to be submitted in microform. The staff has underway a study to determine the feasibility of adopting a computerized automatic retrieval system using microform, and this suggestion will be considered in the conduct of that study.

Noting that § 2.101(a)(5) provided that docketing can be accomplished if one part of the application is complete, a commenter questioned whether the procedure for direct distribution would apply where one part of the application would be complete. It is the intent of the rule that the provision for direct distribution apply to each part of the application which is complete.

It was also suggested that copies be made available on a purchase basis to the interested individuals concerned. The copies to be distributed in accordance with instructions by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards are limited to Federal, State, and local officials and the public through the Technical Information Center, and it would not be appropriate to charge for such copies.

A number of commenters objected to the revised procedure as an unwarranted shift of the administrative support function from the staff to the applicant.

The revised procedure would result in some savings to the Commission and some additional costs to the applicants. Aside from these considerations, it is the Commission's view that the revised procedure is a step in the right direction of removing the NRC from the business of serving as a distribution center for applicants' documents. Further, the revised procedure is more efficient than the present procedure since the majority of copies of applications and amendments received by the NRC are repackaged and distributed outside the NRC. Direct distribution by the applicant of the additional copies of the application and environmental report would result in recipients outside the NRC receiving the documents from 8 to 10 days earlier than under the present procedure.

One commenter expressed the view that a tendered application should be formally docketed at the time the staff determines it is complete and acceptable. The Commission considers, however, that the application should not be formally docketed until the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, has received the required copies of the application and environmental report since a full review of these documents by the technical staff cannot begin until the required number of copies are received.

The amendments set forth below amend Parts 2 and 50 with respect to the initial treatment of an application for a construction permit, or operating license, for a production or utilization facility, or an application for amendment of a construction permit or operating license. If it is determined that the tendered application, including any environmental report required by Part 51 of the Commission's regulations, is complete and acceptable for processing, the applicant will be informed of this determination and requested to (a) submit to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, additional copies of the application and environmental report and (b) make direct distribution of additional copies of the documents to Federal, State, and local officials in accordance with requirements of the Commission's regulations and written instructions furnished by the staff.

The application and environmental report will be formally docketed upon receipt by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards of the re-

quired copies of the application and environmental report. Within ten (10) days after docketing the applicant must provide an affidavit that distribution of the additional copies to Federal, State and local officials has been completed in accordance with regulatory requirements and instructions by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards. Distribution of the additional copies of the application and environmental report shall be deemed to be complete as of the time the copies are deposited in the mail or with a carrier prepaid for delivery to the designated addressees.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, and sections 552 and 553 of Title 5 of the United States Code, as amended, the following amendments of Title 10, Chapter I, Code of Federal Regulations, Parts 2, 50, and 51 are published as a document subject to codification.

PART 2—RULES OF PRACTICE

1. Section 2.101 is revised to read as follows:

§ 2.101 **Filing of application.**

(a)(1) An application for a license or an amendment to a license shall be filed with the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as prescribed by the applicable provisions of this chapter. A prospective applicant may confer informally with the staff prior to the filing of an application.

(2) Each application for a license for a facility, or for receipt of waste radioactive material from other persons for the purpose of commercial disposal by the waste disposal licensee, will be assigned a docket number. However, to allow a determination as to whether an application for a construction permit or operating license for a production or utilization facility is complete and acceptable for docketing, it will be initially treated as a tendered application after it is received and a copy of the tendered application will be available for public inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. Generally, that determination will be made within a period of thirty (30) days.

(3) If the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, determines that a tendered application for a construction permit or operating license for a production or utilization facility, and/or any environmental report required pursuant to Part 51 of this chapter, or part thereof as provided in paragraph (a)(5) of this section, are complete and acceptable for docketing, a docket number will be assigned to the application or part thereof, and the applicant will be notified of the determination. With respect to the tendered application and/or environmental report or part thereof that is acceptable for docketing, the applicant will be requested to (i) submit to the Director of Nuclear Reactor Regulation or Director of Nuclear Material safety and Safeguards, as appropriate, such additional copies as the regulations in Parts 50 and 51 require; (ii) serve a copy on the chief executive of the municipality in which the facility is to be located or, if the facility is not to be located within a municipality, on the chief executive of the county; and (iii) make direct distribution of additional copies to Federal, State, and local officials in accordance with the requirements of this chapter and written instructions furnished to the applicant by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. Such written instructions will be furnished as soon as practicable after all or any part of the application, or environmental report, is tendered. The copies submitted to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, and distributed by the applicant shall be completely assembled documents, identified by docket number. Subsequently distributed amendments to applications, however, may include revised pages to previous submittals and, in such cases, the recipients will be responsible for inserting the revised pages.

(4) The tendered application for a construction permit or operating license for a production or utilization facility will be formally docketed upon receipt by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, of the required additional copies. Distribution of the additional copies shall be deemed to be complete as of the time the copies are deposited in the mail or with a carrier prepaid for delivery to the designated addressees. The date of docketing shall be the date when the required copies are received by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. Within ten (10) days after docketing the applicant shall submit to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, an affidavit that distribution of the additional copies to Federal, State, and local officials has been completed in accordance with requirements of this chapter and written instructions furnished to the applicant by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. Amendments to the application and environmental report shall be filed and distributed and an affidavit shall be furnished to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, in the same manner as for the initial application and environmental report. If it is determined that all or any part of the tendered application and/or environmental report is incomplete and therefore not acceptable for processing, the applicant will be informed of this determination, and the respects in which the document is deficient.

(5) An applicant for a construction permit for a nuclear power reactor subject to § 51.5(a) of this chapter may submit the information required by applicants by Part 50 of this chapter in three parts. One part shall be accompanied by the information required by § 50.30(f) of this chapter, another part shall include any information required by §§ 50.34(a) and 50.34a of this chapter and a third part shall include any information required by § 50.33a. One part may precede or follow other parts by no longer than six (6) months except that the part including information required by § 50.33a shall be submitted in accordance with time periods specified in § 50.33a. If it is determined that any one of the parts as described above is incomplete and not acceptable for processing, the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, will inform the applicant of this determination and the respects in which the document is deficient. Such a determination of completeness will generally be made within a period of thirty (30) days. Except for the part including information required by § 50.33a, whichever part is filed first shall also include the fee required by §§ 50.30(e) and 170.21 of this chapter and the information required by §§ 50.33, 50.34(a)(1), and 50.37 of this chapter. The Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, will accept for docketing an application for a construction permit for a nuclear power reactor subject to § 51.5(a) of this chapter where one part of the application as described above is complete and conforms to the requirements of Part 50 of this chapter. Additional parts will be docketed upon a determination by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, that they are complete.

(b) After the application has been docketed, each applicant for a license for receipt of waste radioactive material from other persons for the purpose of commercial disposal by the waste disposal licensee shall serve a copy of the application and environmental report, as appropriate, on the chief executive of the municipality in which the activity is to be conducted or, if the activity is not to be conducted within a municipality, on the chief executive of the county.

(c) The notice published in the FEDERAL REGISTER announcing docketing of the antitrust information part of the application for a facility license under section 103 of the Act, except for those applications described in § 2.102(d)(2), will state that:

(1) The portion of the application filed contains the information requested by the Attorney General for the purpose of an antitrust review of the application as set forth in Appendix L to Part 50 of this chapter;

(2) Upon receipt and acceptance for docketing of the remaining portions of the application dealing with radiological health and safety and environmental

matters, notices of receipt will be published in the FEDERAL REGISTER including an appropriate notice of hearing; and

(3) Any person who wishes to have his views on the antitrust matters of the application presented to the Attorney General for consideration should submit such views within sixty (60) days after publication of the notice announcing receipt of the antitrust information to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Chief, Antitrust and Indemnity Group.

(d) The Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, will give notice of the docketing of the public health and safety, common defense and security, and environmental parts of an application for a license for a facility, or for receipt of waste radioactive material from other persons for the purpose of commercial disposal by the waste disposal licensee, to the Governor or other appropriate official of the State in which the facility is to be located or the activity is to be conducted, and will cause to be published in the FEDERAL REGISTER a notice of docketing of the application which states the purpose of the application and specifies the location at which the proposed activity would be conducted.

(e) The notice published in the FEDERAL REGISTER of docketing of the application for a facility operating license under section 104b of the Act will, when appropriate, also state that any person who intervened or sought, by timely written notice to the Commission or the Atomic Energy Commission, to intervene in the construction permit proceeding for the facility to obtain a determination of antitrust considerations or to advance a jurisdictional basis for such determination may, within twenty-five (25) days after the date of publication, submit a written petition for leave to intervene and a request for a hearing on the antitrust aspects of the application.

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

2. Paragraphs 50.30(a), 50.30(b), 50.30(c)(1), and 50.30(c)(3) are amended to read as follows:

§ 50.30 Filing of applications for licenses; oath or affirmation.

(a) *Place of filing.* Each application for a license, including where appropriate a construction permit, or amendment thereof, and each amendment of such application, and correspondence, reports, or other written communications from the applicant to the Commission pertaining to such application, for a nuclear reactor, testing facility or other utilization facility, should be filed with the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Each application for a license, including where appropriate a construction permit, or amendment thereof, and each amendment of such application, and correspondence, reports, or other written communications from the applicant to the Commission pertaining to such application, for a fuel reprocessing plant or other production facility, should be filed with the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Communications, reports, correspondence, and applications may be delivered in person at the Commission's offices at 1717 H Street NW., Washington, D.C. or at 7920 Norfolk Avenue, Bethesda, Maryland.

(b) *Oath of affirmation.* Each application for a license, including whenever appropriate a construction permit, or amendment thereof, and each amendment of such application should be executed in three signed originals by the applicant or duly authorized officer thereof under oath or affirmation.

(c) *Number of copies of application.* (1) Each filing of an application for a license to construct and operate a production or utilization facility (including amendments to such applications) should include three signed originals and the following number of copies:

(i) For an application for a license for a facility described in § 50.21(b) or § 50.22, or a testing facility: Fifteen (15) copies of that portion of the application containing the information required by §§ 50.33 and 50.37 (general information) and forty (40) copies of that portion of the application containing any of the information required by §§ 50.34 and 50.34a (safety analysis report); an additional ten (10) copies of the general information and thirty (30) copies of the safety analysis report or part thereof shall be retained by the written instructions of the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate. The Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards may request additional copies of applications and the safety analysis report where the design is of a unique nature or for applications submitted pursuant to 10 CFR Part 50, Appendices M, N, and O.

(ii) For an application for an amendment to a license for a facility described in § 50.21(b) or § 50.22, or a testing facility: Nineteen (19) copies of that portion of the application containing the information required by § 50.33 (general information) and 40 copies of that portion of the application containing the information required by §§ 50.34 and 50.34a (safety analysis report);

(iii) For an application for a license for any other facility, or an amendment to a license for such facility: Nineteen (19) copies of that portion of the application containing the information required by §§ 50.33 and 50.37 (general information) and that portion of the application containing the information required by §§ 50.34 and 50.34a (safety analysis report);

(iv) For an application for a license for a production or utilization facility: Forty-one (41) copies of any applicant's environmental report required by Part 51 of this chapter.

* * * * *

(3) The copies required by paragraphs (b) and (c) (1) and (2) of this section need not be filed until that part of the application has been assigned a docket number or docketed pursuant to § 2.101(a) of this chapter. The following number of copies shall be filed to enable the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, to determine whether the application is sufficiently complete to permit the assignment of a docket number or docketing as appropriate.

(i) Fifteen (15) copies of that portion of the application containing any of the information required by §§ 50.34 and 50.34a (safety analysis report);

(ii) Ten (10) copies of that portion of the application containing the general information required by § 50.33; and

(iii) Twenty (20) copies of any environmental report required by Part 51 of this chapter.

PART 51—LICENSING AND REGULATORY POLICY AND PROCEDURES FOR ENVIRONMENTAL PROTECTION

3. Paragraph 51.20(f) is amended to read as follows:

§ 51.20 Applicant's Environmental Report—Construction Permit Stage.

* * * * *

(f) Number of copies. Each applicant for a permit to construct a production or utilization facility covered by § 51.5(a) shall submit the number of copies, as specified in § 51.40, of the Environmental Report required by § 51.5(a).

§ 51.21 [Amended]

4. Section 51.21 is amended by deleting the words "shall submit with its application two hundred (200) copies of a separate document" and substituting therefor "shall submit with its application the number of copies, as specified in § 51.40, of a separate document."

5. Section 51.40 is amended to read as follows:

§ 51.40 Environmental reports.

(a) Except as provided in paragraph (b) of this section, applicants for permits, licenses, and orders, and amendments thereto and renewals thereof, covered by § 51.5(a) shall submit to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, 150 copies of an environmental report which discusses the matters described in § 51.20. Petitioners for rule making covered by § 51.5(a) shall submit to the Director of Standards Development fifty (50) copies of an environmental report which discusses the matters described in § 51.20.

(b) Applicants for a license to construct and operate a production or utilization facility (including amendments to such applications) covered by § 51.5(a) shall submit to the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate, in accordance with § 50.30(c)(1)(iv) of Part 50 of this chapter, forty-one (41) copies of an environmental report which discusses the matters de-

scribed in § 51.20. The applicant shall retain an additional 109 copies of the environmental report for distribution to Federal, State, and local officials in accordance with written instructions issued by the Director of Nuclear Reactor Regulation or Director of Nuclear Material Safety and Safeguards, as appropriate.

Effective date. These amendments become effective on May 17, 1976.

(Sec. 161, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); Secs 201, 301, Pub. L. 93-438, 88 Stat. 1242, 88 Stat. 1248, (42 U.S.C. 5841, 5871)).

Dated at Washington, D.C. this 8th day of April 1976.

For the Nuclear Regulatory Commission.

SAMUEL J. CHILK,
Secretary of the Commission.

[FR Doc.76-10910 Filed 4-14-76;8:45 am]

rules and regulations

Title 10—Energy

CHAPTER I—NUCLEAR REGULATORY COMMISSION

CONSTRUCTION PERMIT OR OPERATING LICENSE

Initial Treatment of Application

Correction

In FR Doc. 76-10910 appearing at page 15832 in the FEDERAL REGISTER of Thursday, April 15, 1976 make the following corrections:

1. On page 15833 in the second column, third line from the top the word "safety" should be capitalized in § 2.101 (a)(3).
2. On page 15833, in the third column, fourth line from the top the word "by" should be "of" in § 2.101(a)(5).
3. On page 15834, second column, first line of § 50.30(b) the word "of" should be "or".
4. On page 15834, second column, the fourteenth line of § 50.30(c)(1) should read as follows: "shall be retained by the applicant for distribution in accordance with the written instruc-".

Central Files

APR 23 1976

Docket Nos: P-499, P-558, P-564
50-206, 50-361, 50-362
50-275, 50-312, 50-133
50-323

Energy Resources Conservation and
Development Commission
ATTN: Ms. Peggy Dole, Librarian
1111 Howe Avenue
Sacramento, California 95825

Dear Ms. Dole:

In your letter of March 15, 1976 to the Nuclear Regulatory Commission (NRC), and during subsequent telephone conversations with Ms. Brenda Scott of our staff, you requested amendments to the original Safety Analysis Reports and Early Site Review Reports which you had not received for all nuclear power plants in California.

When an applicant submits these reports for NRC review, they are required to submit enough copies for the use of our technical staff and to fill requests such as yours. In addition, we also send these reports to participating local, state and federal agencies.

Nuclear power plants located in California have attracted considerable citizen interest and participation in our licensing activities. Consequently, in addition to sending these reports to the above-mentioned individuals, we have also had numerous requests from the public in California for these reports. I am sorry to inform you that because of the volume of these individual requests, we have exhausted our supply of much of the material you requested and can only fill your request partially. However, the missing materials may be requested directly from the applicant. Enclosure 1 is a list of applicant addresses and material to be requested from them.

We are partially filling your request by forwarding the following material:

1. Department of Water and Power-City of Los Angeles
   San Joaquin Nuclear Project
   Early Site Review Report Amendment Nos. 7, 8, 9, 10, 14, 15, 16 & 17

APR 23 1976

DISTRIBUTION:
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NRC PDR
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HDenton
RHeineman
VStello
RBoyd
JLee
MRushbrook
SSheppard
RIngram
MGroff (NRR-808)
EHughes
BScott Rdg.

2. Southern California Edison Company
San Onofre Nuclear Station, Unit 1
Final Safety Analysis Report Amendment Nos. 47, 49, 50 & 51

3. Southern California Edison Company
San Onofre Nuclear Station, Units 2 and 3
Final Safety Analysis Report Amendment No. 22

4. San Diego Gas & Electric Company
Sundesert Nuclear Plant
Early Site Review Report Amendment Nos. 3 through 9

On April 4, 1976, at your request, Ms. Scott forwarded Amendment Nos. 24-41 to the Preliminary Safety Analysis Report for Pacific Gas & Electric Company's (PG&E) Diablo Canyon Nuclear Power Plant, Units 1 and 2. PG&E also plans to construct and operate a two-unit nuclear power station in Stanislaus County, California. A copy of a related notice is enclosed for your information. You have been added to our distribution list to receive the Preliminary Safety Analysis Report for the two-unit nuclear power station in Stanislaus County, California, when issued by the applicant, and for future amendments to all plants located in California.

I hope these materials and information satisfy your needs.

Original Signed by
Herbert N. Berkow

Herbert N. Berkow
Program Assistant to Director
Division of Project Management

Enclosures:
1. Material To Be Requested From Applicants
2. 1 Thru 4 Listed Above

OFFICE ➤	DPM	DPM					
SURNAME ➤	BScott:no	HBerkow					
DATE ➤	4/ 22 /76	4/ 22 /76					

Enclosure 1

Material To Be Requested From Applicants

Applicant:	Department of Water & Power-City of Los Angeles
Plant:	San Joaquin Nuclear Project
Docket No.:	P-499
Address:	Department of Water and Power (213)481-4670 City of Los Angeles ATTN: Mr. Robert C. Burt Nuclear Project Manager P.O. Box 111 Los Angeles, California 90051
Requested Material:	Early Site Review Report Amendment Nos. 11, 12, 13 & 18

Applicant:	Southern California Edison Company
Plant:	San Onofre Nuclear Stations, Units 1, 2 and 3
Docket Nos.:	50-206, 50-361 and 50-362
Address:	Southern California Edison Company (213)572-2292 ATTN: Mr. Jack B. Moore, Vice President 2244 Walnut Grove Avenue P.O. Box 800 Rosemead, California 91770
Requested Material:	San Onofre Nuclear Station, Unit 1 Final Safety Analysis Report Amendment Nos. 48 & 52 San Onofre Nuclear Station, Units 2 and 3 Final Safety Analysis Report Amendment No. 23

Enclosure 1

Applicant: San Diego Gas and Electric Company (714)232-4252

Plant: Sundesert Nuclear Project

Docket No.: P-558

Address: San Diego Gas and Electric Company
ATTN: Mr. J. E. Thomas, Vice President
Power Plant Engineering & Construction
P.O. Box 1831
San Diego, California 92112

Requested Material: 5 Volume Early Site Review Report and all related amendments

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Applicant: Pacific Gas and Electric Company (415)781-4211 X2237

Plant: Humboldt Bay

Docket No.: 50-133

Address: Pacific Gas and Electric Company
ATTN: Mr. John Morrissey
Vice President & General Counsel
77 Beale Street
San Francisco, California 94106

Requested Material: Complete Final Safety Analysis Report and all related amendments

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Applicant: Sacramento Municipal Utility District (916)452-3211 X537

Plant: Rancho Seco Nuclear Generating Station

Docket No.: 50-312

Address: Sacramento Municipal Utility District
ATTN: Mr. E. K. Davis, General Manager
6201 S Street
P.O. Box 15830
Sacramento, California 95813

Requested Material: Complete Final Safety Analysis Report and all related amendments

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. P-564-A

PACIFIC GAS AND ELECTRIC COMPANY

NOTICE OF RECEIPT OF PARTIAL APPLICATION FOR CONSTRUCTION PERMITS AND FACILITY LICENSE: TIME FOR SUBMISSION OF VIEWS ON ANTITRUST MATTERS

Pacific Gas and Electric Company (the applicant), pursuant to Section 103 of the Atomic Energy Act of 1954, as amended, has filed one part of an application, dated August 14, 1975, in connection with their plans to construct and operate two reactors in Stanislaus County, California. The portion of the application filed contains the information requested by the Attorney General for the purpose of an antitrust review of the application as set forth in 10 CFR Part 50, Appendix L.

