3-15-79

Docket Nos. 50-361
50-362

Mr. James H. Drake
Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
P.O. Box 800
Rosemead, California 91770

Mr. B. W. Gilman
Senior Vice President - Operations
San Diego Gas and Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: SER OPEN ITEMS LIST

Our review of your application for operating licenses for San Onofre 2 and 3 has progressed to the point that our Safety Evaluation Report is being prepared. We have prepared a list of the items of safety concern that have not yet been resolved. The list includes items for which your previous response (through Amendment 13) was incomplete, is still under staff review, or has lead to additional staff questions or modified positions to clarify the issue. The questions or modified positions in the latter category are presented in Enclosure 2.

Resolution of most of these open items is necessary before we can complete our Safety Evaluation Report. We request that you address these items in the FSAR as soon as is practical. If you have any questions regarding any of these items, please contact us.

Sincerely,

Robert L. Baer, Chief
Light Water Reactors Branch No. 2
Division of Project Management

Enclosures:
1. Open Items List
2. Additional Questions and Positions

ccs w/enclosure:
See next page

7903300529

MA 4
6D

X27317 OFFICE ➤	DPM:LWR#2	DPM:LWR#2				
SURNAME ➤	HRood:lm	RLBaer				
DATE ➤	3/15/79	3/15/79				

Mr. James H. Drake
Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Mr. B. W. Gilman
Senior Vice President - Operations
San Diego Gas and Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

cc: Charles R. Kocher, Esq.
James A. Beoletto, Esq.
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Chickering and Gregory
ATTN: David R. Pigott, Esq.
Counsel for San Diego Gas
& Electric Company and
Southern California
Edison Company
Three Embarcadero Center, 23rd Floor
San Francisco, California 94112

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

Alan R. Watts, Esq.
Rourke & Woodruff
1055 North Main Street
Suite 1020
Santa Ana, California 92701

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

cc: Mr. R. W. DeVane, Jr.
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Mr. P. Dragolovich
Bechtel Power Corporation
P. O. Box 60860, Terminal Annex
Los Angeles, California 90060

Mr. Mark Medford
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Henry Peters
San Diego Gas & Electric Company
Post Office Box 1831
San Diego, California 92112

Ms. Lyn Harris Hicks
Advocate for GUARD
3908 Calle Ariana
San Clemente, California 92672

Richard J. Wharton, Esq.
4655 Cass Street, Suite 304
San Diego, California 92109

Phyllis M. Gallagher, Esq.
Suite 220
615 Civic Center Drive West
Santa Ana, California 92701

Mr. Robert J. Pate
United States Nuclear Regulatory Commission
P. O. Box 4167
San Clemente, California 92672

ENCLOSURE 1

LIST OF OPEN ITEMS-SAN ONOFRE 2 AND 3

Auxiliary Systems Branch

1. Flood protection. FSAR Section 3.4. Q010.63 (see Enclosure 2).

2. High & moderate energy line break. FSAR Section 3.6. Q112.2 (response incomplete).

3. Storage of Unit 1 fuel. FSAR Section 9.1.2. Q010.62 (see Enclosure 2).

4. Category I fuel pool makeup water system. FSAR Section 9.1.3. Q010.64 (see Enclosure 2).

5. Tests on RCPs without CCW system. FSAR Section 9.2.2. Q010.58 (response incomplete).

6. Condensate storage capacity. FSAR Section 9.2.6. Q010.65 (see Enclosure 2).

7. Auxiliary building ventilation system. FSAR Section 9.4.2. Q010.60 (response under review by staff).

8. Fire protection systems. FSAR Section 9.5.1. Q010.15. (under review by staff).

9. 24-hour Category I capability for power operated atmospheric relief valves. FSAR Section 10.3. Q010.66 (see Enclosure 2).

Containment Systems Branch

1. Thermal analysis, environmental qualification. FSAR Section 3.6.1.2. Q022.60 (see Enclosure 2).

2. Nodalization of subcompartment analysis. FSAR Section 6.2.1. Q022.59 (response incomplete).

3. Mass release through purge lines after LOCA. FSAR Section 6.2.4 Q022.61 (see Enclosure 2).

4. Debris screens in purge lines. FSAR Section 6.2.4 Q022.62 (see Enclosure 2).

Instrumentation & Control Systems Branch

1. Site visit to be conducted by staff.

2. PPS power supply independence. FSAR Section 7.2.3. Q032.32 (response incomplete).

3. Bypass of RPS channel. FSAR section 7.2.4. (under review by staff).

4. Seismic Qualification. FSAR Section 7.8. Q032.4, 032.6 (response under review by staff).

5. Environmental Qualification. FSAR Section 7.9. Q032.5, 032.28 (response under review by staff).

Power Systems Branch

1. Degraded grid voltage protection. FSAR Section 8.3. Q040.66 (see Enclosure 2).

2. Sequencing safety loads on offsite power system. FSAR Sections 8.2 & 8.3. Q040.65 (response incomplete).

3. Qualification of penetrations. FSAR Section 8.3.1. Q040.51 (response incomplete).

4. Environmental qualification. FSAR Section 1.8. Q040.50 (response under review by staff).

5. Thermal overload protection for MOV's. FSAR Section 8.3. Q040.67 (see Enclosure 2).

Mechanical Engineering Branch

1. Reactor internals analysis. FSAR Section 3.9.2.3. Q112.8 (response incomplete).

2. Seismic Qualification Review Team (SQRT) audit. FSAR Section 3.9.2.2 & 3.10. (response to staff letter of 1/11/79 incomplete).

3. Shutdown equipment seismic review/site visit. FSAR Section 3.9.3.1. (under review by staff).

4. Load combination methods. FSAR Section 3.9.3 & 3.9.5. Q112.34 (see Enclosure 2).

Materials Engineering Branch

1. Appendix G exemptions. FSAR Section 5.3 Q121.13 (repsonse incomplete).

2. Appendix H exemptions. FSAR Section 5.3 Q121.14 (response incomplete).

3. Inservice inspection program. FSAR Section 5.2.4 Q121.16 (response incomplete).

Structural Engineering Branch

1. Structural/seismic audit. (response to audit questions incomplete).

2. 2D vs. 3D combination of seismic responses. FSAR Section 3.7.2. Q131.31 (response incomplete).