The remaining portions of the application consisting of an Environmental Report and the Preliminary Safety Analysis Report (PSAR) pursuant to § 2.101 of Part 2, are expected to be filed in September 1976 and April 1977, respectively. Upon receipt of the remaining portions of the application dealing with radiological health and safety and environmental matters, separate notices of receipt will be published by the Commission including an appropriate notice of hearing.

A copy of the partial application will be available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., 20555, and at the Local Public Document Room, Stanislaus County Free Library, 1500 I Street, Modesto, California 95345. Docket No. P-564-A has been assigned to the application and it should be referenced in any correspondence relating to it.

Information in connection with the antitrust review of the application can be obtained by writing to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Chief, Office of Antitrust and Indemnity, Office of Nuclear Reactor Regulation.

MAR 3 1 1976

Docket Nos. 50-361
and 50-362

Distribution
Docket File
NRC PDR
Local PDR
LWR #3 File
JCollins
EIGoulbourne (2)
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TIC
FJWilliams
ELD
IE (3)
JRBuchanan, ORNL
TBAbernathy, DTI
ACRS
HRood
PKreutzer, EP

Southern California Edison Company
ATTN: Mr. Jack E. Thomas
Vice President - Electric
101 Ash Street
P. O. Box 1831
San Diego, California 92112

San Diego Gas & Electric Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Gentlemen:

RE: San Onofre, Units 2 and 3

In order to prepare our testimony for the upcoming hearing, we require the additional information described in Enclosure 1 by April 14, 1976. If you have any questions regarding the requested information, please contact Mr. H. Rood of our staff.

Sincerely,

Original Signed by
O. D. Parr
Olan D. Parr, Chief
Light Water Reactors
Branch No. 3
Division of Project Management

Enclosure:
Request for additional
information

OFFICE➤	DPM:LWR #3	DPM:LWR #3				
SURNAME➤	HRood:mjf	ODParr				
DATE➤	3/ /76	3/ /76				

cc: Rollin E. Woodbury, General Counsel
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George Spiegel, Esq.
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Washington, D. C. 20036

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ENCLOSURE 1

REQUEST R ADDITIONAL INFORMATION - AS HEARING
SAN ONOFRE UNITS 2 & 3
DOCKET NO. 50-361/362

1. Provide the following joint frequency distributions of wind speed and direction by atmospheric stability class. Distributions should be constructed for each of the 10, 20, & 36m wind measurement levels of the onsite tower. Atmospheric stability must be based on vertical temperature difference.

A. The annual summary for February 1975 through January 1976;
B. Monthly summaries for the individual months of February 1975 through January 1976;
C. Monthly summaries for daylight hours (sunrise to sunset) only for the individual months of May 1975 through September 1975.

2. Provide a magnetic tape containing the hourly data obtained from the San Onofre meteorological tower between January 25, 1975, and January 24, 1976. Include a description of the formatting used on the tape.

3. Provide a comprehensive and fully documented report on the tracer dispersion studies which you performed at the San Onofre site. The following questions reflect information which should be included within the report so that we may independently review the overall tests.

A. <u>Test Layout</u>

Locate on a scaled topographic map the following:

(1) All tracer release points,
(2) All plant structures,
(3) The three meteorological towers, and
(4) All samplers.

Also include a vertical plane projection of all plant structures as seen from the sampler line (normal to the sampler line).

B. <u>Tracer Release</u>

(1) Describe the following concerning the tracer and release:

(a) The type of tracer used, and its chemical and physical properties;
(b) The release apparatus;
(c) The release rate and your monitoring of the release rate to guarantee a steady release;

(d) Your methods to verify the total amount of tracer release;
(e) The direction of release; and
(f) Your calibration procedures, their frequency, and the points calibrated for the release apparatus.

(2) Because SF6 is denser than air, describe how you assure that the gas was well mixed with the air to result in a neutrally buoyant release.

(3) Describe the locations & heights of the tracer releases, to include their proximity to structures. Also describe the size of any obstructions to the flow of the tracer from the release to the samplers.

(4) Discuss the background measurements of the tracer during the test period.

C. Sampling

(1) Describe the following concerning the sampling techniques:

(a) The sampling apparatus;
(b) The size, direction, and height of the sampling aperture;
(c) The sampling rate, and your monitoring of the rate to guarantee a steady rate;
(d) How you activated the samplers, and when, after the beginning of the tracer release, you activated the samplers;
(e) Your calibration procedures, their frequency, and the points calibrated for the sampling apparatus; and
(f) Your method of assuring that the sampler was not contaminated before activation for the test, and that the sample was not contaminated during or after the test.

(2) Discuss whether the sampling system is chemically or physically inert with respect to the tracer. Discuss whether any other objects between the release point and samplers interact with the tracer to reduce its concentration.

(3) Describe the positioning and your bases for the positioning of the samplers.

D. Meteorological Instrumentation & Data

(1) Describe the following concerning the meteorological instrumentation and data:

(a) The locations of the meteorological towers in use during the tests, and your bases for their siting;
(b) The instrumentation on these towers, to include their heights (both above ground and above sea levels), their specifications, and system accuracies; and
(c) Your calibration procedures of all instrumentation during the test period, and the points calibrated.

(2) Discuss whether the data from each tower represents the meteorological conditions on the beach.

E. Sample Analysis

(1) Describe the following concerning the analysis of the samples:

(a) The technique used to analyze the samples;
(b) The analyzing apparatus, to include its specifications, accuracy, sensitivity, and high and low thresholds of detection (discuss whether the sampler or detector could reach an upper-limiting saturation level);
(c) Your method of labeling, controlling, and handling the samples from the sampling period through the analysis period;
(d) The location of the sample analyzer; and
(e) Your calibration procedures for the analyzer, their frequency, and the points calibrated.

(2) Discuss whether the analysis was sensitive to other gases and how this affected the results.

F. Test Program Errors

List the parameters in the test program for which errors are possible. Describe how you quantify such errors and how these errors affect your results. Present confidence limits of your results.

G. Presentation of Results

For each tracer release, present the following information:

(1) Tracer related:

(a) Date and time of release (start and finish),
(b) Location and height of release,
(c) Release rate,
(d) Total weight of tracer released, and
(e) Background tracer concentration.

(2) Sampler related:

(a) Sampling time (start and finish);
(b) Sampling rate;
(c) Total weight of tracer sampled at each sampler;
(d) Volume of air sampled for each sampler; and
(e) Graphical representation of X/Q versus sampler (separately for each test, and compositely for each type of test, e.g. GLR from release point 1.)

(3) Meteorology related:

From all three towers and all levels on the towers, as applicable, a) for 15-minute averages during the tests, (i.e. four sets of averages for an hour test), and b) for the total sampling period average:

(a) Pasquill stability class;
(b) Vertical temperature difference, ΔT;
(c) Lower tower-level temperatures;
(d) Wind speed;
(e) Wind direction;
(f) Standard deviation of horizontal wind direction, σ_θ ; and
(g) The maximum range of horizontal wind direction.

H. Results and Conclusions

(1) Present the results of the tests and your conclusions. Substantiate your conclusions and describe the limitations imposed on them by the test methods and results.

(2) Discuss whether your conclusions are valid over all seasons of the year.

4. Describe any future testing which you may be considering and when such testing may occur. Describe whether such tests may be conducted under conditions different than those for the completed tests (such as for shorter sampling time periods, during unstable atmospheric conditions, during daytime hours).

5. Verify that the seaward ends of the 8-foot chain link fences on the beach within the exclusion area coincide with the mean high water line USGS 1929 datum as of 1-12-63 (as shown in Figure 1.8-C of PSAR Amendment 22). Indicate the reason(s) for not extending the fences to the current mean high water line, and for not placing a fence along the current mean high water line in front of the plants. Discuss the practicality of such additional fencing.

6. Provide estimates (and the basis of the estimates) of the number of persons that could occupy the approximate 5-acre southwest sector of the station site for the purpose of viewing the scenic bluffs and barrancas. Provide estimated capacity (and the basis of the estimate), predicted maximum use, and predicted average use.

7. Provide updates (including the sources of the data) of Tables 1, 2, 3 and 4 of SCE's letter of May 19, 1975, to the Commission, detailing beach use for San Onofre State Beach.

8. Provide all available beach and bottom profile data for the station site, on a monthly basis where possible. Include the monthly beach and bottom profile data from +10 MLLW to -4 MLLW which has been accumulated as part of the Unit 2 & 3 construction monitoring program, and the quarterly beach and bottom profile data accumulated from May 1964 through December 1967.

9. Page 1.8-2bzs of San Onofre Units 2 & 3 Amendment No. 22 indicates the predicted maximum number of persons in the reduced exclusion area based on an evaluation of the current use of San Onofre State Beach. Provide an estimate of the predicted maximum number of persons who could be within the exclusion area considering the data in "Resource Management Plan and General Development Plan for San Onofre State Beach," which was prepared in September 1972 and reprinted in October 1974. Drawing No. 13186 of this report shows the planned parking areas and camp sites proposed for the north and south sections of the coastline which are adjacent to the current plant exclusion area. Indicate the current and projected use of that section of the coast marked for the Enlisted Men's Club. Provide data on the present seasonal use of the beach areas abutting the revised exclusion area.

10. Provide a detailed map which clearly shows all roads and trails to the beach within a 2 mile radius of the site which will be available when the parks are completed.

11. Based on the dumping of spoil from the construction of Units 2 & 3, provide a map which shows the current mean high water line and the mean lower low water line. Indicate whether or not long term exposure to the waves will reduce the beach to its natural width, and provide supporting data including the experience with Unit 1 spoils area and its change with time.

12. Provide topographic maps of the beach area fronting Units 1, 2 and 3 for maximum-beach-width summer conditions. Indicate thereon the location of all temporary or permanent groins, sheet piling, etc. Also indicate thereon all bulkheads, access points to the beach, your property lines, mean sea level, and mean lower low water datums. If topographic maps cannot be provided in a timely manner, provide a map of relatively small scale that accurately locates the profiles discussed in the Marine Advisor's reports and those done for the Sand Disposal Study.

13. For all temporary or permanent groins, provide the date of installation and the schedule for removal if temporary.

let us know within seven (7) days after receipt of this letter the schedule for your response. Feel free to contact us if you have any comments or questions.

Sincerely,

Olan D. Parr, Chief
Light Water Reactors
Project Branch 1-3
Division of Reactor Licensing

Enclosure:
Request for Additional
Information

cc: See page 3

OFFICE ➤	RL:LWR 1-3	RL:LWR 1-3	OELD	RL:LWR 1-3		
SURNAME ➤	DTibbitts:mk	PDO'Reilly	D. Kastalia	ODParr		
DATE ➤	8/29/75	8/29/75	8/29/75	8/29/75		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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OFFICE ➤	RL:LWR 1-3	RL:LWR 1-3	OELD			
SURNAME ➤	PO'Reilly:mk	ODParr				
DATE ➤	8/ /75	8/ /75	8/ /75			

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

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RHeineman	LWR 1 Branch Chiefs
RWKlecker	JPanzarella
WHaass	ACRS (16)

AUG 2 9 1975

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

In your letter dated July 11, 1975, you proposed a new exclusion area for the San Onofre Nuclear Generating Station, Units 2 & 3. As a result of our review of this proposed exclusion area, we find that additional information is required. The specific information requested is contained in Enclosure No. 1.

Additional questions regarding accident doses and the adequacy of the proposed exclusion area distance may be necessary, depending on the results of our review of site meteorological data.

Since the matter of exclusion area control is still pending before the Atomic Safety and Licensing Appeal Board, we will need a complete response to the items in Enclosure No. 1 as soon as possible. Please

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

AUG 2 9 1975

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3
DOCKET NOS. 50-361 AND 50-362

1. On pages 1 and 2 of the July 11, 1975 proposal, you indicate that you are involved in negotiations with the U. S. Marine Corps for the purpose of obtaining an amendment to the existing site easement which would give you authority to determine all activites in a redefined exclusion area.
   (a) Submit a copy of the amended site easement as soon as it becomes available.
   (b) Submit, within 10 days of the date of our transmittal letter, a report on the status of your negotiations with the Marine Corps.

2. On page 2 of the proposal, you state that you will provide certain physical barriers and/or signs "designed and located so as to minimize recreational activities within the exclusion area while maintaining a means of passage between open beach areas upcoast and downcoast of the exclusion area...."
   (a) Submit a map showing the placement, as now conceived, of the physical barriers and signs.
   (b) Describe the planned physical barriers and signs. With respect to the signs, the description should include a statement of what the signs will say.

3. On page 2 of the proposal, you state that you will "provide for enforcement of the control over use of the landward portions of the exclusion area by security personnel such that only passageway transit is permitted."

AUG 29 1975

(a) As used in the quoted statement, does the term "landward portions of the exclusion area" include tidal beach below the mean high tide line?

(b) Describe in detail the proposed enforcement system and how it will operate.

(c) Based on the enforcement system, and on the planned recreational developments in the area, estimate the number and distribution of persons who could be in the exclusion area.

4. Page 3 of the proposal references a 5-acre viewing area in the southwest corner of the site. Provide a map showing clearly the details of this area, including walkways.

5. Page 6 of the July 11, 1975 proposal describes the narrow strand of beach below mean high tide.

(a) Provide a drawing of this beach area showing the seawall for Units 2 and 3 and the width of the stretch of beach below mean low and high tide after all plant construction and grading is completed.

(b) Based on past use of this beach, on the planned recreational developments in the area, and on the distance to future beach access routes, estimate the number of persons who could be in the section below the mean high tide after plant construction is completed.

(c) Indicate the feasibility and current frequency of vehicle access to this area.

(d) Describe in detail how the tide, surge and wave conditions change the width of the strand of beach below the mean high tide line as a function of time.

6. Update, to the extent necessary, all previously submitted information (including hearing testimony) relating to the type, location and expected level of use of recreational facilities in the vicinity of the site.

JUL 3 0 1975

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LWR 1 Branch Chiefs
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Docket Nos. 50-361
and 50-362

Southern California Edison Company
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Rosemead, California 91770

San Diego Gas & Electric Company
Attn: Mr. Martin B. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

As a result of our review of Amendment No. 21 to the San Onofre Nuclear Generating Station Units 2 and 3 PSAR, entitled, "Analysis of a Main Steam and Feedwater Piping System Break Outside the Containment," we find that additional information is required. The specific information requested is identified in the Enclosure. We have discussed this information previously with your representatives.

Amendment No. 21 represents only a partial response to our letter of December 18, 1972 requesting an analysis of high energy fluid piping system breaks outside the containment building. You have informed us that the analysis of breaks in the balance of the high energy piping systems outside of containment will be submitted in December 1976. As a result, our review and evaluation of your high energy line break analysis cannot be completed until you have submitted the analysis of breaks in the balance of the high energy piping systems.

It is our understanding that you intend to revise the portion of Amendment No. 21 that discusses design basis piping break criteria. You may provide the information requested in the Enclosure with the

appl 3/

OFFICE ➤						
SURNAME ➤						
DATE ➤						

JUL 3 0 1975

forthcoming revision to Amendment No. 21, or you may include your response in the submittal of the analyses of the balance of the high energy piping systems. Please inform us within seven days after receipt of this letter of your schedule for furnishing the requested information.

Sincerely,

Original Signed by
Olan Parr

Olan D. Parr, Chief
Light Water Reactors Project Branch 1-3
Division of Reactor Licensing

Enclosure: Request for Additional Information

OFFICE ➤	LWR 1-3	LWR 1-3				
SURNAME ➤	P.O'Reilly:	ODParr				
DATE ➤	6/ /75pam	6/ /75				

JUL 3 0 1975

cc: Rollin E. Woodbury, General Counsel
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George Spiegel, Esq.
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OFFICE ➤						
SURNAME ➤						
DATE ➤						

JUL 3 0 1975

SAN ONOFRE UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

REQUEST FOR ADDITIONAL

INFORMATION

1. Provide details of the design of the protective enclosures that will be provided at San Onofre Units 2 and 3 to protect safety-related electrical equipment from the effects of a possible rupture in a high energy fluid piping system. Include the details of any openings in these enclosures, such as water-tight doors or access hatches.

2. Openings such as water-tight doors or access hatches in the protective enclosures represent potential means of steam-air or jet impingement ingress into the enclosures. If these openings are left open inadvertently, in the event of a high energy line break, safety-related electrical equipment could be subjected to an environment for which it has not been qualified. Discuss the measures that you have taken in the San Onofre Units 2 and 3 design to alert the operator that doors or hatches in the protective enclosures are open. Indicate the safety classification of any associated circuitry and the degree of conformance of such circuitry with Regulatory Guides 1.47 and 1.75.

3. Verify that the main steam isolation valves can close against and withstand backpressure and blowdown forces and prevent both steam generators from blowing down.

4. According to Appendix D of Amendment 21, the essential equipment in the main steam and feedwater isolation valve enclosures will be designed to withstand the environment associated with a high energy line break. Discuss the means, such as prototype testing, that will be used to verify that essential equipment has been designed to withstand the environment associated with a high energy line break.

Docket

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ACRS (16)
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S. Kari, L:RP
H. Berkow
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S. Hanauer, DRTA
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V. Moore, LAW:RP
D. Eisenhut
R. Klecker

FEB 21 1974

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Gentlemen:

In a Commission Memorandum and Order dated December 7, 1973, concerning the LaSalle County Nuclear Station, Units 1 and 2, the Regulatory staff was directed to determine, for facilities under construction and for construction applications under review, if quality assurance personnel have sufficient authority and organizational freedom to perform their critical functions effectively and without reservation.

To make this determination, we require a full and complete response to the following information requests as applicable to your organization and to the organizations of your principal contractors (for example, A/E, NSSS, Constructor). Your response should be provided by March 18, 1974.

1. Identify and chart the organizational structure of individuals or groups performing QA related activities (checking, auditing, inspecting, or verifying that an activity has been correctly performed) in design, procurement, manufacturing, construction and testing. Describe their responsibility, authority (including stop work authority), and primary duties. Indicate from whom technical QA direction is received and from whom administrative control (salary review, hire/fire, position assignment) is exercised. Include communications and reporting paths between field QA personnel and corporate management.

2. Describe the authority, organizational freedom, and independence of personnel, performing QA functions, to identify quality problems; to initiate, recommend or provide solutions; and to verify implementation of solutions.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

If this information is already contained in the hearing record or in your application, specify the page reference containing the information on which you rely for a response to this letter.

Please contact us if you have any questions regarding the information requested.

The information should be filed as an amendment to your application.

Sincerely,

Karl R. Goller /for

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Directorate of Licensing

cc: Mr. Rollin E. Woodbury
Vice President and General Counsel
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

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General Counsel
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*See previous yellow for concurrence. Second page retyped.

OFFICE ➤	L:RP	L:TR:QA	LJCHANDLER OGC	KRG/for LWR:RP	LWR 1-3 PHD P.D.C.
SURNAME ➤	SKari:no	RVollmer	[illegible]	RCDeYoung	
DATE ➤	2/ /74	2/ /74	2/20/74	2/21/74	2/20/74

Docket Nos. 50-361
and 50-362

OCT 2 1973

Southern California Edison Company
ATTN: Mr. Jack B. Moore
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Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

In its July 21, 1972 Report on San Onofre Nuclear Generating Station Units 2 and 3, and in its August 17, 1972 Report on Forked River Nuclear Generating Station Unit 1, the Advisory Committee on Reactor Safeguards (ACRS) noted that the applicant had agreed to design the ECCS for these facilities in accordance with the results of studies similar to those conducted by Combustion Engineering for the ANO-2 (Arkansas Nuclear One - Unit 2) facility and stated, "The final design should be reviewed by the Regulatory staff and the ACRS prior to fabrication and installation of major components".