3. Tendon surveillance requirements. FSAR Section 3.8.1. Q131.33 (see Enclosure 2).

Analysis Branch

1. Core protection calculator (CPC). FSAR Section 4.4. Q221.13 (response incomplete) and 221.18 (see Enclosure 2).

2. Loose parts monitor. FSAR Section 4.4. Q221.17 (see Enclosure 2).

3. DNBR testing of revised FEA design. FSAR Section 4.4. Q221.19 (see Enclsoure 2).

4. SLB & FLB analyses. FSAR Section 15.1 Q222.27 through 222.33 (see Enclosure 2).

Core Performance Branch

1. LOCA & SSE loads on FEA grids. FSAR Sections 3.7 and 4.2.2. Q231.26 (response incomplete).

2. EOL fuel rod pressure. FSAR Section 4.2.2. Q231.25 (response incomplete).

3. CEA guide tube wear. FSAR Section 4.2.2. Q231.32 (response incomplete).

4. Spent fuel surveillance. FSAR Section 4.2.3. Q231.31 (response incomplete).

Reactor Systems Branch

1. Staff RHR position. FSAR Section 5.4.7. Q212.157 (see Enclosure 2).

2. SCDS pipe break. FSAR Section 5.4.7. Q212.155 (see Enclosure 2).

3. LPSI valve position indication. FSAR Section 6.3. Q212.156 (see Enclosure 2)

4. Small break LOCA analysis. FSAR Section 6.3. Q212.151 (response incomplete).

5. HPSI pump reliability. FSAR Section 6.3. Q212.153 (see Enclosure 2).

6. Leakage detection system for 3rd HPSI pump. FSAR Section 6.3. Q212.147 (response incomplete).

7. Sump vortex test report. FSAR Section 6.3. Q212.127 (response incomplete).

8. Amount of fuel failures. FSAR Section 15. Q212.148 and 212.150 (responses incomplete).

9. Boron dilution. FSAR Section 15.4 Q212.152 (see Enclosure 2).

10. RCP shaft break. FSAR Section 15.3.4. Q212.154 (see Enclsoure 2).

Accident Analysis Branch

1. Explosion hazards. FSAR Section 2.2.3 Q312.42 (response incomplete).

2. Gas pipeline hazards. FSAR Section 2.2.3. Q312.36 (response incomplete).

3. Toxic gas isolation of control room. FSAR Section 6.4.2. Q 312.37 (response under review by staff).

4. Leak-off connections for ESF valve stems. FSAR Section 15.2. Q312.39 (response under review by staff).

5. Boron plugging of ESF lines. FSAR Section 15.2. Q312.40 (response under review by staff).

6. Negative atmospheric pressure over spent fuel pool. FSAR Section 15.4 Q312.38 (response under review by staff).

Geosciences Branch

1. Basis for SSE ground motion. FSAR Section 2.5. Q361.33 (response incomplete).

2. Evaluation of offshore "E" fault. FSAR Section 2.5. Q361.34 (response incomplete).

3. Basis for Jack C. West conclusions. FSAR Section 2.5. Q361.35 (response incomplete).

4. Evaluation of regional tectonics. FSAR Section 2.5. Q361.36 (response incomplete).

5. Dewatering well cavities. FSAR Section 2.5. Description of dewatering well demobilization and cavity investigation is incomplete.

Quality Assurance Branch

1. Startup test description and acceptance criteria. FSAR Section 14.2 Q423.26, 423.27, 423.28, 423.30 (see Enclosure 2).
2. Reg. Guides 1.68.2 and 1.108. FSAR Section 14.2.Q423.29 (see Enclosure 2).
3. NARC responsibilities. FSAR Section 13.4.2. Q422.11A (see Enclosure 2).
4. Resumes of plant personnel. FSAR Sections 13.1.3.2 and Appendix 13.1A. Q422.13 (see Enclosure 2).
5. Update QA commitment. FSAR Section 17.2.Q421.2 (see Enclosure 2).

010.0 AUXILIARY SYSTEMS BRANCH

010.62 (9.1.2, 9.1.3, 9.1.4) In Amendment 9, you proposed modifying the Unit 2 and 3 spent fuel storage rack design to increase storage capacity and proposed temporary storage of Unit 1 fuel assemblies in the Unit 2 and 3 spent fuel pools. In Amendment 13, you provided additional design details and a safety analysis for the modified spent fuel storage facility. However, you did not address the effects of the temporary storage of spent fuel from San Onofre Unit No. 1 with respect to maintaining fuel assembly subcriticality and assuring safe handling, cooling and storage in the Unit 2 and 3 spent fuel pool. We need the following information:

1. Provide the results of your criticality calculation using the higher enrichment fuel from San Onofre Unit No. 1, including worst case conditions and assumptions including use of unborated water and fuel placement eccentricity. Confirm that Keff <0.95 will be maintained in the spent fuel pool.

2. State the maximum quantity of Unit 1 spent fuel assemblies that will be stored in the Unit 2 and 3 fuel storage facilities. Also indicate whether the storage facility will reserve sufficient rack space for emergency unloading of a full core of Unit 2 and 3 fuel.

3. Describe the provisions made for safe handling and storage of Unit 1 spent fuel assemblies which have different dimensions than Unit 2 or 3 fuel assemblies, including the following:

a. Describe any spacers of adapters that may be needed to accomodate Unit 1 spent fuel in Unit 2 and 3 storage racks. With spacers and adapters in the racks, demonstrate that the racks still meet seismic Category I requirements and that cooling water flow necessary for spent fuel decay heat removal will be maintained. Provide drawings that show how Unit 1 fuel assemblies will be positioned in the Unit 2 and 3 storage facks.

b. Describe the measures that will be taken to preclude putting Unit 1 spent fuel into Unit 2 and 3 spent fuel locations and vice versa.

c. Describe any modifications to the Unit 2 and 3 fuel handling hoist necessary to safely handle Unit 1 fuel assemblies. Confirm that the fuel handling hoist, while handling Unit 1 fuel, will not exceed the maximum uplift capability of racks.

d. Describe the special fuel handling tools, grapple and hoist may be needed for Unit No. 1 fuel.

e. The storage rack space and location assigned within the Unit 2 and 3 spent fuel pools for temporary storage of spent fuel from San Onofre Unit No. 1.

f. Evaluate the consequence of dropping Unit 1 fuel assembly on Unit 2 and 3 fuel assembly in storage racks and vice versa.

g. You have not evaluated the consequences of dropping or tipping the Unit 1 fuel cask in the Unit 2 or 3 fuel handling facility. Provide the necessary information to demonstrate a postulated dropping of Unit 1 spent fuel cask in the Unit 2 and 3 facilities will not damage any safety related equipment or spent fuel.

4. Discuss the adequacy of spent fuel pool cooling and perification system capacity with Unit 1 fuel in Unit 2 and 3 storage pools, including the following:
   a. Unit 1 spent fuel decay time before they are loaded into Unit 2 and 3 storage pool.
   b. Confirm that the spent fuel pool temperature will be maintained below 140°F, considering the worst schedule and loading combination of refueling batches of spent fuel from the different Units in the storage facilities.
   c. Confirm that the spent fuel pool temperature will be maintained below boiling point, considering an emergency full core unload of Unit 2 or 3 plus the worst case reload batches from Unit 1.
   d. State resulting pool temperatures for the above two heat load conditions.
   e. Before Unit 1 spent fuel can be stored in the Unit 2 or 3 fuel pool, we will require that a seismic Category I makeup water system be provided for the Unit 2 and 3 spent fuel pools, independent of the shutdown cooling system.

5. Demonstrate that Unit 2 or Unit 3 plant systems or portions of systems necessary to support safe storage, handling and cooling of Unit 1 spent fuel in the Unit 2 and 3 handling facilities will be

complete and operational before Unit 1 spent fuel is transferred to the Unit 2 and 3 fuel handling facilities. This information should address features and systems such as construction and installation of the spent fuel pools and storage racks; pool cooling and purification system; fuel cask and fuel assembly handling systems; fuel building ventilation system; fuel building fire protection; and safeguards measures. Also describe the measures that will be taken to preclude any construction activity in the remainder of Unit 2 or Unit 3 from adversely affecting the safe storage, handling and cooling of Unit 1 fuel in the Unit 2 and 3 fuel handling facilities.

010.63 (3.4) (RSP) Your response to item 010.56 is not complete. In Amendment 11, you indicate that the bottom of the diesel building exterior doors are located below the probable maximum flood level and that these doors are not watertight. Also, certain openings through the Safety Equipment Building exterior walls are below the probable maximum flood level without flood protection. These openings are to rooms adjacent to areas containing safety related equipment. We find these areas unacceptable. We require taht you (1) provide watertight doors in the diesel buildings or an acceptable means of flood protection for safety related equipment inside the diesel building and(2) provide flood protection for the openings that are below the probable maximum flood level on the safety equipment building walls, or justify that the safety related equipment inside the interior rooms will not be indirectly flooded as a result of water ingress through the exterior wall openings.

010.64 (RSP) (9.1.3) Your response to item 010.57 is not acceptable. We require that a seismic Category I makeup water system be provided for the spent fuel pool, independent of the shutdown cooling system.

010.65 (9.2.6) (RSP) Your response to item 010.59 is not satisfactory. In Amendment 13, you state that the 150,000 gallon water supply from the Seismic Category I condensate storage tank is sufficient for at least two hours of hot standby and four hours for cooldown of the reactor coolant system to the point that the shutdown cooling system can be used. You also state that the 500,000 gallon non-seismic Category I Condensate storage tank is enclosed behind seismic Category I concrete walls and could be used as a backup water source. However, you did not provide sufficient information to demonstrate that these seismic Category I concrete walls will be sealed against leaks through cracks in the walls that may develop after a postulated safe shutdown earthquake. We require that you provide sufficient water from a seismic Category I source to (1) maintain the plant at hot standby conditions for four hours, and (2) allow sufficient cooldown of the reactor coolant so that the shutdown cooling system can be used, assuming the most limiting single failure. A total of twent-four hours water supply including hot standby time will be acceptable. If you want take credit for the 500,000 gallon non-seismic Category I condensate storage tank as a backup to the required water supply, you must show that the seismic Category I wall that encloses the tank will not develop cracks during the SSE that will result in leakage such that the required water volume is unavailable when needed.

010.66 (RSP) (10.3) The San Onofre 2 and 3 power operated atmospheric relief valves are operated by a non-seismic Category I air supply. The air supply is backed up by a Seismic Category I nitrogen bottle which contains a four and one half hour nitrogen supply for the valves. Handwheels are also installed at the valves for local manual operation. In order to assure operation of the atmospheric relief valves for a 24 hour shutdown time period (See Item 010.65), manual operation of the atmospheric relief valves using local handwheels or additional nitrogen supply must be available after the four and one-half hour seismic Category I nitrogen supply is used. It is our position that you must either (1) demonstrate the capabilityfor satisfactory local manual operation of the atmospheric relief valves and the capability for communications with the control room during the plant startup test program, or (2) provide additional onsite Seismic Category I nitrogen sufficient for a 24 hour period of operation of the atmospheric relief valves.