In its January 17, 1973 Report on Waterford Steam Electric Station Unit No. 3 the ACRS noted that the applicant had described flexibility in design which can be used to improve ECCS effectiveness and stated, "The Committee believes it important that improvements in ECCS effectiveness be included in Waterford Unit No. 3 and recommends that the final design of the ECCS be reviewed by the Regulatory staff and the ACRS prior to fabrication and installation of major components".

By a letter dated July 24, 1973, we requested the ACRS to advise us whether it still believes that ECCS designs for these facilities should be reviewed by the Regulatory staff and the ACRS prior to fabrication and installation of major components. The ACRS replied on September 11, 1973 in a letter to L. M. Muntzing. A copy of each of these letters

LB

OFFICE ▶						
SURNAME ▶						
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OCT 2 1973

is enclosed. The ACRS stated that it had decided that these three facilities are sufficiently similar to warrant review of the ECCS on a generic basis, rather than as individual cases and recommended that this generic review be made at an appropriate time, but prior to the operating license stage.

In view of the changes in circumstances since the ACRS completed its review of your application for a construction permit and in accordance with the comments of the ACRS in its letter of September 11, 1973, you may, at your discretion and risk, proceed with fabrication and installation of ECCS components without prior further review by the Regulatory staff or the ACRS.

We are requesting Combustion Engineering to provide us with a schedule for the completion of generic studies applicable to the final ECCS designs that will be installed in these plants and to submit the results of these studies to us at least six months prior to receipt of the FSAR for an operating license for any of these plants. We are also requesting that Combustion Engineering use ECCS evaluation models modified in accordance with changes in the Regulations that are expected in the next several months as a result of the ECCS rulemaking proceedings. A copy of our letter to Combustion Engineering is enclosed.

The individual operating license reviews of each of the final ECCS designs will be based in part on our review of the Combustion Engineering generic studies, which may not be completed prior to the fabrication and installation of major components of the ECCS in your facility. Therefore, to reduce the probability that your final ECCS design will be found unacceptable during our operating license review, you should consider the incorporation of ECCS improvements in your final design as they are identified by the CE generic studies prior to Regulatory staff and ACRS review of these studies.

If further information is needed, please contact us.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

OFFICE ▶	L:PWR-3 x7415	L:PWR-3	L:DD/TR	L:AD/PWRs		
SURNAME ▶	RABirkel:esp	KRColler	JMHendrie	RCDeYoung		
DATE ▶	9/ /73	9/ /73	9/ /73	9/ /73		

OCT 2 1973

Enclosures:
1. Letter to ACRS dtd. 7/24/73
2. Letter from ACRS dtd. 9/11/73
3. Letter to CE

cc w/encls:
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Southern California Edison Company
2244 Walnut Grove Avenue
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Chickering & Gregory, General Counsel
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Chairman, Board of Supervisors
County of San Diego, California

Mayor, City of San Clemente
San Clemente, California 92672

Dr. David M. Heslep, Chief
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Department of Public Health
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~~Docket 50-363~~
Docket 50-362

OCT 2 1973

Combustion Engineering, Inc.
ATTN: Mr. F. M. Stern
Director, NSSS Projects
Combustion Division
Windsor, Connecticut 06095

Gentlemen:

In our Safety Evaluation Reports for Forked River Unit 1, San Onofre Units 2 and 3, and Waterford Unit 3 we stated that the review of the final designs of the ECCS would be completed prior to installation and fabrication of major components. As stated in letters to the applicants for these facilities on this date we now consider that a generic review of the design bases for the final design of the ECCS's for these facilities is appropriate because each of these facilities incorporate Combustion Engineering's 3410 MW NSSS design. Copies of our letters to these applicants are enclosed.

We therefore, request that you provide us with the results of design studies for your 3410 MW NSSS ECCS and other ECCS improvement studies that will serve as the basis for selection by the applicants of the final ECCS design for each of these plants. This information should be submitted as a Topical Report soon after modifications to the Combustion Engineering ECCS evaluation models are approved. It should include both the results of design parameter studies like those performed for the ANO-2 (Arkansas Nuclear One - Unit 2) facility and the results of Combustion Engineering's ongoing ECCS improvement studies. The information should be of such extent and in sufficient detail as to provide the bases for the final designs of the ECCS's for the above referenced facilities, even though some differences may exist among the final designs selected by the applicants. We believe that as a result of your extensive participation in our reviews of the construction permit applications for these facilities and in the preparation and review of the ANO-2 study, you already have a thorough understanding of our information needs.

In order that we may schedule our review please advise us of your schedule for submitting this information. This can be done relative to the approval date for modified ECCS evaluation models. We would expect, however, that all the information needed to complete our generic review would

be submitted to us at least six months prior to receipt of the FSAR for an operating license from any of the involved applicants.

If you should need further clarification, please contact us.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosures:
1. Letter to Jersey Central Power and Light Company
2. Letter to Louisiana Power and Light Company
3. Letter to Southern California Edison Company

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FOR PREVIOUS CONCURRENCES SEE ATTACHED YELLOW

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KRGoller
VHWilson (2)
ACRS (16)

Docket Nos. 50-361
and 50-362

JAN 2 2 1973

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

The enclosed Errata Sheet for "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" amends our letter to you of December 18, 1972. The letter concerned the potential effects of steam line breaks in the San Onofre Nuclear Generating Station, Units 2 and 3, respectively.

Please contact us if you desire discussion or clarification of this material.

Sincerely,

Original Signed By
O. D. Parr

Karl R. Goller, Chief
Pressurized Water Reactors
Branch No. 3
Directorate of Licensing

Enclosure:
Errata Sheet

cc w/encl:
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P. O. Box 800
Rosemead, California 91770

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DATE ▶	1/ /73	1/ /73				

JAN 22 1973

cc w/encls:
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ERRATA SHEET FOR "GENERAL INFORMATION REQUIRED FOR CONSIDERATION OF THE EFFECTS OF A PIPING SYSTEM BREAK OUTSIDE CONTAINMENT"

The following lists the changes that have evolved on our initial information request:

1. Page 2, Item 2--Insert the following in 2. to precede the existing first sentence:

   "Design basis break locations should be selected in accordance with the following pipe whip protection criteria; however, where pipes carrying high energy fluid are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width."

2. Page 2, Item 2(a)(2)--Change nomenclature to read "any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n ... "

3. Page 4, Item 2.(b)(2)--Change 0.9 $(S_h + S_A)$ to 0.8 $(S_h + S_A)$.

4. Page 6, Item 7 --Add "structural" to read "The structural design loads..."

5. Page 7, Item 11.(a)--Add "required" so as to read, "Loss of required redundancy..."

6. Page 7, Item 11.(a)--Delete "the steam line break" and replace with "that" to read "...the consequences of that accident..."

7. Page 8, Item 11.(b)-- Replace (b) with the following: (b) "Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted but a loss of function is not permitted. For such situations plant shutdown is required."

8. Page 8, Item 13--Change wording in the first sentence to read "Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy fluid line break."

Docket Nos. 50-361
and 50-362

DEC 1 8 1972

Southern California Edison Company
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Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

The Regulatory staff's continuing review of reactor power plant safety indicates that the consequences of postulated pipe failures outside of the containment structure, including the rupture of a main steam or feed-water line, need to be adequately documented and analyzed by licensees and applicants, and evaluated by the staff as soon as possible. Criterion No. 4 of the Commission's General Design Criteria, listed in Appendix A of 10 CFR Part 50 requires that:

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

Criteria of the previous version of the Commission's General Design Criteria also reflect the above requirements.

Thus, a nuclear plant should be designed so that the reactor can be shutdown and maintained in a safe shutdown condition in the event of a postulated rupture, outside containment, of a pipe containing a high energy fluid, including the double ended rupture of the largest pipe in the main steam

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and feedwater systems. Plant structures, systems, and components important to safety should be designed and located in the facility to accommodate the effects of such a postulated pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

Based on the information we presently have available to us on the San Onofre Station, Units 2 and 3, it is our understanding that the main steam lines pass from the containment structures over the top of the safety injection buildings, and enter the turbine building at a point adjacent to the auxiliary building. From this it appears that failure of the lines which are generally located outside the safety injection and auxiliary buildings could damage the walls or roofs of areas which may house vital equipment needed to bring the plant to a safe shutdown so that modification of the station design may be necessary.

We request that you provide us with analyses and other relevant information needed to determine the consequences of such an event, using the guidance provided in the enclosed general information request. The enclosure represents our basic information requirements for plants now being constructed or operating. You should determine the applicability for the San Onofre Station, Units 2 and 3 of the items listed in the enclosure.

If the results of your analyses indicate that changes in the design of structures, systems, or components are necessary to assure safe reactor shutdown in the event this postulated accident situation should occur, please provide information on your plans to revise the design of your facility to accommodate the postulated failures described above. Any design modifications proposed should include appropriate consideration of the guidelines and requests for information in the enclosure.

Please inform us within 7 days after receipt of this letter when we may expect to receive an amendment with your analysis of this postulated accident situation for the San Onofre Station, Units 2 and 3, and a description of any proposed modifications. Sixty copies of the amendment should be provided.

A copy of the Commission's press announcement on this matter is also enclosed for your information.

Sincerely,

Original Signed by
Roger S. Boyd

A. Giambusso, Deputy Director
~~for Reactor Projects~~
Directorate of Licensing

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DATE	12/ /72	12/ /72	12/ /72		12/ /72	12/ /72

Enclosures:
1. General Information Required
2. AEC Press Release

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General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:
   (a) Both of the following piping system conditions are met:
      (1) the service temperature is less than 200° F; and
      (2) the design pressure is 275 psig or less; or
   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or
   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or

(d) The internal energy level[1] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

2. The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2] breaks should be postulated to occur at the following locations in each piping run[3] or branch run:

(1) the terminal ends;

(2) any intermediate locations between terminal ends where the primary plus secondary stress intensities S_m (circumferential or longitudinal) derived on an elastically

---

[1]The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

[2]Piping is a pressure retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

[3]A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

calculated basis under the loadings associated with one - half safe shutdown earthquake and operational plant conditions[4] exceeds 2.0 S_m[5] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) any intermediate locations between terminal ends where the cumulative usage factor (U)[6] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) at intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) the terminal ends;

---

[4] Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

[5] S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

[6] U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.9 $(S_h + S_A)$[7] or the expansion stresses exceed 0.8 S_A; and

(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under 2 above should be equivalent to the following:

(a) Longitudinal[8] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

---

[7] S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

[8] Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

(b) Circumferential[9] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:

(a) The locations and number of design basis breaks on which the dynamic analyses are based.

(b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

(c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.

(d) Diagrams of mathematical models used for the dynamic analysis.

(e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structure, systems, or components important to safety, such as the control room.

---

[9] Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet and reactive forces including:

   (a) Pipe restraint design to prevent pipe whip impact;

   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;

   (c) Separation of redundant features;

   (d) Provisions to separate physically piping and other components of redundant features; and

   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:

   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;

   (b) The allowable design stresses and/or strains; and

   (c) The load factors and the load combinations.

7. The design loads, including the pressure and temperature transients, the dead, live and equipment loads; and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including nonseismic Category I structures, caused by the accident, will not cause failure of any other structure in a manner to adversely affect:
    (a) Mitigation of the consequences of the accidents; and
    (b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:
    (a) Loss of redundancy in any portion of the protection system (as defined in IEEE-279), Class IE electric system (as defined in IEEE-308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of the steam line break accident and place the reactor(s) in a cold shutdown condition; or

(b) Loss of the ability to cope with accidents due to ruptures of pipes other than a steam line, such as the rupture of pipes causing a steam or water leak too small to cause a reactor accident but large enough to cause electrical failure.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown of the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a steam line or feedwater line break. The information required for our review should include the following:

(a) Identification of all electrical equipment necessary to meet requirements of 11 above. The time after the accident in which they are required to operate should be given.

(b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

(c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingment from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

(d) An evaluation of the capability for safety related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. P-429
Contact: Frank Ingram
Tel. 301/973-7771

FOR IMMEDIATE RELEASE
(Wednesday, December 13, 1972)

AEC REGULATORY STAFF REQUESTS DATA ON PIPE BREAKS IN NUCLEAR PLANTS

The Atomic Energy Commission's Regulatory Staff is asking all utilities that operate nuclear power plants or have applied for operating licenses to assess the effects on essential auxiliary systems of a major break of the largest main steam or feedwater line. These lines carry steam from inside the reactor containment building to the main turbine in the turbine building, and hot feedwater back from the turbine condenser. The utility assessments will be evaluated by the AEC's Regulatory Staff.

The probability of a steam-line rupture is low. Nonetheless it will have to be considered in the AEC's safety evaluation.

The review of the pipe break problem has been under way for several weeks. It was started after the Advisory Committee on Reactor Safeguards received a letter raising questions about the location of pipes in the two-unit Prairie Island plant in Minnesota.

The Regulatory Staff has reviewed the Northern States Power Company application to operate Prairie Island, and on the basis of data available it has concluded that design changes will be required at Prairie Island.

Based on the new information--to be submitted by utilities as soon as possible--the Staff will determine what corrective action, if any, is necessary in each case. The changes could include such steps as relocating piping, providing venting of compartments, the addition of piping restraints, and, in some cases, structural strengthening.

#

See Central Files for concurrence
●y and distribution

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-361
and 50-362

November 20, 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
Rosemead, California 91770

San Diego Gas and Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

RE: San Onofre Nuclear Generating Station, Units 2 and 3

Gentlemen:

The Commission's Regulatory Staff has completed a review of fuel densification and its effect on reactor operation including transients and postulated loss-of-coolant accidents. The Staff's investigations and conclusions are reported in "Technical Report on Densification of Light Water Reactor Fuels" dated November 14, 1972, a copy of which is enclosed for your information and guidance. This report concludes that densification of fuel may occur and that the resulting formation of fuel column gaps should be anticipated in all light water reactor fuels. The report also provides the essential elements to be included in calculational models used to account for the effects of fuel densification.

The Regulatory Staff believes that the fuel for the subject facility(s) is susceptible to densification. Therefore, we request that you provide the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance provided in the enclosed report. If the analyses indicate that changes in design or operating conditions are necessary to maintain required margins, you should submit proposed changes and operating limitations with the analyses.

LB

In order that the Regulatory Staff can conduct an expeditious and orderly review of these matters, we request that you submit the analyses and additional information within 45 days from the date of this letter.

It is requested that this information be provided with one signed original and thirty-nine additional copies. If your submittal is for more than one unit, a total of sixty copies is needed.

Sincerely,

A Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
Technical Report on Densification
(November 14, 1972)

cc w/o encl:
Mr. Rollin E. Woodbury, Vice President
and General Counsel
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas and Electric Company
111 Sutter Street
San Francisco, California 94104

cc w/encl:
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San Clemente, California 92672

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L Reading

OCT 24 1972

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

Ten copies of the Safety Evaluation prepared by the Directorate of Licensing concerning your application for construction permits for the San Onofre Nuclear Generating Station, Units 2 and 3, are enclosed for your use.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Safety Evaluation

cc: See attached

bcc: H. J. McAlduff, ORO
T. W. Laughlin, DTIE
J. R. Buchanan, ORNL
N. Goodrich, ASLBP
A. S. Rosenthal, ASLAB

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE ▶	AD:PWR x7415:esp	L:PWR-3	L:PWR-3	L:AD/PWRs		
SURNAME ▶	EIGoulbourne	RABirkel	KRGoller	RCDeYoung		
DATE ▶	9/ /72	9/ /72	9/ /72	9/ /72		

cc: Mr. Rollin E. Woodbury, Vice President
and General Counsel
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket Nos. 50-361
and 50-362

OCT 5 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

Your application for licenses to construct and operate the proposed San Onofre Nuclear Generating Station, Units 2 and 3, indicates that the design pressure for the reactor containment building would be significantly in excess of the highest pressure calculated for any loss-of-coolant accident using conservative assumptions for the energy released to the containment. During our licensing review we indicated that we would perform an additional evaluation to verify that an adequate margin has been provided between the design pressure and pressures during loss-of-coolant accidents, assuming that all potentially significant means of transferring heat energy from the reactor core and cooling system to the containment are taken into account in a conservative manner.

During this ongoing review, we find that additional information is required to complete our evaluation. The specific information required is listed in the enclosure and should be provided in sufficient detail for the performance of independent analyses.

OFFICE ▶						LB
SURNAME ▶						
DATE ▶						

OCT 5 1972

Our tentative schedule is based on the assumption that this additional information will be available for our review by October 23, 1972. If you cannot meet this date, please inform us within 7 days after receipt of this letter so that we may revise our scheduling.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Additional Information Required

cc w/encl:
Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

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DFKnuth, L
RLTedesco, L
HRDenton, L
PWR Branch Chiefs
RWKlecker, L
OGC
RO (3)
VWilson, L
RABirkel, L

OFFICE ▶	L:PWR-3 x7415	L:PWR-3	L:AD/PWRs			
SURNAME ▶	RABirkel:esp	KRGoller	RCDeYoung			
DATE ▶	10/ /72	10/ /72	10/ /72			

ADDITIONAL INFORMATION REQUIRED

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

Provide the following information. All assumptions used in the analysis should be explained. Assumptions should be conservative with respect to the calculation of containment pressures.

1. Containment pressure-time response analyses should be provided for selected design basis loss-of-coolant accidents. Double-ended breaks of the largest reactor outlet pipe and double-ended breaks of the reactor coolant pump suction and discharge pipes should be included. Smaller pipe breaks should also be analyzed and should be selected to be representative of the spectrum of break sizes for both inlet and outlet reactor coolant pipes. The analyses should be extended, as a minimum, through the blowdown, reflood and post-reflood phases of the accidents (i.e., for about one hour following the accident).

2. The reflood model that is used following blowdown should be described in detail. The description should include the assumptions used to develop the model, e.g., hydraulic modeling of the primary coolant system, resistances of components (primary coolant pump, steam generator, piping and reactor core), and the methods used in computing steam generation in the core and other energy sources (core stored energy, decay heat [short and long term], thick and thin metal-stored energy, and steam generator-stored energy).

3. If the blowdown model differs from that described in the SAR for containment calculations, the differences should be discussed in detail.

4. For the cold leg break, the size and location resulting in the highest calculated containment pressure (analyzed in Item 1), tables of mass release (pound/sec), the enthalpy of the mass (BTU/pound) released from the core, and the mass and enthalpy released to the containment should be provided throughout the blowdown and reflood phases of the accident. A graph showing core inlet velocity as a function of time should also be provided for the reflood phase of the accident.

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RBirkel
OGC
Robert Brodsky, NR
VHWilson (2)

Docket Nos. 50-361
and 50-362

JUL 26 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

A copy of a letter to Chairman Schlesinger, dated July 21, 1972, concerning the Advisory Committee on Reactor Safeguards's review of your application for permits to construct the San Onofre Nuclear Generating Station, Units 2 and 3, is enclosed for your information.

Sincerely,

Original Signed by
Albert Schwencer for
R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
ACRS ltr dtd 7/21/72

cc w/encl:

Mr. Rollin E. Woodbury, Vice President
and General Counsel
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE	L:AD:PWR	L:PWR-3	L:AD:PWR's
SURNAME	VHWilson:eg	RBirkel KGoller	RCDeYoung
DATE	Ext. 7415 7/25/72	7/ /72	7/26/72

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

JUN 6 1972

Docket Nos. 50-361
and 50-362

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

In order for us to complete our review of your financial qualifications in connection with your application to construct the proposed San Onofre Nuclear Generating Station, Units 2 and 3, it will be necessary for you to furnish updated information on the following:

1. For each of the two units, current estimates of (a) total nuclear production plant costs; (b) transmission, distribution and general plant costs; (c) nuclear fuel inventory cost for first cores; and (d) the total of these. This should be accompanied by a statement describing the basis from which the estimate is derived, and an indication of the amounts of these costs already expended. If fuel is to be acquired by lease or other arrangement than purchase, the applicant should so state. The items included in the above categories should be as defined in the applicable electric plant and nuclear inventory accounts prescribed by the Federal Power Commission.