022.0 CONTAINMENT SYSTEMS BRANCH

022.60 Your response to question 022.58 is not complete. Specifically, additional information is needed regarding Items a and c. For Item a, provide typical sectional diagrams of components and the supporting heat transfer analysis performed to establish interior component temperatures as presented in Figures 022.58-1 through 022.58-11. For item c, furnish justification to support the conclusion that the component cross section provided is the most conservative.

022.61 Provide a calculation of the mass of containment atmosphere released through the open purge lines in the event of a postulated LOCA. Describe and justify the analytical model, major assumptions and input data used in the calculation. Provide the mass of containment atmosphere calculated to leave the containment and justify the conservatism of the mixture (steam/air) content. Justify that the LOCA considered represents the worst case (i.e., small breaks may take longer to generate isolation signals resulting in a larger mass release to the environment).

022.62 The response to 022.53 does not demonstrate that adequate provisions are made to ensure that any debris entrained in the vented containment's atmosphere in the event of a LOCA will not prevent closure of the containment purge system isolation valve. It is our position that the ducting must be capable of remaining intact under accident conditions and that the registers in the ducts must be of sufficiently small mesh size to preclude the passage of debris which could inhibit valve closure. Therefore, either demonstrate that the currently proposed system design meets the above requirements or provide an alternative design which assures that blockage of the purge isolation valves will not occur.

040.0 POWER SYSTEMS BRANCH

040.66 (8.3) Your response to item 040.45 is incomplete. In order to complete our evaluation of your degraded grid voltage protection, we require the delay times associated with the various undervoltage set points. Provide this information.

040.67
(8.3)
(RSP)

Prior to amendment 11 to the FSAR, we found section 8.3.1.1.3.13.E second paragraph acceptable. Specifically, this paragraph provided design criteria equivalent to that found in Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves" position C.2. these criteria required that setpoints be selected such that spurious trips would be precluded. Amendment 11 to the FSAR deleted these criteria. We find the currently documented criteria to be unacceptable. We require either reinstatement of the above referenced criteria or as an alternative (and as provided by the guide in position C.1) the thermal overload protection be bypassed in a manner that meets IEEE Std 279-1971 requirements.

112.0 MECHANICAL ENGINEERING BRANCH

112.34 The response to Q112.33 is not entirely acceptable and requires further clarification: (1) identify the systems and BOP Class 1 piping for which dynamic load responses were combined by the square root of the sum of the squares. (2) For the piping so identified, compare the responses (stresses) combined by both the SRSS and absolute sum methods of load combination to allowable stresses. (3) If, in the algebraic expression contained in your response to Q112.29, the individual responses which comprise the LOCA response do not occur simultaneously, indicate the time interval separating the individual responses.

131.0 STRUCTURAL ENGINEERING BRANCH

131. 33 (3.8.1.7.2) (RSP) It is our position that your proposed deviations from position C.4.2 of Regulatory Guide 1.35 Revision 2 are unacceptable. You must either meet position C.4.2 in its entirety, or propose an acceptable alternate surveillance program, such as that described in Revision 3 of Regulatory Guide 1.35 (to be published) and Regulatory Guide 1.35.1 (to be published).

212.0 REACTOR SYSTEMS BRANCH

212.152 (15.4) Your responses on questions about boron dilution (Q212.18, 212.29, 212.105, 212.124) have not provided sufficient information on the alarms which warn the operator that a boron dilution event is in progress. Clearly specify the control room alarms which alert the operator to such an event, the quality of the alarm instrumentation, and the period of time before recriticality after annunciation of the alarm for modes 2, 3, 4, 5, and 6.

212.153 (6.3) Your response to question 212.118 concerning demonstration of the ability of the HPSI ECCS pumps to operate unattended for extended periods is not satisfactory. Discuss in detail the similarities and differences in design between the HPSI pumps and the boiler feed pumps you reference as comparable. Include in your discussion a comparison of operating environments and justify why the environment of the boiler feed pumps is more severe. Detail the number of stages, types of seals, and methods of lubrication for the boiler feed pumps.

Provide documentation that the boiler feed pumps have run with no maintenance for 5 years as discussed in your response to question 212.118. Also discuss the normal amount of maintenance performed on these pumps (if any).

212.154 (15.3.3) Provide an analysis of a reactor coolant pump shaft break where the break occurs at the most limiting location. Provide a plot of DNBR with time. Indicate the number of fuel pins which have a DNBR of less than 1.19 at any time during the transient. Justify that your initial conditions, including core power are conservative. Describe which trip or trips mitigate this accident.

212.155 (Q212.132) SDCS Pipe Break--Your analysis of an SDCS pipe break in response to question 212.132 did not justify some of the assumptions used in the evaluation. In particular, your assumed initial pressurizer level was ≥60% (indicated). Provide justification for this value or reevaluate the event with a more limiting initial level. Provide an analysis of the SDCS pipe break based on SDCS initiation pressure and temperature conditions (361 psig, 350°F) which will maximize the pipe break discharge rate.

212.156 (Q212.69) LPSI Valve Position Indication--The staff requires that position indication in the control room be provided for LPSI pump suction valves 16"-022-C-173 and 16"-023-C-173.

212.157 (Q212.139) RHR Position--During a February 14, 1979 meeting between the staff and the applicant, the applicant discussed the ability of the SONGS 2 and 3 units to go to SDCS initiation conditions using only seismic Category I equipment. Based on this review the staff requires that manual valve 2"-091-C-334 in the single CVCS discharge line be locked open. The staff requires that a bypass be installed around valve 2HV-9201 in the auxiliary pressurizer spray line to preclude a single failure from preventing cooldown of the pressurizer.

The staff requires that a natural circulation test be performed at SONGS Unit 2 to demonstrate the capability to cool down the plant to SDCS initiation conditions within 7 hours under minimum cooldown capability. This test should also demonstrate the boron mixing attained during natural circulation consistent with assumptions used in the evaluation.

The staff requires a discussion of control room and local instrumentation required by the operator to perform a safe and orderly cooldown of the plant using only seismic Category I equipment.

221.0 Reactor Analysis Section, Analysis Branch

221.17 (RSP) Your response to Question 221.17 addresses the loose parts monitoring system (LPMS) to be provided for San Onofre Units 2 and 3. The response is not sufficient in the areas of seismic and environmental design of the instrumentation. Additional description should include a discussion of the capability of the components inside containment to remain operational following any seismic event up to and including the Operating Basis Earthquake. A discussion should also be provided to address any analyses and/or tests which demonstrate that the system will be adequate for the normal operating radiation, vibration, temperature and humidity environment of the reactor system.

221.18 With regard to the Core Protection Calculator system, we require that the following information be provided:

(1) Identification of the revisions to the Software Specifications CEN-57(A)-P and CEN-58(A)-P made for San Onofre Units 2 and 3; and

(2) The test report for verification of the San Onofre 2 and 3 CPC software

(3) The data base constants and changes to the CPC algorithms, and

(4) Modifications to the proposed technical specifications.

(5) Provide a commitment to (a) implement the final software change procedure approved for the ANO-2 facility in accordance with Appendix B provisions of 10 CFR Part 50, and (b) utilize the services of a qualified computer consultant to provide independent verification that approved changes in the software are properly made. Provide documentation or a reference to documentation describing the final version of the software to which change procedures are to be applied.

221.19 (4.4) (RSP) It is our understanding that San Onofre 2 and 3 will use fuel assemblies with support grids which are thicker and higher than comparable grids for the 16x16 fuel design in ANO-2. Also the grid spacing has been increased relative to the grid spacing for ANO-2 by using one less grid for the bundle. The new San Onofre 2/3 fuel design is not presently described in the FSAR. The effect of these changes in grid design may be to reduce the critical heat flux for San Onofre fuel relative to that for ANO-2 and other plants which use the same grid design as ANO-2. Therefore, provide data to justify the use of the CE-1 CHF correlation for San Onofre or propose another, acceptable, correlation for use on San Onofre 2 and 3.

The test for the effects of bundle corners on CHF which have been discussed informally with CE and SCE are not appropriate to satisfy the data needed to justify the use of the CE-1 CHF correlation for San Onofre.