2. Each Company's total estimated costs (including costs for San Onofre 2 and 3) for all construction or acquisition of property, plant, and facilities by year through the latest year in which both units will be completed.

LB

OFFICE ▶						
SURNAME ▶						
DATE ▶						

RABirkel, DL
RCarroll, OC
OGC
VHWilson, DL (2)

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LA PDR
DL Reading
PWR-3 Reading

JUN 6 1972

3. For each Company, the approximate annual amount (or range) of financing that is expected to be provided by internally generated funds during construction of the two San Onofre units.

4. For each Company, published annual reports for Calendar Year 1971 and current interim financial statements.

5. The percentage of San Onofre 2 and 3 construction cost to be borne by each Company, and the percentage of the two units to be owned by each Company.

6. A prospectus from each Company for its most recent issue of securities.

7. Estimated dates for completion of construction of San Onofre 2 and 3.

8. Any other information which would be needed to accurately assess your financial qualifications.

The above data should be filed as an amendment to your application within thirty days to allow sufficient time for review of the information.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: Mr. Rollin E. Woodbury, Vice President
and General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE ▶	AD:PWR x7415:esp	L:PWR-3	L:PWR-3	L:AD/PWRs		
SURNAME ▶	EGoulbourne	RABirkel	KRGoller	RCDeYoung		
DATE ▶	5/ /72	5/ /72	5/ /72	5/ /72		

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Docket Nos. 50-361
and 50-362

JUN 5 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

This is in response to your letter of March 30, 1972, requesting that proprietary data, prepared by Western Geophysical Company of America on migrated depth-sections, Line WS70-3 and WS70-18, be withheld from public disclosure pursuant to 10 CFR 2.790(b).

After reviewing this data, we have determined that its disclosure would adversely affect the interest of the respective companies and is not required in the public interest, nor by the provisions of Part 9 of the Commission's regulations. Accordingly, we are withholding from public inspection the migrated depth-sections, Line WS70-3 and WS70-18, pursuant to the provisions of section 2.790(b) of 10 CFR Part 2.

Please note, however, that withholding of this information from public inspection shall not affect the rights, if any, of persons properly and directly concerned to inspect these documents.

Sincerely,

Original Signed by
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

cc: See attached

PREVIOUSLY CIRCULATED ON 4/13/72. SENT TO OGC 4/13/72 AND APPARENTLY LOST IN OGC.

FOR CONCURRENCES SEE DOCKET 50-361

OFFICE ▶	AD:PWR	L:PWR-3	L:PWR-3	L:AD/PWRs	OGC	L:RP
SURNAME ▶	x7415 VHWilson:esp	RABirkel	KRGoller	RCDeYoung		AGiambusso
DATE ▶	5/ /72	5/ /72	/ /72	/ /72	/ /72	/ /72

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

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SURNAME ▶						
DATE ▶						

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RABirkel, DRL

Docket Nos. 50-361
and 50-362

APR 13 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore
Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

The regulatory staff has prepared the attached draft criteria regarding industrial security. This draft material reflects preliminary thinking by the staff on this subject. It is furnished to applicants for the purposes of illustrating the scope of security planning that is considered appropriate and identifying specific aspects of security planning that should be addressed in security plans. Applicants are encouraged to use the draft criteria as a "checklist" in the preparation of security plans.

These criteria are not and should not be regarded as firm requirements of the regulatory staff; conformance with every criterion is not essential.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
Draft Criteria on
Industrial Security

cc: See attached

see 50-361 for encl

FOR CONCURRENCES SEE DOCKET NO. 50-361

OFFICE ▶	DRL:PWR-3	DRL:PWR-3	DRL:AD/PWRs			
SURNAME ▶	x7415 RABirkel:esp	KRGoller	RCDeYoung			LP
DATE ▶	4/ /72	4/ /72	4/ /72			

ccs: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P.O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

Docket Nos. 50-361
and 50-362

MAR 24 1972

Southern California Edison Company
ATTN: Mr. Jack B. Moore, Vice President
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: Mr. Martin R. Engler, Jr.
Senior Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

The Commission has adopted new regulations, effective March 21, 1972, concerning the prohibition of site preparation and related activities prior to the issuance of a construction permit. A copy of the new regulations is enclosed for your information. You will note that the San Onofre Nuclear Generating Station, Units 2 and 3 are affected by the changes to 10 CFR 50.10 and 50.12 regarding site preparation and exemptions. In order that we may determine the extent to which San Onofre Units 2 and 3 are affected by the revised regulations we need answers to the following questions:

1. What is the per cent completion of site preparation activities for the San Onofre Units 2 and 3 site?

2. If the answer to question 1 is less than 100%:

   a. What site preparation activities are completed?

   b. What site preparation activities are presently in progress?

   c. What site preparation activities, not yet undertaken, are still to be accomplished?

LP

OFFICE ▶						
SURNAME ▶						
DATE ▶						

3. Paragraph 50.10(d)(1) indicates that if you are, on March 21, 1972, conducting site preparation activities previously permitted, but now prohibited by the Commission's regulations, you may furnish to the Commission within 30 days after March 21, 1972, a written statement of any reasons, with supporting factual submission, why the activities should be continued pending the issuance of a construction permit. Do you plan such a filing? If so, forty copies of this information, certified by oath or affirmation, should be provided.

 This information, in addition to your "statement of reasons, with supporting factual submission," should include discussions on the balancing of the four considerations listed in 10 CFR 50.10(d)(2). In addition, this information should include your specific plans to minimize or reduce the environmental impact of the site preparation activities you are undertaking.

4. Any site preparation activities, not yet undertaken, are now prohibited by 10 CFR 50.10(c). Paragraph 10 CFR 50.12(a) indicates that the Commission may, upon application, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. However, as indicated in the initial discussion of these rules "...it is expected that specific exemptions will be used only sparingly...". Do you plan to request an exemption? If so, when and for what specific activities?

Ten copies of the information requested in this letter should be provided by March 31, 1972, so that we may plan the review activities required by the revised regulations in conjunction with our present review of your application for a construction permit for San Onofre Units 2 and 3.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Division of Reactor Licensing

Enclosure:
10 CFR - Prohibition of
Site Preparation and
Related Activities

cc: See attached

FOR CONCERNCES SEE DOCKET NO. 50-361

OFFICE ▶	~~DRL:PWR-3~~ x7415	~~DRL:PWR-3~~	~~DRL:AD/PWRs~~			
SURNAME ▶	RABirkel:esp	KRGoller	RCDeYoung			
DATE ▶	3/22/72	3/ /72	3/ /72			

MAR 24 1972

cc w/encl:
Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

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RWKlecker, DRL
DRL/DRS Branch Chiefs
VHWilson, DRL (2)
Attorney, OGC

Docket Nos. 50-361
and 50-362

SEP 16 1971

Jack B. Moore, Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Martin R. Engler, Jr.
Senior Vice President
San Diego Gas & Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

We appreciated the opportunity of meeting with representatives of Southern California Edison Company and San Diego Gas & Electric Company on August 20, 1971, and September 7, 1971, to discuss geology and seismology considerations for your proposed San Onofre Nuclear Generating Station Units 2 and 3. As a result of these meetings, it is our understanding that you plan to develop a program involving additional investigation and exploration to provide additional data and information that can be used to establish appropriate seismic criteria for the San Onofre Station.

We look forward to learning more about this program when you have had the opportunity to develop it.

Sincerely,

Original signed by
Frank Schroeder

for Peter A. Morris, Director
Division of Reactor Licensing

cc: See attached

OFFICE ▶	DRL:PWR-3	DRL:PWR-3	DRL:AD/PWRs	DRS:DIR	DRL:D/DIR	DRL:DIR
	x7415 RB	KRG				
SURNAME ▶	RABirkel:esp	KRGoller	RCDeYoung	EGCase	FSchroeder	PAMorris
DATE ▶	9/10/71	9/10/71	9/10/71	9/11/71	9/13/71	9/13/71

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LB

SEP 16 1971

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

OFFICE ▶						
SURNAME ▶						
DATE ▶						

50-362

Docket Nos. 50-361
and 50-362

AUG 11 1971

Southern California Edison Company
ATTN: Jack B. Moore, Vice President
P. O. Box 800
Rosemead, California 91770

San Diego Gas & Electric Company
ATTN: C. M. Laffoon, Vice President
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

During our review of your application for construction permits for the San Onofre Nuclear Generating Station, Units 2 and 3, we requested the comments and recommendations of the U. S. Department of the Interior, Fish and Wildlife Service. A copy of the Fish and Wildlife Service's report, containing comments and recommendations on environmental effects associated with the proposed construction and operation of the San Onofre Station, Units 2 and 3, is enclosed for your information. The radiological safety aspects of the material in the report will be considered in the safety evaluation of the San Onofre Nuclear Generating Station, Units 2 and 3. Copies of the Fish and Wildlife Service report are also being sent to the appropriate State and local officials.

We call your attention to the Department of the Interior's comments and recommendations and request you to cooperate with appropriate Federal and State agencies in implementing the necessary program for the pre-operational and post-operational environmental radiological monitoring surveys. Twenty copies of reports on the results of such surveys should be transmitted to us for our review and distribution to the Fish and Wildlife Service. We also urge you to give full consideration to the

OFFICE ▶						
SURNAME ▶						
DATE ▶						

other recommendations contained in the Department of the Interior's letter. In order that the Commission can provide the necessary assurances to the Department of the Interior, please give us your response to these comments as soon as possible.

Sincerely,

Original Signed by
P. A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Fish and Wildlife Report,
dtd. 7/30/71

cc w/encl:
Rollin E. Woodbury, General Counsel
Southern California Edison Company
P. O. Box 800
Rosemead, California 91770

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

Dr. John M. Heslep, Chief
Environmental Health and Consumer
Protection Program
Department of Public Health
2151 Berkeley Way
Berkeley, California 94704

Chairman, Board of Supervisors
County of San Diego, California

Mayor, City of San Clemente
San Clemente, California

Commanding General
U. S. Marine Corps Base
Camp Pendleton, California

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JRTotter, DBM
HDenton, DRL
VHWilson, DRL (2)
RABirkel, DRL

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SURNAME ▶	x7401 VHWilson:esp	RABirkel	KRG KRGoller	RCDeYoung	PAMorris	
DATE ▶	8/4/71	8/4/71	8/9/71	8/9/71	8/10/71	

AUG 11 1971

cc w/o encl:
Mr. William M. White, Chief
Division of River Basin Studies
Bureau of Sport Fisheries & Wildlife
U. S. Department of the Interior
Washington, D. C. 20240

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket 50-361

Docket Nos. 50-361
and 50-362

MAR 1 1971

Mr. Jack B. Moore, Vice President
Southern California Edison Company
601 West Fifth Street
P. O. Box 351
Los Angeles, California 90053

Mr. C. M. Laffoon, Vice President
San Diego Gas & Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

On several occasions during the course of our review of your application for a construction permit for the proposed San Onofre Generating Station Units 2 and 3, we have indicated that the information relevant to site geology and seismology submitted in your application and available from other sources can be interpreted in different ways by different individuals competent in the applicable technical disciplines. We have informed you that we need to review the results of additional analyses performed on the basis of these different interpretations before we can determine the acceptability of your current conclusions on this subject. It is our understanding that as a result of our meeting with your representatives on February 24, 1971, you intend to develop additional information to provide the broader base of reference needed to support your conclusion.

To be complete, the information you submit should include the following:

(1) A determination of the potential seismic acceleration values at the site for postulated earthquakes of various magnitudes, up to and including a magnitude of 8, occurring on the zone of deformation that exists approximately 5 miles offshore from the San Onofre site, and assuming that this zone connects with the Newport-Inglewood zone. The analytical model used in the determination, and the bases for the model, should be described in detail.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

MAR 1 1971

(2) An evaluation of the possible lengths of the assumed zone of deformation described in (1) above, including consideration of a possible connection of the offshore zone with the onshore Rose Canyon fault. The technical rationale for assuming various lengths of the tectonic zone should be presented.

(2) An evaluation of the potential effect and significance of the February 9, 1971, San Fernando earthquake on the general assessment of potential seismic events in the San Onofre area and, in particular, on the selection of the acceleration values for the design basis earthquake and operating basis earthquake for the proposed San Onofre site.

In order for us to complete our evaluation on a timely basis, your prompt response will be needed. Please contact us if you desire additional discussion or clarification of the information requested.

Sincerely,

Original signed by F. Schroeder
gow

Peter A. Morris, Director
Division of Reactor Licensing

cc: Rollin E. Woodbury, General Counsel
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601 West Fifth Street
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Los Angeles, California 90053

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

NOTE: Draft cleared by telephone on February 26, 1971, with:

Dr. Elmer Baltz
Mr. J. Devine
Dr. W. Hall

Distribution:

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CKBeck, DR	B/C's, DRL & DRS
MMMann, DR	ACRS (16)
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OFFICE ▶	DRL:PWR-3 X-7415 UB	DRL:PWR-3 KRG	DRL:AD/PWRs	DRL:D DIR	DRL: DIR	
SURNAME ▶	Birkel/sp	Goller	DeYoung	Schroeder	Morris	
DATE ▶	2/27/71	2/27/71	2/27/71	2/27/71	/ /71	

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DRS/DRL Branch Chiefs
NMBrown, DRL
Attorney, OGC
ACRS (16)
W. Nyer
Newmark & Hall

Docket Nos. 50-361
and 50-362

FEB 9 1971

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601 West Fifth Street
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C. M. Laffoon, Vice President
San Diego Gas & Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

As a result of discussions with members of your staff on January 18 and 19, 1971, we understand that you intend to supplement your responses to our letter of October 28, 1970, in specific, mutually agreed upon areas. In addition, on the basis of our review of your application inclusive of Amendment No. 5, we need the additional information described in the enclosure.

As we have noted previously, some of the information requested may be available in the public record in the context of our regulatory review of similar features of other facilities. If such is the case, you may wish to incorporate the information by reference in your application.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional Information

cc: See attached

OFFICE	DRL:PWR-3	DRL:PWR-3	DRL:AD/PWR's	DRL:AD/DIR	DRL:DIR
SURNAME	x7415 RAB RABirkel:esp	KRG KRGoller	RCDeYoung	FSchroeder	PAMorris
DATE	2/6/71	2/6/71	2/6/71	2/6/71	2/9/71

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

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OFFICE ▶						
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ADDITIONAL INFORMATION REQUIRED

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE UNITS 2 AND 3

DOCKET NOS. 50-361 & 50-362

1.0 GENERAL

1.3 Appendix A of the PSAR inclusive of Amendment No. 5 provides a discussion of your Quality Assurance Program. Expand this discussion to include the following specific items:

a. Provide an outline (contents) of your QA Manual indicating the scope of the plans, policies and procedures contained therein.

b. Provide a discussion of the manner whereby the required quality assurance requirements of your program have been followed since the original submission of your PSAR.

c. Describe the specific QA and QC requirements that will be met and implemented by your turbine generator supplier. Discuss the role of your European inspectorate, (Kennedy and Donkin) including the extent to which authority has been assigned to the inspectorate to approve and disapprove work and to stop work on the basis of its findings without prior concurrence from other segments of the QA organization.

1.4 The proposed San Onofre 2/3 facility will use a reactor design that is significantly different from past designs developed by Combustion Engineering, Inc. During the course of our review various important reactor design parameters have varied and we do not yet have a clear understanding of the reference design that should be used for evaluation purposes. Provide a suitable reference design and/or design criteria that should be used for assessing the acceptability of nuclear and thermal-hydraulic performance characteristics and the response of the reactor to accident conditions.

3.0 REACTOR

3.28 Section 3.3.2 of the PSAR indicates that a small bleed opening is provided in each CEA buffer section to prevent excessive pressure buildup and to provide cooling flow. Provide a summary analysis evaluating the consequences that could result from blockage of one or more bleed openings.

3.29 Discuss the use of water displacers to limit local power peaks including locations, fuel assembly markings identifying those containing displacers, potential consequences of mislocation of elements with displacers, the consequences of the removal of a single displacer pin on adjacent pin peaking factors, and the intended use of displacers in cores after the initial core and the effect of such use (or removal) on subsequent fuel cycles.

3.30 The response to request 3.12 provided by Amendment No. 4 did not include details concerning the intended vessel model flow tests for the 4-loop/217 assembly configuration. Provide the following information:

a. The preliminary design of the vessel model, including the internals and the fuel assembly simulator.

b. A detailed description of instrumentation, noting the types and intended locations of the instruments to be used in the model.

c. The range of test parameters to be studied during the test program.

d. A discussion of the modeling laws which have been applied.

e. The proposed utilization of the model flow test results in design calculations involving thermal performance.

3.31 The response to request 3.22 provided by Amendment No. 4 requires additional information. Provide a discussion of the 2-channel representation of the core and hot channel and include a discussion for any credit taken for interchange of fluid between the hot channel and surrounding channels during the transient.

3.32 The response to request 3.23 provided by Amendment No. 4 did not include adequate information regarding the planned experimental DNB program at Columbia University to demonstrate satisfactory performance of the fuel assembly design. Provide the following information:

a. A description of the test sections to be tested, including the number of heated rods, the length of rods (heated and non-heated), and the distance between grid spacers.

b. The axial and radial flux distributions to be studied. If available, peak to average values should also be provided.

c. The parameters that will be varied as part of the test program.

d. A definition of the experimental observance that will be considered to be the onset of DNB.

e. The types of transient DNB experiments to be performed.

f. A discussion of the method, employing a multi-dimensional code such as the COSMO/INTHERMIC Codes, that was or will be used to calculate local fluid conditions in the test bundle.

3.33 The response to request 3.24 provided by Amendment No. 4 did not provide adequate information concerning the minimum DNBR calculation with regard to axial flux distributions and radial peaking factors. Provide the results of calculations for the following conditions:

a. A symmetrical (cosine) axial flux shape.

b. An axial flux shape with the peak skewed toward the top of the core.

c. An axial flux shape with the peak skewed toward the bottom of the core.

The calculations should be performed at both rated and design overpower. For each calculation the following should also be provided:

a. The prediction of the critical heat flux value, including the non-uniform heat flux correction factor (the F - factor).

b. The predicted values of mass velocity and enthalpy along the hot channel.

c. The worst expected values of radial flux peaks for the three axial flux shapes.

3.34 The response to request 3.9 provided by Amendment No. 4 did not provide adequate information concerning the ability of the out-of-core nuclear instrumentation to monitor and control the symmetry of power distributions, both axially and in the X-Y plane. Provide a discussion of how the required power distribution information will be obtained and how it will be used to assure that design peaking is not exceeded for expected conditions of power maldistributions. Include in your discussion specific consideration of typical and limiting power distributions, the responses of normal and smallest permissible number of out-of-core instruments, the sensitivity of the measurements, error allowances, and action setpoints (alarms and trips).

3.35 The response to request 3.10 provided by Amendment No. 4 did not provide adequate information concerning errors in controlling azimuthal xenon oscillations. Provide analyses showing the behavior of uncontrolled azimuthal oscillations, the response of the core to proper control, the response to inadvertent use of the wrong set of control rods, and the detectability of such oscillations. Include a detailed discussion of the response of the fuel relative to the design limits.

3.36 The response to request 3.11 provided by Amendment No. 4 did not provide adequate information to assure that improper use of part-length control rods for control of axial oscillations will not lead to fuel damage in the absence of provisions for automatic protection. Provide analyses bounding the response of the fuel to power peaks in the upper and lower halves of the core from motion of the part-length rods in the improper direction including consideration of adverse combinations of the following identified contributing factors:

a. The permissible range of control values of the axial power ratio.

b. The possible deadband in the control point, including the nominal deadband, uncertainty in the in-core versus out-of-core power ratio correlation, and instrument errors.

c. The position of rod banks permitted to be in the core.

d. The time during the control history at which the error occurs, and

e. The location of the part-length rods when the error occurs.

3.37 We understand that multi-dimensional multi-node calculations have been made indicating the designer's ability to predict the details of local power distributions necessary to achieve high power density goals. Provide a summary discussion including examples of such calculations illustrating power distributions representative of reactor burnup conditions and operations expected with normal and part-length rods.

3.38 As part of your effort to assure that design peaking factors will be achieved and maintained in the San Onofre 2/3 design, we understand that lower values will be assumed as design goals. For example, for a design nuclear radial peaking factor of 1.55, the actual design goal would be for some lower value, i.e., 1.38. Provide documentation supporting the design philosophy to be used, your quantitative design objectives, and the degree of conservatism.

3.39 It is our understanding that "part strength" control rods (full length rods of lesser absorbing strength than normal CEA's) will be used in the design, if necessary, to provide control for azimuthal xenon instability. Provide documentation of the present status of design and development of such rods including materials, number, location, control features, and criteria for use. Reduction in azimuthal xenon control rod strength lowers potential radial peaking factors in the case of misused rods, but at some point makes the rods too weak to control the power distribution. Indicate the status of evaluation of this tradeoff as well as the research and development being performed or contemplated to demonstrate the efficiency of "part strength" control rods.

3.40 Provide analyses, for representative points in the fuel cycle, of the potential power peaking toward the top of the core. The natural tendency of BOL operation would be for peaking to occur toward the bottom of the core, so that depletion there might later produce power peaks in the upper half. Provide analyses indicating relative thermal limits for power peaking in the upper and lower portions of the core. You have indicated that margins to thermal limits would be reduced if the design axial peaking factor of 1.68 existed with the power peak in the top half of the core. Quantify the degree of peaking permissible as a function of location along the core axis for maintenance of constant thermal margins.

3.41 Provide a discussion of the planned startup tests that have been specifically designed to verify that the design power distribution objectives have been attained. Include discussions of the test methods, the instrumentation to be used, and the precision of measurement that will be required.

3.42 We understand that the adequacy of the physics calculational methods and the sensitivity of the out-of-core detectors will have been proven during startup programs on CE reactors scheduled for startup well in advance of San Onofre 2/3. Provide a summary of the information to be obtained from these programs as well as a specific time schedule when this information is to become available. What options are available should delays in acquiring proper data be encountered?

3.43 The response to request 3.17 provided by CE Topical Report CENPD-8 did not include results of tests to support the credit taken for mixing between subchannels. Test results which demonstrate the conservatism resulting from the choice of a 0.0035 value for the inverse Peclet number should be provided.

4.0 REACTOR COOLANT SYSTEM

4.30 Provide the results of an analysis of a reactor coolant pump overspeed incident resulting from a double ended pipe failure in a primary coolant loop. Include in a discussion of the analysis (using drawings where appropriate) the following considerations:

a. The potential for failure of the impeller, motor, and bearings, for shearing of motor mounts, and for rotor seizure during overspeeding. Identify the predicted mode of failure and list the speed that each of the principal components considered in the analysis would be expected to withstand without failure.

b. The size and weight of the pump motor rotor, the impeller, and the flywheel.

c. The maximum kinetic energy calculated for potential missiles that could be generated by a rotating component.

d. The probable trajectories of missiles (using drawings) that could originate from rotating components (include ricocheting) in relation to engineered safety feature equipment, and the need for missile protection.

e. The minimum energy required to fail a primary coolant loop cubicle wall, and to rupture a safety injection line, and a safety injection tank.

f. Test results of a pump - motor - flywheel assembly at design overspeed conditions and test data to support the predicted failure overspeed condition.

4.31 In the event that the consequences of a reactor coolant pump overspeed incident are considered indeterminate, what design measures could be taken to prevent overspeed beyond design limits?

4.32 Proper design of piping restraints and supports requires the performance of dynamic analyses for normally expected loadings due to anticipated occurrences as well as those due to seismic events and accidents. Describe the specifications that are or will be established to require that suitable dynamic analyses be performed for all significant credible dynamic loadings for all Class I systems.

5.0 CONTAINMENT AND STRUCTURES

5.21 Provide a table indicating energy distribution prior to the design basis loss-of-coolant accident, the amount of energy generated and absorbed from the time of pipe rupture to the time of the peak pressure, and the distribution of energy at the time of peak pressure. This listing should include the energy content of the following:

a. Primary coolant

b. Safety injection tanks

c. Fuel and clad

d. Core internals

e. Reactor vessel

f. Decay heat

g. Steam generators

h. Piping

i. Pumps and valves

j. Secondary coolant

k. Containment steam-air mixture

l. Steel structures

m. Concrete structures

6.0 ENGINEERED SAFETY FEATURES

6.14 The potential exists for the safety injection tank motor operated isolation valves to become closed inadvertently. Discuss the design features that will be provided to preclude such an event from taking place. Consider such aspects as control room alarms, independent and redundant valve position indication and circuitry to insure that the valves will be opened when required to be open, and to prevent inadvertent valve closure under credible operating and accident conditions.

6.15 Provide a summary of the results of an analysis of the consequences of transferring the Safety Injection System from the injection mode to the recirculation mode subsequent to a LOCA at a time too early to assure adequate pump NPSH. Include the effect on the core mechanical integrity and on predicted clad temperatures.

6.16 Provide a discussion concerning maintenance of equipment that is required to operate in the post accident environment during long-term recirculation conditions. Include aspects of accessibility of equipment and controls for maintenance or replacement.

6.17 The response to request 6.1 provided by Amendment No. 3 is not adequate for our needs. In the event that you plan to use an acceptance criterion of more than 2300 ^{o}F as an upper limit for the peak clad temperature calculated for a loss-of-coolant accident, provide additional information to justify the selected temperature limit.

6.18 The practice of permitting small diameter piping for essential systems to be "field-run" should be limited insofar as it is practicable to do so. When it is permitted then (1) stringent quality assurance measures should be taken to insure that the installation has been performed in such a manner that the assumptions made for design and safety assessment purposes remain valid, and (2) tests should be performed on the completed item to provide a final indication of acceptability. Provide the following information:

a. A discussion of the extent to which you will permit "field-running" of small diameter piping for essential systems, including all engineered safety features, in your facility.

b. The reasons why it is not practicable to limit the use of "field-running" to a greater extent.

c. The special rigorous quality assurance measures and performance tests that will be conducted to assure satisfactory installations.

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.8 Provide the following information regarding the San Onofre 2/3 turbine generators:

a. List the components to be fabricated in the United States.

b. State the requirements and depth of documentation required of materials of fabrication for all major components.

c. Discuss the review and approval of vendor specifications to be performed by SCE.

d. Provide an evaluation of previous failures of English Electric turbine generators, e.g., CEGB Hinkley Pt., and indicate the measures that will be taken to prevent occurrence of a similar event during operation of the San Onofre 2/3 turbine generators.

e. Indicate the extent to which the turbine overspeed protection system will be designed to conform with the requirements of IEEE-279.

11.0 RADIOACTIVE WASTE DISPOSAL AND RADIATION PROTECTION

11.9 The response to requests 11.1 and 11.2 provided by Amendment No. 4 require the following additional information:

a. Regarding the coolant and boric acid recycle system, provide an analysis of the use of concentrators (evaporators) in series and an analysis of the decontamination that may result from dissolving stable isotopes (spikes) in liquids prior to final demineralization. (The stable isotopes may scavenge the trace quantities of radionuclides in the liquids).

b. If it is proposed that some high conductivity liquid waste materials not be processed by the waste disposal system but rather be released directly to the environs if the radioactivity, determined from sample analysis, is within acceptable limits, provide justification that it is not practicable to treat these wastes further prior to release.

11.10 Provide a description of the planned treatment of steam generator blowdown and other potential radioactive wastes that originate from the secondary system.

14.0 SAFETY ANALYSIS

14.10 Provide an evaluation of the expected consequences of the use of incorrect enrichments in the core fuel, and of fuel assembly misorientation (worst case) during core loading or refueling. Include a discussion of the effects of these errors on the design peaking factors and the measures to be taken for the detection and prevention of such errors.

14.11 Provide additional information to justify the assumption that the CEA's will insert and remain in the core, as required to permit adequate ECCS operation, in the event of a design basis earthquake concurrent with a LOCA.

50-362

Docket Nos. 50-361
and 50-362

OCT 2 8 1970

Jack B. Moore, Vice President
Southern California Edison Company
601 West Fifth Street
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Los Angeles, California 90053

C. M. Laffoon, Vice President
San Diego Gas & Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

On the basis of our review of portions of your application for a permit to construct Units No. 2 and 3 of the San Onofre Nuclear Generating Station, we find that additional information is required to complete our evaluation. The specific information requested is described in the enclosure and is categorized into groups which correspond directly to sections of your application. This information was discussed with your representatives in meetings held on July 15, August 5 and August 21, 1970. We are continuing our review of your application and may request additional information as our review of your application proceeds.

We recognize that some of the information requested may have been provided by other applicants in connection with our review of other plants, and if you wish, this information may be incorporated in your application by reference. You may wish to amend your application by submitting revised pages (and annotating the change) for the appropriate portions of the Preliminary Safety Analysis Report rather than by submitting separate responses to each question.

Applic.

Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

Original Signed by
Frank Schroeder

for Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Additional Information Req'rd

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Southern California Edison Company
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Attorney, OGC
ACRS (18)
WNyer
Seismic Design Consultant

OFFICE ▶	DRL:PWR-3	DRL:PWR-3	DRL:AD/PWR's	DRL:D/DIR	DRL:DIR	
SURNAME ▶	x7415 RAB RABirkel:esp	ODP for KRGoller	RCDeYoung	FSchroeder	PAMorris	
DATE ▶	10/22/70	10/22/70	10/23/70	10/28/70	10/28/70	

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 28, 1970

Docket Nos. 50-361
and 50-362

Jack B. Moore, Vice President
Southern California Edison Company
601 West Fifth Street
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Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

Frank Schroeder

for Peter A. Morris, Director
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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

SAN ONOFRE UNITS 2 AND 3

DOCKET NOS. 50-361 & 50-362

1.0 GENERAL

1.1 Identify those design parameters for the San Onofre 2/3 nuclear steam supply system (NSSS) that are significantly different from the corresponding parameters for a NSSS designed by CE for a nuclear facility for which a construction permit recently has been issued.

a. Summarize the safety significance of these differences, including consideration of significant changes in any safety margins.

b. In each case, provide the technical basis for the change which supports the acceptability of the changes from the point of view of nuclear safety.

c. Provide a description of any R&D or experimental programs underway or planned that are intended to support any of these changes.

1.2 Appendix C of the PSAR provides a functional evaluation of the components of the systems shared by Units 2 and 3. Expand this evaluation to include Unit 1 and present a summary evaluation to indicate that each individual unit possesses sufficient independence to permit safe operation (or shutdown) in the event of an accident at one of the other units.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

2.0 SITE AND ENVIRONMENT

2.1 It is indicated in Section 2.6.3, Amendment 2 of the PSAR, that meteorological Δ T measurements will be made at elevations 58 feet apart. Explain your basis for concluding that this separation will provide adequate data to determine the site atmosphere stability characteristics. Also indicate the schedule for installation of the instrumentation and the method to be used for the determination of stability categories for the site. Include information on the sensitivity and accuracy of the instrumentation to be used.

2.2 In our calculation of the short-term doses that might result from postulated accidents we use meteorological parameters based on the worst diffusion conditions (including calms) that exist at a nuclear facility site 5% or more of the time. Table 2.6-7, Sheet 1, of the PSAR, indicates that stability index "F" conditions, with an average wind velocity of 3.6 meters per second, occur 25.73% of the time. Explain the basis for your use of Pasquill type "E" conditions, with an average wind speed of 2.2 meters per second for the short-term accident dose calculations. Explain the seeming inconsistency in the data presented in Table 2.6-7, Sheet 1 and Table 2.6-8, Sheet 1.

2.3 It is indicated on page 14.4.2-14 of the PSAR, that approximately 13 days following a loss-of-coolant accident the hydrogen concentration in the containment may be controlled by venting at a rate of less than 2% per day and that such venting would result in a maximum offsite thyroid dose of less than 20.4 rads. What whole body dose would result from the venting operation? Would the release be vented from ground level or from a stack? If it is from a stack, indicate the stack height in relation to adjacent buildings. What meteorological conditions were assumed in calculating the doses and what X/Q value was used?

2.4 What are the tornado design criteria to be used in the design of San Onofre 2/3, including the criteria specifying the controlling wind speeds, pressure drops and missiles associated with the design basis tornado?

2.5 Indicate, on a map, any known offshore oil wells or deposits near the site which would cause ground subsidence if the oil were removed during the lifetime of this nuclear facility.

2.6 Section 2.8.1 of the PSAR indicates that the California Parks and Recreation Department is planning a State Park that will abut the south property line of the site. If and when this park is built, how will consideration of it be factored into the emergency plan for the San Onofre 2/3 facility?

2.7 It is stated in Section 2.7, Amendment 2 of the PSAR, that the plant drainage system is designed to accommodate a storm with an estimated frequency of 1-in-100 years and an intensity characterized by 3 inches of precipitation per hour. What would be the potential effects of a maximum probable precipitation storm on the San Onofre 2/3 facility?

2.8 Provide the basis for the conclusion that the single strong motion seismograph already installed for Unit 1 will be sufficient to determine the nature of the seismic forces imposed on the structures of Units 2 and 3 in the event of an earthquake.

2.9 Provide additional information to demonstrate that you now have the authority to control all activities on the land included within the proposed exclusion area for the plant as required by 10 CFR 100, including the authority to exclude or remove persons and property from the land, and that you will retain this authority for the expected life of the plant.

2.10 The PSAR discusses (page 1.8-2d) the location of the Cristianitos Fault in terms of subdued geomorphic features, slumping and forms of mass wasting along the fault. Identify and describe the location and geomorphic features used to support your interpretation of the time origins of these features and their relevance to dating movements on the Cristianitos Fault. Reference sources of data on these features should be included.

2.11 Amendment 1 of the PSAR (pages 2.9-3a and 3b) discusses the ages of the coastal terraces in the Dana Point - San Clemente - Horno Canyon region. Provide geologic evidence to document the stratigraphic positions of the various collections (marine shell material) with respect to each other or to the abrasion terrace at the San Onofre site. Also, provide an analysis of the confidence levels for the minimum ages given in the table on page 2.9-3b.

2.12 Amendment 1 of the PSAR (page 2.9-3b) refers to a dissertation by Dr. Leonard Palmer. Provide the following information:

a. Availability of report.

b. Specific page reference supporting the suggestion that no deformation took place as a result of faulting.

c. Analysis and evaluation of Palmer's discussion of terraces in the San Onofre Bluffs - Los Pulgas quadrangles (pgs. 370-372).

d. Identification and analysis of all portions of Palmer's report used in your site evaluation.

2.13 Provide USC&GS data to support your conclusion regarding "areal subsidence", as discussed in the PSAR (page 2.9-5). If data for periods prior to 1933 are available, provide this information and your analysis of the information.

2.14 The PSAR states (Appendix 2A, Section 3.3.3) that a minimum of 90 feet of displacement has occurred on the Cristianitos Fault near San Onofre. Provide the basis for this conclusion, including geologic (and other) data and the significance of this displacement value. Include a discussion of the maximum amount of displacement near San Onofre and the significance of this displacement value.

2.15 The PSAR states (pg. 2.9-4) that seismic profiling indicates that at a short distance from the shore, the offshore trend of the Cristianitos fault changes from northerly to southeasterly and becomes Fault "D". Appendix 2B (Marine Advisors Report, pages 19 and 20) discusses Fault "D" and indicates that significant changes in sediment methodology occur near shore and attributes the changes to faulting. The PSAR does not provide adequate data or analysis to demonstrate that Fault "D" exists or to rule out interpretations other than faulting. Furthermore, since Figure 1 of the Marine Advisors Report shows that sparker data were not acquired between the last onshore exposures of the Cristianitos fault and seismic profiled line S-26, the PSAR interpretation that the fault bends offshore into a southeasterly trend has not been substantiated. Provide additional data to support your interpretation of the offshore trend and extent of the Cristianitos fault.

2.16 Figure 4, Appendix 2B of the PSAR (Marine Advisors Report) indicates that north-trending anticlinal and synclinal structures occur offshore between Faults "D" and "E". Provide data and analyses to confirm your interpretation of the extent and nature of these structures and a determination whether alternate interpretation of faulting are possible.

2.17 Figure 1, Appendix 2B of the PSAR (Marine Advisors Report) indicates that there are no seismic profiles (sparker) lines between the coast and line S-26. Therefore, the relationship (or lack thereof) of the structures discussed in question 2.16, to the onshore Cristianitos fault has not been demonstrated. Similarly, the relationship of these structures to the folds and faults further south (and generally east of line S-24) has not been established. Additional data and analysis is required of the relationship of these structures before a staff evaluation can be made of the relationship portrayed on Figure 4.

2.18 Figure 9, Appendix 2B (Marine Advisors Report) shows a possible northwest offshore extension of Fault "A" past the onshore position of the Pelican Hill fault. Also, on page 18 of the Marine Advisors Report it is stated that there is some evidence for the continuation of Fault "A" along the edge of the continental shelf from Dana Point north to Corona del Mar. Provide your analysis and data to support the conclusion stated in the PSAR (page 1.8-2e) that evidence shows offshore Fault "A" is not connected to the Newport-Inglewood fault, and that the same evidence suggests that Fault "A" is a seaward extension of the Pelican Hill fault.

2.19 The PSAR (page 1.8-2f) states that sparker records indicate similar rocks are juxtaposed by Fault "A", suggesting a small amount of displacement of Miocene-Pliocene units. The PSAR statement concludes, "that Fault 'A' is the seaward extension of the Pelican Hill fault...". Appendix 2B, page 16 (Marine Advisors Report) states, "The relative motion of the fault blocks along 'A' cannot be inferred from the structures observed on the sparker records ...". Provide justification for your interpretation that the Marine Advisors statement constitutes evidence that Fault "A" is a seaward extension of the Pelican Hill fault.

2.20 The PSAR (page 1.8-2f) states that "Fault 'A' exhibits no perceptible evidence of surface rupture, surface warping, or offset of geomorphic features, alluvium or manmade structures". However, Appendix 2B, pages 16-17, (Marine Advisors Report) states that Fault "A" disrupts near-surface shelf strata, but it probably is covered by a thin layer of unconsolidated sediments. Provide data and analysis supporting your conclusion regarding surface conditions (page 1.8-2f), including an analysis of the age of ruptured sediment along Fault "A" in the vicinity of the sparker profiles illustrated in Figures 5, 6 and 8, Appendix 2B (Marine Advisors Report).

3.0 REACTOR

3.1 We understand that hydraulic forces acting on the fuel assemblies could cause them to lift off the core support plate under certain flow conditions. Describe the design bases and criteria for the fuel assembly holddown device, including those with respect to fatigue failure considerations. Describe the testing that will be performed to assure its functional integrity under all loading conditions. Identify the applicable stress and deformation limits for all loading combinations.

3.2 What are the allowable design deflection limits for critical reactor internals components and how do these limits compare with the loss-of-function deformation limits?

3.3 Provide justification for the conclusion that the allowable diametral strain of the control element assembly (CEA) tubing is sufficient to accommodate the swelling of the boron carbide pellets that will occur with irradiation over the design life of the CEA.

3.4 Provide justification for the conclusion that CEA operation under steady state and transient conditions will be satisfactory with the core pressure drop characteristics of the San Onofre 2/3 design. Include a discussion of planned CEA performance tests including the extent to which these tests will simulate actual San Onofre 2/3 operating conditions.

3.5 Identify the extent, method and findings of analyses of the thermal stresses which would result in the core barrel and core support structures in the event of a design basis loss-of-coolant accident and subsequent operation of ECCS equipment.

3.6 We understand that an amplitude limiting device or snubber will be installed to limit core barrel vibrations. Provide the following information on this device:

a. A sketch of the device indicating the detailed location and the relationship of the component parts.

b. The basis for the design.

c. The requirements for fatigue analyses.

d. A summary of the analyses and tests which have been conducted to demonstrate that such devices will function as designed.

3.7 Provide a summary of the consideration given in the design of the reactor flow skirt to the potential for a fatigue failure or flow blockage.

3.8 With respect to the reactor vessel internals, describe the method of analysis that will be used to take into account each of the following:

a. Vertical and horizontal seismic loadings resulting from a design basis earthquake.

b. Blowdown hydrodynamic forces resulting from the design basis loss-of-coolant accident.

c. Combination of the design basis earthquake and loss-of-coolant accident.

d. Vibrations induced under normal reactor operating conditions.