222.0 Systems Analysis Section, Analysis Branch

222.27(a) The answer to question 222.5 in Amendment 12 is unsatisfactory. The CESEC code (CENPD-107) does not describe the calculational method used for determining the consequences of a steam line break accident. Accordingly, provide the details of your calculational method for evaluating the steam line break accident. Describe and justify all input variables and data transfer between codes used to perform these analyses.

(b) Describe in detail how the core operating limit supervisory system (COLSS) limits the axial and radial power distribution following a steam line break accident with return to power from the hot standby condition assuming a stuck control rod.

222.28 (15.1.1) Steam Line Break Analysis - The Steam Line Break (SLB) analysis presented by the applicant for the three SLB cases (full power with and without loss of offsite power, and hot zero power with loss offsite power) presented DNB calculations for approximately the first ten seconds of each accident. The staff is concerned about the possibility of fuel failures during periods of power increase (approximately 60 seconds for full power with loss of offsite power and approximately 50 seconds for full power without loss of offsite power) and when shutdown margin is at a minimum (approximately 180 seconds for full power with loss of offsite power and approximately 50 seconds for full power without loss of offsite power). Provide a DNBR analysis at these points in time for the hot channel, taking into account the technical specification shutdown margins and a stuck control rod. Justify that the core temperatures used are conservative, including the case where the stuck rod is in the core area near the hot channel which receives cold coolant from the faulted steam generator. Provide the hot channel peaking factors assumed in the analyses. Discuss your method of calculating core thermal-hydraulic parameters (including hot channel pressure drop for closed channel calculations) and your time dependent core power distribution calculations including a stuck rod.

222.29 It is stated in Amendment 12 that the pressure drop in the average and the hot fuel channel are calculated using TORC code. Provide the value of the pressure drop for these two channels. Describe how fuel bundle cross flow was considered in the pressure drop calculations.

222.30 The equation given in response to staff question 222.14 in Amendment 12 does not consider the boron transport time within each node. With a loss of AC power, the transport time in the coolant pipe and reactor vessel downcomer region may be significant. Provide an analysis for the steam line break accident which includes the boron transport delav assuming offsite power and the loss of offsite power.

222.31 The response to staff question 222.15 refers to CENPD-107 for the detailed description of the steam generator model used in the steam line break analysis. We note that CENPD-107 does not sufficiently describe the details of the dynamics of the steam generator during a steam line break accident. Provide a description of the dynamic steam generator model including: (1) heat transfer models, (2) steam generator liquid level, (3) steam generator discharge rates, (4) main and auxiliary feedwater flow rates, and (5) secondary system transient pressure.

222.32 The answer to staff question 222.22 does not satisfactorily describe the methods used in the analysis of feedline break. Provide a complete description of the computer codes and details of the calculational procedure for feedline break analyses, including justification of the input variables and data transfer between codes.

222.33 Provide the details of your model for calculating the discharge rate from the steam generator following a feedline break. It is noted that the assumption of single phase discharge would result in a conservative heatup of the primary system. Accordingly, provide justification that the discharge will be two phase fluid.

421.0 QUALITY ASSURANCE

421.2 (17.2) (RSP) Modify the first sentence in FSAR Section 17.2 to reference the approved version of topical report SCE-1-A (Amendment 2, approved by letter dated November 5, 1978, Walter P. Haass to J. B. Moore).

422.0 CONDUCT OF OPERATIONS

422.11A (13.4.2) (RSP) In our request for additional information dated September 20, 1978, we included the following position, inadvertently numbered 422.11. Please renumber this position 422.11A in your response.

"We do not agree that the NARC review of reports and meeting minutes of the OSRC assures that they review the evaluations of proposed changes to procedures to verify that such proposed changes do not constitute unreviewed safety questions, or proposed changes in procedures which may involve an unreviewed safety question. Therefore, it is the staff's position that you modify the responsibilities of your NARC to include the review of:

1. Evaluations of proposed changes to procedures completed under the provisions of 10 CFR 50.59(a) to verify that such proposed changes do not constitute an unreviewed safety question,

2. proposed changes in procedures which may involve an unreviewed safety question as defined in CFR 50.59(c).

Amend your response to address this position."

422.13 (13.1.3.2, Appendix 13.1A-6) We require the submittal of resumes for all the key members of the plant staff before our review can be completed.