3.9 In Section 3.4.4 of the PSAR, it is stated that "CEA's will be used to the minimum extent possible and in configurations that will result in a combined radial and axial peaking factor below the design limits used to determine the core thermal margins". Discuss how this will be accomplished, including the instrumentation requirements for monitoring the power distribution and the method of obtaining peaking factor information from the instrument readings.

3.10 Provide additional information concerning provisions for reactor control, monitoring of control effects, and automatic protection if the core should be unstable or marginally stable with respect to azimuthal xenon oscillations. The out-of-core flux detectors may be able to detect such oscillations, but it is uncertain that they are adequate to monitor control effects, particularly if it is assumed that an error is made in the use of the control provision. If no automatic protection is to be provided, assess the possibilities for core damage from such an error and, if fuel damage could occur, explain your basis for not providing automatic protection to preclude such a possibility.

3.11 Discuss the possibility that fuel damage might result from improper use of part length control rods in controlling axial xenon instabilities or from failing to recognize the presence of an axial oscillation. If fuel damage could occur, explain your basis for not providing automatic protection to preclude such a possibility.

3.12 In Section 3.5.2 of the PSAR, reference is made to flow tests used to evaluate differences between previous designs and that proposed for this plant. With respect to these studies, provide additional information on the results of flow model tests directly applicable to the proposed design. For those aspects of the hydraulic design of the reactor for which the design bases have not previously been confirmed by model studies, discuss the studies you propose to conduct in order to obtain this information.

3.13 Provide a schematic flow diagram that shows the distribution of the total coolant flow rate through the reactor and distinguishes between the active flow and the leakage flow. Discuss the results of calculations and experiments that support the conservatism of the value used for the active core flow rate. Discuss the experiments which justify the use of internal leakage (between fuel assemblies) within the core as coolant available for removing heat from fuel assemblies. Discuss experiments concerning the flow in the reflector region which is considered to be in excess of that required for cooling.

3.14 In Section 3.5.2 of the PSAR, the reactor vessel pressure drop from nozzle to nozzle is stated to be 40 psi and, for design conservatism, a value of 42 psi was used. Discuss the adequacy of the 5% margin in reactor pressure drop in view of the high design flow rate provided for the high performance core. Discuss the additional safety features applied with respect to hydraulic resistances for the remaining portion of the primary loop. Provide a graph showing the design basis pump head versus flow. Include curves of system pressure drop versus flow for the nominal, design, and 10% above design case. To the extent possible, information from the Palisades plant should be included as verification of the conservatism in the design basis.

3.15 Section 3.5.3 of the PSAR presents the engineering factor on local heat flux (F_{LOC} = 1.03) and an engineering factor for hot channel heat input (F_{Ave} = 1.03) based on variations over the length of four fuel rods that are assumed to enclose the hot channel. The selections of these values were based on information obtained during the fabrication of the Palisades fuel assemblies. Discuss the conservatism of these values with regard to manufacturing experience (i.e., the anticipated as-built factors for Palisades) as well as the significance of the differences in fuel enrichments, pellet diameter (tighter dimensions), and fuel rod outside diameter between the Palisades and San Onofre fuel rod designs. Present information on the sampling program followed during the Palisades fuel manufacturing process to obtain confirmation of the conservatism in the engineering factors selected for this initial design. Discuss the procedures followed to distinguish enrichment errors from density errors. Discuss any anticipated manufacturing processes that may differ from those applied to the Palisades fuel rods and might negate the bases for the assumed engineering factors. Using the same degree of detail provide additional information to

support the selection of an engineering factor of 1.05 for hot channel flow reductions due to fuel rod bowing, pitch, and outside diameter.

3.16 Provide additional information to support the selection of an inlet flow maldistribution factor of 1.03. Include the results of relevant model studies which support the use of the proposed design parameter.

3.17 Section 3.5.3 of the PSAR presents a discussion of possible fluid interchange between adjacent channels and the possible axial enthalpy profiles which may be calculated for adjacent channels. A discussion of laboratory tests is also presented to support a value for the mixing coefficient of 0.93 at the location of the minimum DNBR. To permit evaluation of this parameter the following additional information is required:

a. A discussion of the developed form of the equations from which mixing was calculated, based on conservation of mass, energy, or momentum. This discussion should also include the methods used for obtaining solutions to the equations.

b. A discussion of the empirical correlations developed for considering mixing and the experimental data to support the choice of the design parameter.

3.18 The amount of coolant mixing permitted by the core mechanical design has been shown to be a sensitive parameter in evaluating core thermal performance. For this reason, the applicability of mixing coefficients obtained from close bundle geometries for use in an analysis of a hot assembly having an open lattice geometry should be discussed. Any conservatism in the value of the mixing coefficiencts applied to core design should be stated.

3.19 Section 3.5.3 and Appendix F of the PSAR include a brief discussion of the COSMO code used for performing steady state thermal and hydraulic calculations. It is requested that additional information be provided discussing the details of this code. This information should include discussion of the basic conservation equations and the procedure used to obtain the solution. The correlations used to calculate single-phase and two-phase pressure drops across the core should also be discussed.

3.20 Provide a discussion to justify the use of AECL-1552 "Heat Transfer Coefficients Between UO_2 and Zircaloy-2", as modified by Kjaecheim and Rolstad for design application to the San Onofre 2/3 core. Include a list of the values for the constants necessary to calculate the fuel-to-clad gap conduction using the referenced equation. Present the calculated value for the gap conductance using this equation and show how the value is expected to change with burnup.

3.21 Provide a discussion of the applicability of the W-3 DNB correlation to the San Onofre 2/3 core design for the range of thermal conditions postulated for normal operation and for anticipated transients. Discuss in detail the applicability of the W-3 DNB correlation for the fuel rod geometry proposed. Discuss the uncertainties in safety margins that may be present when parameters selected for the San Onofre 2/3 reactors extend beyond the range of the experimental data from which the correlation was developed.

3.22 Describe the computer code used to calculate core thermal and hydraulic performance during anticipated transients such as loss of coolant flow and loss of load.

3.23 Discuss the planned experimental program underway at Columbia University to investigate conditions sufficient to cause DNB for the San Onofre 2/3 fuel assembly. This discussion should include the analytical verification of the W-3 DNB correlation for the proposed fuel assembly design for the anticipated range of axial flux distributions and hot channel fluid conditions. Provide additional details in terms of geometry, heat flux patterns, instrumentation and test conditions for the three DNB test sections discussed in Section 1.4.2.7 of the PSAR.

3.24 Provide plots of (1) heat flux, (2) mass velocity, (3) enthalpy, and (4) DNB ratio for the core, the hot assembly, and the hot channel as functions of distance along the channel at design operating conditions.

3.25 Describe the core exit coolant water thermocouples to be provided in the San Onofre 2/3 reactors, including number, type, and location. Discuss how they will be used as operational aids during the life of the facility.

3.26 List the core components for which buckling could be a possible mode of failure for a combination of loads including those due to the design basis earthquake and the loss-of-coolant accident. Describe the method of analysis to be used and the safety margin (margin between condition considered to constitute failure and the as-calculated condition for the combined loading) to be required.

3.27 Discuss how seismic stresses will be determined for the reactor internals. Give details to show development of seismic loadings from ground motion inputs for the supporting structures to the final input used for the analysis of the internal structural members. Indicate points at which there is a change in method of analysis (e.g., dynamic to static, elastic to plastic) and give the basis for the load values used.

4.0 REACTOR COOLANT SYSTEM

4.1 The code classifications of components, such as pressure vessels, storage tanks, piping, pumps and valves, as described in Sections 4, 6, 9, 10 and 11 of the PSAR, are insufficient to enable evaluations to be made of the extent to which these classifications apply to components within the reactor coolant pressure boundary, and the boundaries of engineered safety systems, shutdown and control systems, steam and feedwater systems, cooling water systems, purification and cleanup systems, sampling systems, and radwaste systems. By the use of quality group letters A, B, or C on appropriate Piping and Instrumentation Diagrams for the reactor coolant pressure boundary and for each of the above-mentioned systems, delineate the system boundary limits within which each of these quality groups (and their associated code classifications) apply.

The industry codes for water and steam containing components associated with these quality group classification letters A, B, or C are identified in the Summary of Minimum Nuclear Codes and Standards for Components of Water-Cooled Nuclear Power Units that is presented on page 21 of this document. To provide further assistance in responding to these requests for additional information, a definition of the reactor coolant pressure boundary is presented on page 20.

4.2 Provide the following information with respect to vibration analyses, testing and monitoring:

a. To what extent have or will vibration analyses of the reactor internals and primary coolant system be made which take into account both normal and emergency modes of operation? Describe the amplitude and frequency limits that have been established as design goals.

b. What preoperational vibration testing will be performed? Describe the number, type and location of instruments for each test. Describe the operating conditions under which vibration tests will be conducted. If the tests do not include the temperature and flow conditions representative of normal power operating conditions, state the basis and methods for applying the results of the tests to the evaluation of vibration under these conditions.

c. What provisions will be made for long term monitoring for vibration or for the presence of loose parts in the reactor pressure vessel and other portions of the primary coolant system?

d. To what extent will results obtained from vibration analyses and testing programs for other facilities be used in the design of the San Onofre 2/3 facility? Provide the bases for determining the applicability of such results to the San Onofre 2/3 facility.

4.3 The list of design transients to be used in the fatigue analysis of the reactor coolant system, as given in Section 4.3.2 of the PSAR, is incomplete. Provide a list of other transients that will be used in the analysis, including transients due to inservice hydrostatic testing, single operator errors, and malfunctions of components in the control and other systems. Include the number of cycles that will be assumed for each transient. Discuss the conservatism of the selected transients and describe the methods to be used to control the number and severity of each transient that the plant will experience during its service lifetime.

4.4 Provide the criteria that will be applied in the design of the reactor coolant system component supports. Provide a description of each type of support to be used, with accompanying drawings or sketches and identification of the materials to be used and the applicable design codes in each case.

4.5 Describe the plans that will be followed to avoid partial and/or local severe sensitization of austenitic stainless steel during heat treatment and welding operations for core structural load-bearing members and components of the reactor coolant pressure boundary. Describe the welding methods, heat input and the quality controls that will be employed in the fabrication of these items.

4.6 Provide the following information to permit an assessment to be made as to whether seismic design bases will be correctly translated into the required specifications, drawings, procedures, and instructions so that the necessary structures, systems, and components will be able to withstand seismic loads combined with the other appropriate concurrent loads:

(1) A description of the design organizations that will be involved in the seismic design of all structures, systems, and components of the plant that are related to safety.

(2) A description of the responsibilities of the organizations to be involved in the seismic design, the extent to which these responsibilities will be promulgated to the orgnizations in writing, and the identification of the design organization that will be assigned overall responsibility for the adequacy of the seismic design.

(3) A description of the documented procedures that will be promulgated to provide for the interchange of needed design information and changes thereto and the coordination of the various facets of the seismic design among the involved design organizations.

(4) A description of the manner by which you will assure that the design procedures described in (3) above will be followed.

(5) A description of the design control measures that will be instituted to verify or check the adequacy of the seismic design and by whom they will be performed, and of the design procedures that will be promulgated to provide for these measures.

(6) A description of the requirements that will be included in the purchase specifications for safety-related equipment to assure that this equipment is adequately designed to withstand and can function under the seismic design conditions, and of the provisions that will be included in the purchase specifications to permit the purchaser to verify that these requirements are satisfied.

4.7 Describe the design and installation criteria to be used for the mounting of the pressure-relieving devices (safety valves and relief valves) within the reactor coolant pressure boundary and on the main steam lines outside of containment. In particular, specify the design criteria to be used to take into account the full discharge thrust loads of the valves and indicate the provisions made to accommodate these loads. Specify the design criteria, applicable to the main steam lines, that take into account the bending and torsional loadings that will be developed in the event that all safety valves in the line discharge concurrently.

4.8 The proposed seismic design spectra for the Operating Basis Earthquake (OBE) produce a peak amplification factor of approximately 2.2 for 2% damping. Analyses of historic seismic records have indicated amplification factors in the range of 2.5 to 4.5. In addition, the use of different magnification factors for the OBE and the Design Basis Earthquake (DBE) may not be sufficiently conservative. Provide a more appropriate seismic design basis for the OBE and DBE by considering either:

a. Seismic design spectra for the site which include a more appropriate amplification factor and the effects of distance between the seismic disturbances and the site on the predominant periods in defining acceptable seismic design spectra, or

b. The selection of appropriate damping factors to be used with other spectra taking into consideration the relation of these factors to the applicable construction code allowable design stress limits which will be used in designing structures, systems, and components for the DBE and the OBE.

4.9 Provide the criteria which will be used to determine the time histories associated with the seismic design spectra of the site, and specify the time histories utilized and the enveloping technique employed.

4.10 List all Class I structures, systems, and components. Indicate the method of seismic analysis (modal analysis reponse spectra, modal analysis time-history, equivalent static load analysis, empirical test analysis or other method) used for the design of each item, and the applicable stress and deformation criteria and the damping values used in each analysis. Provide a brief description of all the methods that were used for the seismic analysis of the Class I items.

4.11 The use of constant vertical load factors for the vertical response component in lieu of a combined vertical, torsional, and horizontal multi-mass dynamic analysis may not be sufficiently conservative. Provide the basis for determining the combined vertical, torsional and horizontal response loads for Class I structures, systems, and components.

4.12 Equipment and floor response spectra for various locations within the building structures are not directly obtainable from the modal response spectra multi-mass seismic system method of analysis. Provide a more appropriate basis for obtaining floor response, either by demonstrating that the proposed method is equivalent to a multi-mass time history method or by submitting other theoretical or experimental justification. Provide the design basis for consideration of the differential movement between floors.

4.13 Provide the criteria to be used to compute shears, moments, stresses, deflections and/or accelerations for each seismic-excited mode as well as for the combined total response, including the criteria for the combining of closely spaced modal frequencies.

4.14 Appendix B (B.3.2.2.1.2) of the PSAR states that the categories of loading conditions (normal, upset, and faulted) are applicable to reactor coolant system components. Identify any other equipment and components not part of the reactor coolant pressure boundary for which the stress limits associated with faulted conditions will apply. If faulted limits are used for such cases, provide the bases for the loading conditions and the justification for applying such limits.

4.15 Provide a summary description of the materials to be used in the fabrication of the primary coolant system, recirculation pump, flywheels, the procedures for quality control of the fabrication, and planned preoperational and inservice inspection requirements. Provide an analysis to assess the probability of failure of the flywheels under operating conditions. Support the probability analysis with a summary of available operating experience with similar components. Provide a description of the potential equipment damage and radiological doses that could result if the flywheel fragmented and produced missiles, in spite of careful design, fabrication, and inspection procedures. Consider the complete spectrum of missile energies and trajectories.

4.16 Provide a discussion of the design criteria regarding the protection of the containment, control room and other vital plant features against missiles that could result from tornadoes or turbine failures.

4.17 In assessing the consequences of pipe ruptures you have considered both circumferential failures and slot-type longitudinal failures. For the latter type of failure you assume a break area not greater than the cross sectional area of the ruptured pipe. Provide justification for not assuming that the break area for this type of failure could be as large as twice the cross sectional area of the pipe.

4.18 What consideration has been given in the design of the primary cooling system piping to the thermal cycles which would be caused by the injection of cold water by the ECCS? Describe the design analysis that will be made to assure the integrity of the injection lines that will not be protected by thermal sleeves at the connection to the reactor coolant lines.

4.19 Provide a discussion of the design criteria that will be used to assure that the containment liner and essential equipment within the containment, including components of the primary and secondary coolant systems, engineered safety features, and equipment supports, will be adequately protected against missiles generated within the containment, blowdown jet forces, and pipe whip. The discussion should include:

a. Pipe restraint design requirements to prevent plastic hinge formation.

b. The features to be provided to shield vital equipment from pipe whip.

c. The measures to be taken to physically separate piping and other components of redundant engineered safety features.

d. A description of the analyses to be performed to determine that the failure of lines, with diameters of 3/4 inch or less, will not damage the containment liner under the most adverse design basis accident conditions.

e. The spectrum of missiles and the penetration formulae to be assumed for design purposes.

f. The analytical methods to be used.

4.20 Provide a discussion of the pressurizer design characteristics and the basis for the selected value of each major design parameter. Describe the methods of analysis to be used to calculate stresses in the heater penetrations in the lower head of the pressurizer and to demonstrate that the requirements of Section 3 of the ASME Code are met. Discuss the effects of heater penetration on the thickness of the vessel head, the method of determining these effects and the manner by which compliance with subarticle N-450 of Section 3 will be accomplished. Provide a summary of the maximum stress intensities and accumulative damage use factors for the pressurizer and indicate the points of analysis on appropriate sketches.

4.21 Provide a discussion of leak detection provisions, including the following:

a. Provide reactor coolant pressure boundary leakage criteria for both contained and uncontained leakage. List leakage design limits for all reactor coolant pressure boundary equipment. Include the maximum leak rate from unidentified and identified sources that will be permitted during operation.

b. Describe the leak detection system provided for all other Class I (seismic) fluid systems and list those Class I (seismic) fluid systems for which no special leak detection system is provided.

c. Discuss in detail the sensitivity and response time of the leak detection system for the reactor coolant pressure boundary.

4.22 Describe the extent to which you have reviewed the plant design to determine that annealing of the reactor pressure vessel will be feasible should it be necessary because of radiation embrittlement after several years of operation. State the maximum reactor vessel temperature that can be obtained using an in-place annealing procedure.

4.23 With respect to the proposed reactor vessel material surveillance program provide the following information:

a. What is the expected neutron exposure rate of the test specimens compared with that of reactor vessel wall?

b. Will it be possible to insert capsules after the reactor becomes operational?

c. Has sufficient archive material been set aside to fabricate enough specimens for a minimum of two capsules?

d. Provide the chemistry of the vessel materials, including the residual content in weight percent to the nearest 0.01%.

e. Since capsule brackets are to be welded to the vessel wall, provide justification that the safety of the vessel will not be jeopardized; also, indicate the closeness of the capsules to the reactor vessel wall.

4.24 It is stated in the PSAR (Appendix E) that ASME Section 11 will be complied with "to the greatest extent practicable". Will your design meet the IS-140 accessibility requirements of the code? Identify any of the reactor coolant pressure boundary area initial base line tests called for in Section 11 that are to be omitted, and justify the omission. Provide a discussion to justify any design feature that prohibits inspection because of inaccessibility.

4.25 Discuss the inservice inspection program for mechanical systems outside the reactor coolant pressure boundary, including items to be inspected, accessibility requirements, and the frequency and types of inspection. Consider vessel supports, primary pump flywheels, components of the secondary system, including the main steam line, and all the engineered safety features, including emergency core cooling system lines external to primary containment.

4.26 Describe the design of the control element assembly drive housing thermal sleeves located in the upper head. If removable thermal sleeves are contemplated, provide the design criteria, for the sleeves, including requirements with respect to fatigue analyses.

4.27 We understand that you plan to use an automated or mechanized ultrasonic inspection technique to inspect the reactor pressure vessel belt line region, nozzles, lower head, and flange areas to meet certain requirements of your planned Inspection Program. Describe the preliminary design of the systems to be used to conduct these inspections with particular reference to rails, brackets, and other equipment that may require permanent mounting on the reactor vessel.

4.28 For all pressure-retaining ferritic components of the reactor coolant pressure boundary, (see definition page 20) for which the lowest pressurization temperature is below 250 °F, provide fracture toughness data (Charpy V-notch fracture energy curves and Drop-Weight Test NDT temperature) for plates, forgings, piping and weld material. The lowest pressurization temperature of a component should be taken to be the lowest temperature at which the pressure within the component exceeds 25 percent of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50 °F/hr., under normal operation, system hydrostatic tests or anticipated transients.

4.29 For reactor vessel belt line materials, including welds, provide:

a. The highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the weak direction, and

b. The minimum upper shelf energy value for the weak direction, which will be acceptable for continued operation toward the end of the service life of the vessel.

AEC Definition - Reactor Coolant Pressure Boundary.

(a) Reactor coolant pressure boundary[1/] means all those pressure containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) part of the reactor coolant system or

(2) connected to reactor coolant system, up to and including any and all of the following:

(a) the outermost containment isolation valve in system piping which penetrates primary reactor containment,

(b) the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

(c) the reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping.

1/ Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary may be excluded from these requirements provided:

(a) For postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

(b) The component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and its closure time must be such that for postulated failure of the component during normal reactor operation and assuming the other valve is open, the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

SUMMARY OF MINIMUM NUCLEAR CODES AND STANDARDS FOR Components of Water-Cooled Nuclear Power Units

COMPONENT	NUCLEAR CLASSIFICATION		
	GROUP A	GROUP B	GROUP C
Pressure Vessels and Storage Tanks (> 0 psig and not vented)	ASME Section III(1) Class A	ASME Section III(1) Class C	ASME Section VIII(2) Division 1
Storage Tanks (Atmospheric) in Groups B and C only		API-650(3) AWWAD100(4) or USAS B 96.1(5) With Supplementary Requirements (a)	API-650(3) AWWAD100(4) or USAS B 96.1(5) With Supplementary Requirements (b)
Piping	ANSI B 31.7(6) Class I	ANSI B 31.7(6) Class II	ANSI B 31.7(6) Class II
Pumps and Valves	ASME Standard Code for Pumps and Valves Class I(7)	ASME Standard Code for Pumps and Valves Class II(7)	ASME Standard Code for Pumps and Valves Class III(7)

Codes:

(1) ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1968 Edition, Applicable Code Cases and Addenda.

(2) ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, Division 1, 1968 Edition, Applicable Code Cases and Addenda.

(3) American Petroleum Institute Standard, Welded Steel Tanks for Oil Storage, API-650, Third Edition and Supplement.

(4) American Water Works Association Standard for Steel Tanks - Standpipes, Reservoirs, and Elevated Tanks-For Water Storage AWWA D100-67.

(5) ANSI B96.1-1967, Specification for Welded Aluminum-Alloy Field-Erected Storage Tanks (Formerly USAS B96.1 - 1967).

(6) ANSI B31.7 Nuclear Power Piping Code, 1969 Edition, Applicable Code Cases and Addenda, (Formerly USASI B31.7).

(7) Draft ASME Code for Pumps and Valves for Nuclear Power, Nov. 1968 Edition.

Supplementary Requirements:

(a) 100% Volumetric Examination of the Sidewall Weld Joints and 100% Surface Examination of the Sidewall - to - Bottom and Sidewall-to-Roof Joints in Accord with the Rules of ASME Section III, Class C.

(b) 100% Surface Examination of Weld Joints in Accord with the Rules of ASME Section VIII.

5.0 CONTAINMENT AND STRUCTURES

5.1 You state that the Class I reinforced concrete structures will be designed in accordance with Building Code Requirements for Reinforced Concrete, ACI 318-63, including the 1970 Proposed Revisions. What structural components will be designed for shear using the concepts of Section 11.15, Shear Friction, of the 1970 Proposed Revisions and what is the origin of the loads that are being designed for by this technique? For those components to be designed by this concept, indicate the value of the coefficient of friction that will be assumed.

5.2 The joint testing program that will be imposed during construction for the Cadweld mechanical splice system for reinforcing steel, as described in Appendix 5E of the PSAR, is a less extensive program than that which we have accepted for other facilities (see for example, Bechtel Corporation's "Rebar Splicing Procedure Specification", RB-MS-T, Revision 3, 2/2/70). Justify the adequacy of your proposed program for quality control of Cadweld splicing.

5.3 Provide further justification for the proposed tendon surveillance program. We understand that Bechtel has been conducting a study since late 1969 in order to establish a tendon surveillance program using statistical analyses. Provide a discussion of the results of the Bechtel study available to date.

5.4 The Proposed Revisions of 1970 for the ACI 318-63 Code define a more conservative approach for computing the flexural cracking load at a section. The modifications to ACI 318-63 presented on page 5.1-28 of the PSAR do not reflect this conservatism since M' is defined at a distance of d/2 from the section. Explain whether the 1970 Proposed Revisions will be incorporated in your criteria.

5.5 Will the construction technique known as "slip forming" be used in the construction of the containment building? If slip forming is to be used, will conformance to ACI-347 be required?

5.6 Provide a discussion of your design criteria for small diameter instrument lines which penetrate the containment, including (1) isolation provisions in the event of a line break during normal operation and following a loss-of-coolant accident, (2) testability of isolation provisions, and (3) inservice inspection for these lines.

5.7 We understand that connectors will be used in power and control electrical containment penetrations for voltages up to 600 v. What are the design bases and criteria for these connector-type penetrations?

5.8 What design bases and criteria will be applicable to containment penetrations for small and/or multiple pipes?

5.9 To what extent will the integrated leakage tests which will be performed periodically during plant life be used to check the structural characteristics of the containment? Will the containment structures be designed to permit testing at the calculated peak accident pressure at any time during plant life? We understand that a pressure test is to be made on the completed containment structure using air at 69 psig. Describe the measurements that will be taken and the observations to be made during the test. Will the results of this test be compared with the predicted results and with the results from similar containments? What instrumentation and equipment located in the containment must be protected or removed prior to periodic leak rate testing? List the items to be protected and those to be removed.

5.10 Provide a discussion of the components within containment that have been coated to protect them from the post-accident environment, describe the coating materials, the bases for their selection, their functional performance requirements and their predicted performance under accident conditions. Describe the extent to which predicted deterioration of protective coating materials under accident conditions could affect the performance of the emergency core cooling system and the containment spray systems. Include consideration of (a) the effects of dissolved chemicals from coated materials on crud deposition in pipes and heat exchangers, and (b) the possible clogging of strainers and spray nozzles by flakes and particles of protective coatings.

5.11 We understand that cooling may be required to reduce the concrete temperature in areas near containment penetrations for piping carrying hot fluids. Provide the design criteria and a preliminary design for the equipment that will perform this function.

5.12 Describe how potential bending and axial loads and movements will be considered in the design of both hot and cold pipe penetrations.

5.13 What design provisions will be made for periodic surveillance inspections of the containment liner and periodic structural testing of the containment?

5.14 What is the basis for the proposed spot examination of only two percent of the containment liner welds during erection?

5.15 Expand Table 5.1-3 of the PSAR, "Containment Isolation Valve Arrangements" to include the following information by including sensor and/or instrumentation lines and by adding the following information:

a. Pipe size

b. Valve position indication

c. Indication of valve position upon loss of motive force (power, air spring, motor loss)

5.16 Will any means be provided for the detection of a penetration failure or other breach of the containment during normal operation? If so, state the smallest leak that will be detectable by this method and describe the basis for the stated value.

5.17 Discuss the bases for each of the design criteria that you propose to use in the design of the spent fuel storage pool. Consideration of the following should be included:

a. The maximum fuel storage capacity.

b. Provisions made in the design to limit the consequences of a loss of pool cooling capability with the maximum permitted fuel in storage.

c. The temperature gradients to be assumed for the design of the fuel pool concrete structure. Describe whether a cracked concrete section will be used in design and how much thermal cracking will be assumed. How will the design accommodate the thermal temperature gradients?

d. Design provisions in the fuel pool cooling system piping and components for periodic testing and inspection (especially for leakage).

e. Specific provisions to protect the spent fuel pool against missiles that could result from a tornado or a turbine failure, including protection against missiles reaching stored fuel.

f. Design provisions to preclude or accommodate the loss of pool water that might result from the drop of a cask in the cask loading area of the pool, and provisions to preclude the loss of function of vital equipment as the result of consequent flooding of compartments below the fuel storage floor elevation.

g. Provisions made in the design of cranes and crane structures so that the occurrence of the design basis earthquake could not result in collapse of these items. Include loaded and unloaded conditions of the crane.

h. Pool water level monitors and alarms (locally and in the control room).

i. Assurance that the auxiliary building iodine filtration system would be operational in the event of high radioactivity levels in the fuel pool area.

j. The auxiliary building design to maintain controlled leakage characteristics around the fuel pool region whenever fuel handling operations are performed and whenever a crane is used within the fuel pool area while spent fuel is in storage.

k. Administrative and design provisions, such as interlocks and mechanical crane stops, to prevent the auxiliary building crane from moving heavy equipment loads over spent fuel in storage.

5.18 Provide curves of the mass and energy release rate to the containment building for the design basis loss-of-coolant accident (2.04 square foot break).

5.19 Provide a plot of the condensing heat transfer coefficient for structural surfaces within the containment as a function of time following the design basis loss-of coolant accident.

5.20 Has the COPATTA Code been modified to use the same thermodynamic model for separating the blowdown steam and water as that used in the CONTEMPT Code? If so, provide the details of this modification.

6.0 ENGINEERED SAFETY FEATURES

6.1 What is the maximum fuel clad temperature used as a design criterion for the ECCS performance? What is the basis for this value?

6.2 Discuss the NPSH design requirements for the containment spray and safety injection pumps, and state the margin between the required NPSH and that which will be available. What tests will be performed to verify this margin?

6.3 What is the basis for the amount of borated water in the refueling water storage tank? How does this relate to the volume of water required for the design basis accident? What will be the water level in the containment building (especially around the reactor vessel) during the loss-of-coolant accident after all of this water has been introduced into the containment building? Discuss the design of the containment sump with respect to the use of screens to prevent debris from entering the emergency core cooling system.

6.4 What features will be provided for the detection and isolation of leaks in pipes, pumps, valves, and heat exchangers in the containment spray system and ECCS during long term emergency cooling operation? Discuss the design bases for these features.

6.5 Your proposed design allows for isolation of safety injection tanks. Discuss your anticipated mode of reactor operation with less than four safety injection tanks in service at any time.

6.6 How does the design of the ECCS provide for the disposition of the nitrogen gas in the safety injection tanks in the event of an accident that activates the safety injection tanks?

6.7 Discuss the design basis for preventive maintenance of ECCS components during plant operation. Describe the design provisions to be made for the inservice inspection of containment spray and safety injection pump seals, valve packing, flanged joints and safety valves during flow tests in order to detect leakage. What design features will be provided to enable periodic determination that the safety injection tank check valves are properly seated and that leakage is within specific limits when the reactor coolant system is pressurized?

6.8 What is the basis for the amount and concentration of the sodium hydroxide in the chemical storage tank? Figure 6.2-1 of the PSAR, shows that the chemical storage tank used for the sodium hydroxide is vented to the atmosphere. Discuss the potential for a reaction of the sodium hydroxide solution in the chemical storage tank with carbon dioxide in the air to form a precipitate. What design features or operating procedures will preclude this possibility? If precipitation did occur during long-term storage, could it cause blockage of the containment spray nozzles?

6.9 Describe the extent to which the design of the containment spray system will enable the operator to take action to prevent the addition of sodium hydroxide to the spray solution, and the information and procedures that will be available to the operator to aid him in making the decision.

6.10 Provide the design criteria and bases for the required horsepower, design life, and materials of fabrication for the containment spray and safety injection pumps.

6.11 What are the design pressures and temperatures for the containment spray system piping?

6.12 Provide the following information with respect to the valves in the ECCS system:

a. An analysis to demonstrate that the design of the system is such that closure of a normally open valve prior to or during that period of time in the course of a loss-of-coolant accident when it is required to be open, cannot occur or would not result in an unacceptable degradation of system performance.

b. Describe the safety injection tank isolation valves and the control circuits that will assure that the isolation valves will remain open if needed for emergency cooling during startup operations.

6.13 Provide the design criteria and their bases for the containment emergency air coolers, including:

a. Design heat transfer data.

b. Operating environment.

c. Qualification of components under accident environment conditions.

d. Condensate drainage.

e. Pressure drops and fan and pump power requirements.

f. Service lifetime.

g. Isolation capability.

7.0 INSTRUMENTATION AND CONTROL

7.1 With regard to the protection systems which actuate reactor trip and engineered safety feature action, provide the following information:

a. A list of those systems designed and built by Combustion Engineering that are identical to those of the Millstone 2 (as documented in the SAR), a list of those that are different, and a discussion of the design differences.

b. A list of those systems and their suppliers that are designed and/or built by suppliers other than Combustion Engineering, and

c. Identification of those features of the design which do not conform to the criteria of IEEE 279 or the Commission's proposed General Design Criteria and an explanation of the reasons for these deviations.

7.2 With regard to the control systems designed by Combustion Engineering, provide the following information:

a. Identification of the major plant control systems (e.g., primary system pressure control, reactor vessel water level control, recirculation system flow control) which are identical to those in Millstone 2, and

b. A list and a discussion of the design differences in those systems not identical to those used in Millstone 2. This discussion should include an evaluation of the safety significance of each design change.

7.3 Discuss the seismic design criteria for the reactor protection system, engineered safety feature systems, and the emergency power system. These criteria should address: (1) the capability to initiate a protective action during maximum seismic accelerations, and (2) the capability of the engineered safety feature circuits to withstand seismic disturbances during post accident operation. Describe the qualification testing requirements that will be used to assure that the criteria are satisfied and the means by which these requirements will be imposed on equipment suppliers.

7.4 What are the criteria and their bases which establish the minimum requirements for preserving independence and redundancy within the reactor protection system, the engineered safety feature initiation and control systems and Class IE *Electrical Systems? What procedures and checks will be provided to assure compliance with these criteria during the design and installation of these systems? The criteria and bases for the installation of electrical cable for these systems should, as a minimum, address:

a. Cable derating.

b. Cable routing in containment penetration areas, cable spreading rooms, control rooms and other congested or hostile environment areas.

c. Sharing of cable trays with non-safety related cables or with cables of the same system or other systems.

d. Fire detection and protection in the areas where these cables are installed.

e. Cable and cable tray markings.

f. Spacing of wiring and components in control boards, panels, and relay racks.

7.5 Provide a description of the instrumentation systems included in your design for remote monitoring of post-accident conditions within the primary containment. Provide an analysis to show that these systems provide redundancy and appropriate wide range information for the full spectrum of postulated accidents.

7.6 Identify all safety related equipment and components (e.g., motors, cables, filters, pump seals) located within the containment building which are required to be operable during and subsequent to any loss-of-coolant or steamline break accident. In what manner do the design criteria for these components take into account the potential effects of exposure to normal and accident environmental conditions? Describe the qualification tests which will be performed on each of these items to assure their availability in a combined high temperature, pressure, humidity and radiation (TID-14844 source term) environment.

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*Class IE electrical systems and design basis events are defined in the Proposed IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations (IEEE-308).

7.7 State the criteria which have been established to assure that loss of the air conditioning and/or ventilation system will not adversely affect operability of safety related control and electrical equipment located in the control room and other equipment rooms. Describe the analysis performed to identify the worst case environment (e.g., temperature, humidity). State the limiting condition with regard to temperature that would require reactor shutdown, and how this was determined. Describe any testing (factory and/or onsite) which has been or will be performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions.

7.8 Describe how reactor protection system and engineered safety feature instrumentation, control and electrical equipment will be identified physically as safety related equipment in the plant to assure appropriate treatment, particularly during maintenance and testing operations.

7.9 Describe the method that will be used in San Onofre 2/3 for periodic testing of engineered safety feature instrumentation and control equipment. We interpret IEEE 279 to require the same high degree of on-line testability for engineered safety feature equipment as is required for the reactor trip system.

7.10 Identify and evaluate the CEA interlocks which are safety related. The evaluation should include the consequences of single failures in the interlock systems.

7.11 The reactor protective system (Table 14.1.3-1) indicates delay times for the various reactor trip parameters. These times vary from 0.4 seconds to 0.9 seconds. What provisions will be made to verify these times in the as-built equipment? What subsequent retesting will be performed during the life of the plant to ascertain that these times are still acceptable? Include a discussion of how the safety margins might be affected if the delay time should change appreciably. Will a surveillance requirement for these delay times be included in the Technical Specifications?

8.0 ELECTRICAL SYSTEMS

8.1 Page 8.2-1 of the PSAR states that a grid stability analysis was performed considering the loss of one or more units at San Onofre. Page 8.2-8 discusses an analysis of various line faults. Provide a summary of a grid stability analysis considering the loss of the largest unit on the grid other than at San Onofre.

8.2 Figure 8.2-1 of the PSAR shows nine transmission lines connecting into the San Onofre switchyard. The text, however, states that the two 138 kV lines are loads and not sources. Figure 8.2-2 shows the four 220 kV lines from the SCE grid reducing to two 220 kV lines on common towers between Viejo and Black Star. Discuss the offsite power sources available to the San Onofre Station in sufficient detail to show that offsite power into the station will not be vulnerable to a fault of the common row of towers between Viejo and Black Star. This discussion should include information concerning any other grid interties at Santiago or Viejo and between San Onofre and the San Diego System.

8.3 Page 8.2-2 of the PSAR states that the SCE - SDG&E systems are normally interconnected, but a discussion of a plug board located in the Unit 1 control room for system separation is presented on page 8.2-8. Please discuss the operation of the two systems in sufficient detail to allow a determination to be made of the safety significance of the plug board and its location in the Unit 1 control room.

8.4 Discuss the San Onofre switchyard in sufficient detail to include:

a. Distance from plant to switchyard.

b. DC sources available in switchyard for circuit breaker operation.

c. Criteria relative to operability of switchyard, including Unit 3 transformers, prior to initial operation of Unit 2.

d. Provisions that will be made to protect the original San Onofre Unit 1 switchyard and transmission lines during construction.

8.5 State the ratings of the reserve auxiliary transformers and relate these to startup, safe shutdown and engineered safety features loads.

8.6 Discuss the potential use of automatic load dispatching within the SCE and SDG&E systems and the potential safety significance on the San Onofre Station.

8.7 Page 8.2-14 of the PSAR states that the diesel generators connected to buses A04 and A06 are loaded to 2650 kW. Discuss the criteria used for sizing the diesel generators. Include in the discussion a justification for the apparent overloading of the 2500 kW continuous rated diesel generators.

8.8 Present a summary of a failure analysis for the fuel oil system for the five diesel generators.

8.9 Discuss the basis for your criterion for providing batteries which are rated for 30 minute operation in the event of the loss of all AC power. Include in this discussion information regarding the ability to maintain the plant in a safe condition and the limiting condition for operation.

8.10 Discuss the basis for providing the fifth diesel generator. Include in the discussion information regarding any automatic switching and show why the switching does not compromise the split bus concept.

8.11 The PSAR does not include any information regarding the startup of a unit without offsite power (black start). Is there any intent to consider black starts for this station?

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.1 Describe the design characteristics, criteria and bases for each of the major components including isolation valves and piping, within the chemical and volume control system. Provide a justification for any components not designated as Class I for seismic design.

9.2 What are the design criteria and bases for the rate at which the chemical and volume control system can add boric acid or negative reactivity into the reactor?

9.3 Provide either a summary of a failure analysis of the chemical and volume control system, or applicable design criteria.

9.4 What are the design criteria for the boric acid batching and makeup tank heaters and the heat tracing for boric acid piping?

9.5 To what extent will the design and fabrication of the regenerative heat exchanger in the letdown line from the primary coolant system exceed ASME Section 3, Class C requirements?

9.6 Describe the design characteristics, criteria and bases for each of the major components within the shutdown cooling system?

9.7 Provide either a summary of a failure analysis of the shutdown cooling system, or applicable design criteria.

9.8 The isolation valves between the shutdown cooling system and the reactor primary coolant system will be interlocked to preclude their being opened when the primary coolant system is above a preset pressure. Discuss the potential for an accident in which these isolation valves are inadvertantly left open when the reactor is started up from a cold, depressurized condition.

9.9 Describe the design characteristics and/or criteria and bases for each of the major components in the Component Cooling Water System and discuss each of the following considerations:

a. If a break occurs in the component cooling water system, how will the operator determine which valves to close to isolate the break? How long will this take? How will the detection for the break and ultimate action taken, affect the performance of the emergency containment cooling system and the safety injection systems?

b. The valves in the two lines joining the component cooling pump suction header and the surge tank are shown (Figure 9.4-1) to be locked open. In the event of a pipe rupture in either of the two component cooling water sybsystems what would prevent loss of all the component cooling water in both subsystems via the surge tank?

c. How would a leak in the component cooling water system, downstream of an emergency air cooler and inside the containment building, be detected? What would prevent loss of all cooling water in the surge tank and subsequent loss of pump circulation? How long would it take to locate and isolate a leak? Where are the controls for the component cooling water system valves located?

d. What would be the potential effect on the performance of the component cooling water system of a leak in the component cooling water/salt water cooling system heat exchanger? How would such a leak be detected?

e. Provide a layout drawing to show how redundant system components will be physically separated or, alternately, the design criteria that will be used relative to physical separation of redundant components.

9.10 Indicate the design characteristics and/or criteria and the bases for each of the major components in the salt water cooling system (SWCS) and discuss each of the following considerations:

a. Provide a P & I drawing of the SWCS.

b. Provide a summary of a failure analysis to verify that no single failure of any component in the SWCS could adversely affect the cooling capability of the component cooling water system.

c. What considerations have been given in the design of the SWCS for protection against flooding, including tsunami? How does this protection assure that a supply of cooling water will be available after the tsunami?

d. What design codes and standards will be used for the SWCS? Specifically indicate any components or structures which will not be designated as Class I (seismic).

e. Provide a layout drawing to show how redundant system components will be physically separated or, alternately, the design criteria that will be used relative to physical separation of redundant components.

9.11 The use of both HEPA and charcoal filters are contemplated within various ventilation systems at San Onofre 2/3. Provide the design bases and criteria for selection of each filter. Include the following considerations:

a. Discuss the potential for post-accident ignition of charcoal filters and describe equipment provided to control such ignition.

b. What user testing will be conducted on the activated charcoal used in the auxiliary building ventilation system to determine that it is activated and meets the design specification? What radiation source term will be used in the design of the filters in the auxiliary building?

c. How will the design of each filtration system provide for periodic testing and inspection so that its availability is not compromised during such testing or inspection?

9.12 Will any ductwork be associated with the containment emergency air coolers or the containment atmosphere charcoal filters? If so, what are the design criteria for this ducting relative to the pressure transients that will occur within the containment during a LOCA?

9.13 The PSAR indicates that the final design of the vent stack has not yet been completed. Provide the design criteria to be used in the final vent stack design including tornado design capabilities and indicate the anticipated location and size of the stack. What would be the effect on Class I (seismic) structures and systems if the stack should fail?

9.14 What portions (ref. pg. B.2-5 of PSAR) of the fire protection system are not Class I (seismic)? Could a failure in any of these portions of the fire protection system result in loss of function of any engineered safety feature?

9.15 Provide a summary of an analysis to demonstrate that the instrument air system is not required for a safe plant shutdown or for any of the engineered safety features to accomplish their function in the event of an accident.

9.16 Will there be a "Service Water System" for San Onofre 2/3 similar to that for Unit 1? If so, provide the following information on this system:

a. Functions.

b. Design criteria and basis.

c. Preliminary design.

d. Any sharing of components.

If there will be no service water system for San Onofre 2/3, how will the functions provided by this system for Unit 1, including service as a backup source of water for the auxiliary feedwater system and the fire protection system, be provided for Units 2/3?

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Since the nuclear steam supply system and the power conversion system are interdependent and are to be supplied by different vendors, describe the technical and organizational relationships and areas of responsibilities of the respective parties involved. Discuss the previous experience and present qualifications of the turbine generator manufacturer in designing and fabricating large, low pressure, saturated steam turbines, including the details of his quality assurance program.

10.2 Provide the following additional information regarding the San Onofre 2/3 turbine-generators:

a. The operating experience with the turbine control system and the degree of redundancy incorporated in the design.

b. The maximum turbine speed obtained following a turbine trip due to an overspeed condition or a sudden loss of load, taking into account all steam volumes, including the volumes in the separators and reheaters.

c. The maximum stresses in the turbine and generator rotors at (1) rated speed, (2) maximum design overspeed, (3) anticipated overspeeds following turbine trip, and the ratio of these stresses to the materials yield and ultimate stresses.

10.3 Provide a summary of an analysis to determine potential trajectories and other characteristics of potential missiles from a failure of any one of the three San Onofre main turbine-generators. Indicate critical target areas, such as containment building, spent fuel pit, control room, diesel generator compartments, battery rooms, and emergency electric busses. What criteria will be used to design the San Onofre plant to withstand the éffects of such turbine missiles?

a. Several recently licensed reactor power plants have provided an additional overspeed prevention system on the turbine generator in order to reduce the probability of a turbine failure due to excessive speed (overstressing). What features will be provided at the San Onofre plant to prevent turbine failures due to excessive speed?

b. Another potential cause for turbine failures is defects in materials. What criteria will be used to assure quality material in turbine fabrication?

10.4 State the capacities of the steam system main safety valves and justify the basis for the selected capacities.

10.5 What is the seismic design classification of the San Onofre 2/3 circulating water (and salt water cooling system) intake and outfall structures and any associated piping or conduit? If not Class I, what is the basis for any other classification? What would be the potential consequences if it were not possible to obtain any flow through one of the intake conduits for Unit 2 or 3?

10.6 Describe the design and proposed method of operation of the tsunami gates within the circulation water system. Include the design bases and criteria used in the design. Do these criteria assume that there will be a warning of an approaching tsunami? If so, what time period is assumed and on what basis? If the design is based on receiving a warning and taking operator action, what would be the potential consequences if this action were not taken?

10.7 Provide the following information regarding the steam generator blowdown system:

a. The maximum rate of blowdown and how the blowdown is processed under both normal conditions and conditions of unusually high radioactivity concentrations and/or leakage from primary to secondary coolant systems.

b. Blowdown appears to be the only means available for controlling the level of radioactivity in the secondary coolant system. If other provisions are available, provide a description of this capability.

11.0 RADIOACTIVE WASTE DISPOSAL AND RADIATION PROTECTION

11.1 Indicate how and to what extent the liquid, gaseous and solid radwaste systems for San Onofre 2/3 will be used to assure that radioactive releases will be "as low as practicable".

11.2 Provide a tabulation comparing the anticipated yearly discharge for each significant isotope to be released from San Onofre 2/3 in both gaseous and liquid effluents with the limits of 10 CFR 20. Describe the basis for these estimates, including the assumptions made with regard to amount of failed fuel, decontamination factors due to holdup, decay, filters, evaporators, and demineralizers to be used for waste treatment.

11.3 Describe the procedure to be utilized to monitor the activity that will be discharged directly as waste to the Pacific Ocean. To what extent will discharges be monitored for each significant nuclide? Under what conditions will a determination of isotopic composition be made? What maximum short-term release concentrations will be permitted? Describe the monitors to be used for gaseous wastes. Will this equipment be adequate to measure the annual average concentrations of Iodine-131 that may be released from the plant?

11.4 Section 11.1.1.4 of the PSAR, indicates that the radwaste system will be designed primarily as a seismic Class II system. Provide a tabulation of all major components, including piping and valves, in the radwaste system and indicate the seismic design classification and the associated concentration of radioactivity anticipated for each. Provide an evaluation of the potential consequences of the simultaneous release of all radioactive materials contained in seismic Class II components. Indicate the maximum amount of radioactivity which could be released to the environment and all other assumptions used in the evaluation.

11.5 It is indicated on Figure 1.3-2 of the PSAR that the radwaste primary tanks, the radwaste secondary tanks, the primary plant makeup water storage tank, and the refueling water storage tank, are all located on pads outside the containment buildings. Appendix B of the PSAR indicates that these components will be designed to Class I (seismic) standards. What are the tornado design criteria to be used in the design of these tanks? What maximum quantity of radioactivity could be contained in each of these tanks? Evaluate the potential consequences of a sudden rupture of each of these tanks.

11.6 Each radwaste primary and secondary tanks will be equipped with diaphragms to prevent loss of tritium by evaporation. What is the basis for this design?

11.7 Describe the methods to be employed for monitoring tritium concentrations prior to discharge from the plant.

11.8 Describe the failed fuel detector including the design capability of the detector and its associated instrumentation with respect to both response time and sensitivity.

12.0 CONDUCT OF OPERATIONS

12.1 Figure 12.2-2 of the PSAR indicates that you propose to have only one Senior Reactor Operator and a total of six men per shift for San Onofre Units 2 and 3. Unless there are special considerations involved, we generally require a Senior Reactor Operator on each shift for each unit and at least 8 men per shift for two units with a common control room. What special design features or other considerations can you identify to justify reduced manning for San Onofre 2/3?

12.2 Provide a chronological summary, by position, of the total training program which will be given each member of the management, technical, operations, and maintenance groups of the San Onofre 2/3 staff. Indicate each instance where no training will be required because existing qualifications are acceptable. Include course content, duration, method of instruction and qualifications and/or organization of instructors. In particular, indicate the extent to which Unit 1 "on the job" training will be utilized to provide PWR operating experience.

12.3 To what extent will the operators for Unit 1 and Units 2 and 3 be interchanged? How will the operator training program provide for this interchange?

12.4 Provide information on the emergency plan for the San Onofre site in accordance with Appendix "E" of 10 CFR 50.

12.5 Indicate to what extent a review has been or will be performed of the plant layout and design to assure that critical equipment necessary for safe operation and/or shutdown is adequately protected from acts of industrial sabatoge.

14.0 SAFETY ANALYSIS

14.1 The loss-of-coolant accident (LOCA) as presented in the PSAR (14.4.2) is based on the modified FLASH-2 computer code which represents a multi-node analysis of the system thermal-hydraulics but does not provide nodes for the core itself. Certain analytical questions have been raised regarding the FLASH-2 formulation; these include the possible omission of significant terms in the energy and momentum balance relationships, and the consequences of heat addition to the core treated as a flow path instead of as a specific region or node. Furthermore, recent independent analyses on other, but similar, systems indicate that substantial periods of virtual flow stagnation may occur in the core during blowdown. These independent analyses used more detailed computer codes in which the core nodes are included.

Submit a discussion of these points, that includes the following:

a. The results of an evaluation of the LOCA using a multi-node analysis for the full spectrum of pipe breaks which, in addition to providing the usual information on clad temperature and system pressure, includes details of indicated coolant flow through the core to fully characterize the core hydraulics during blowdown (i.e., core pressure drops, coolant quality, coolant velocities in the core, the upper plenum, and the lower plenum, and the calculated flows out of the cold leg, the hot leg, and through the core). Identify the heat transfer correlations used for the various phases of the blowdown and the refill period. Specify the range of applicability of the correlations and justify their use explicitly in terms of the core hydraulic parameters. Justify the use of homogenized coolant flow wherever used. Specify the coolant water delivery rate from the safety injection tanks and idenfify the core by-pass flow as a function of time.

b. Identification of the limitations of the multi-node analysis used and your assessment of the adequacy of this code to predict core hydraulic behavior. Include an estimate of the uncertainty or "confidence band" on the calculated core hydraulic parameters in view of the one-dimensional character of the code, the analytical questions on heat addition and possible omissions noted above, as well as the potential flow stagnation in the core during blowdown. Establish a basis for the choice of system nodes from the standpoint of node size and location with particular attention to the core region and vessel. Provide suitable bounding analyses for the appropriate regions of uncertainty.

c. A sensitivity analysis to illustrate the effect of relative system impedances or resistances on the calculated coolant flow through the core. That is, evaluate the influence of changes in resistances in the steam generator, hot legs, pumps, and inter-connecting piping on the core flow and justify the specific impedances chosen for the final analysis. Include an assessment of the use of steady-state resistances under single phase flow for transient two phase flow conditions during the blowdown.

d. An analytical evaluation and sensitivity study of accumulator location, injection pressure, and coolant inventory and the relation of these design parameters to performance margins (e.g., peak clad temperatures) for the range of break sizes.

e. A summary of any additional analytical and experimental work which you plan in order to provide assurance of your ability to predict core thermal-hydraulic characteristics during a blowdown with confidence.

f. Using your best available information and analytical technique, the following information for the loss-of-coolant accident:

(1) The largest loss-of-coolant inlet and outlet break sizes for which assured core cooling is predicted.

(2) The highest core power level for which assured core cooling is predicted.

(3) An assessment of potential flow instability or "chugging" in the core or between the parallel intact loops and its effect on the ability to cool the core.

(4) An estimate of the effect of the break location on the foregoing.

14.2 Indicate all actions that the reactor operator would have to perform or may have to perform in the event of a design basis loss-of-coolant accident and the time periods within which these actions would or might have to be performed.

14.3 To what extent does the upgraded core thermal-hydraulic design for San Onofre 2/3 make the potential consequences of loss-of-coolant flow incidents (ref. Section 14.2.4 of PSAR) more severe than for previous CE designs? What is the confidence level for the results obtained for the loss-of-coolant flow incident analyses? In the analysis of a loss-of-coolant flow due to a seized rotor on one pump, what is the basis for the assumption of complete loss of flow in the affected loop?

14.4 Provide the summary of an analysis to determine whether one or more CEA could be ejected from the core by blowdown forces resulting from a loss-of-coolant accident.

14.5 Analyze the potential consequences of a design basis loss-of-coolant accident during the period when the reactor has been shut down, the primary coolant system is being cooled down, the pressurizer low pressure signal SIAS has been bypassed and the safety injection tanks have been isolated. Consider all potential situations during this period such as when the primary coolant system is at a relatively low pressure (~300 psi) but the safety injection tanks have been isolated and the low pressure safety injection pumps have been valved into the shutdown cooling system.

14.6 Provide an evaluation of the probability and the potential consequences of a failure of the shell of a steam generator. Include a discussion of the following, for a range of assumed break sizes: (a) the adequacy of the steam generator supports to resist the blowdown forces, (b) the extent of the blowdown of the secondary coolant with identification of those components and actions required to limit the blowdown to the assumed maximum volume, (c) the likelihood for concurrent or subsequent failure of steam generator tubes, (d) the response of the reactor core and the primary coolant system (e) the response of the containment, including internal structures, to the secondary coolant blowdown without concurrent steam generator tube failures, and with the concurrent double-ended failure of 5, 10 and 25 tubes (or other parametric equivalent study), (f) the potential for damage to other components due to the jet forces associated with the blowdown, and (g) the response of the emergency core cooling and containment cooling systems.

14.7 Provide a summary and the results of an analysis to determine how many steam generator tubes must fail in conjunction with rupture of a primary coolant system cold-leg pipe to cause steam binding sufficient to prevent emergency core cooling water from rising above the midplane of the core.

14.8 Combustion Engineering has previously indicated that it is performing two studies with respect to reactor protection:

a. Reactor Protection System Diversity (Systematic Failure).

b. Anticipated Plant Transient with Failure to Scram.

Describe the scope of these studies, list the accidents to be considered and provide a schedule when preliminary and final results will be submitted.

14.9 You state, in Section 14.4.2 of the PSAR, that you intend to provide a hydrogen control system, designed to engineered safety feature standards, for the post-accident control of hydrogen in the containment. You state further, that the design of the equipment is expected to be identified by mid-1973. The final design of the system will depend upon the basic assumptions made to determine the rate of accumulation of hydrogen within the containment. We have discussed with your representatives the assumptions that we have concluded should be used to calculate the rate of hydrogen accumulation in order to assure that a reasonably conservative result is obtained. These assumptions are listed in the table below. Is it now your intent to use these, or more conservative assumptions? Provide a detailed justification for each instance where you propose to use a less conservative assumption than that listed in the table below, and an analysis of the resulting decrease in time to attain the control limit established for the hydrogen concentration in the containment atmosphere. Provide a curve of hydrogen concentration in the containment as a function of time for the assumptions listed in the table and for those you intent to use as a design basis. The assumed amounts, surface areas, and corrosion rates for materials generating hydrogen as a result of corrosion should be identified.

TABLE

The parameter values listed in this Table should be used for the purpose of evaluating hydrogen and oxygen gas concentrations in containments and the designs provided to control combustible gases evolved in the course of the accident. These values may be changed on the basis of additional experimental evidence and analyses.

Assumptions

1.	Fraction of fission product radiation energy absorbed by the coolant[1]	(a) Beta (1) Betas from fission products in the fuel rods: 0 (2) Betas from fission products intimately mixed with coolant: 1.0 (b) Gamma (1) Gammas from fission products in the fuel rods, coolant in core region: 0.1[2] (2) Gammas from fission products intimately mixed with coolant, all coolant: 1.0
2.	$G(H_2)$[1]	0.5 molecules/100ev
3.	$G(O_2)$[1]	0.25 molecules/100ev
4.	Extent of metal-water reaction (percentage of fuel cladding that reacts with water)	5
5.	Aluminum corrosion rate for aluminun exposed to alkaline solutions. (This value should be adjusted upward for higher temperatures early in the accident sequence)[3]	200 mils/yr
6.	Fission product distribution model	(a) 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water. (b) All noble gases are released to the containment. (c) All other fission products remain in fuel rods.

7.	(a) Lower hydrogen flammability limit (no steam is assumed to be present)	4 volume percent
	(b) Lower oxygen flammability limit	5 volume percent
	(c) The limits given in (a) and (b) should not be exceeded concurrently.	

1/ For water, borated water, and borated alkaline solutions; for other solutions, data should be presented.

2/ This fraction is thought to be conservative; further analysis may show that it should be revised.

3/ For other materials, equivalently conservative assumptions should be made.

Docket Nos. 50-361
and 50-362

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Mr. Jack B. Moore, Vice President
Southern California Edison Company
601 West Fifth Street
P. O. Box 351
Los Angeles, California 90053

Mr. Carthrae M. Laffoon, Vice President
San Diego Gas and Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

We have received your application for authorization to construct and operate two pressurized water nuclear reactors, the San Onofre Nuclear Generating Station Units 2 and 3 at Camp Pendleton, California. A copy of a related notice which has been transmitted to the Office of the Federal Register for publication is enclosed.

Docket Nos. 50-361 and 50-362 have been assigned to this application. Please refer to these docket numbers in all future correspondence relating to this project.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Federal Register Notice

cc: Rollin E. Woodbury, General Counsel
Southern California Edison Company
P. O. Box 351
Los Angeles, California 90053

Chickering & Gregory, General Counsel
San Diego Gas & Electric Company
111 Sutter Street
San Francisco, California 94104

w/bcc to: H. J. McAlduff, ORO
E. E. Hall, GMR/H
J. A. Harris, PI
R. Leith, OC
J. R. Buchanan, ORNL
T. W. Laughlin, DTIE
A. A. Wells, ASLB

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NOS. 50-361 AND 50-362

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

NOTICE OF RECEIPT OF APPLICATION FOR CONSTRUCTION PERMIT AND FACILITY LICENSE

The Southern California Edison Company, 601 West Fifth Street, Los Angeles, California, 90053 and the San Diego Gas and Electric Company, 101 Ash Street, San Diego, California, 92112, pursuant to Section 104(b) of the Atomic Energy Act of 1954, as amended, have filed an application, dated May 28, 1970, for authorization to construct two pressurized water nuclear reactors, designated as the San Onofre Nuclear Generating Station Units 2 and 3, on the applicants' site located at Camp Pendleton, San Diego County, California.

The site is located on the West Coast of Southern California, approximately 62 miles southeast of Los Angeles, approximately 51 miles northwest of San Diego, and is within the United States Marine Corps Base, Camp Pendleton.

Southern California Edison Company (SCE) and San Diego Gas and Electric Company (San Diego) are joint applicants for the construction permit for the San Onofre Nuclear Generating Station Units 2 and 3. The ownership for the two units will be shared in the proportion of 80 percent by SCE and 20 percent by San Diego. SCE, as project manager for the utilities, will have responsibility for the technical adequacy of the design and construction of the San Onofre plant.

The proposed nuclear power plants which will be located adjacent to San Onofre Nuclear Generating Station, Unit 1, will consist of two pressurized water nuclear reactors, each of which is designed for initial operation at approximately 3390 thermal megawatts with a net electrical output of approximately 1140 megawatts.

A copy of the application is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C.

Dated at Bethesda, Maryland, this day of 1970.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing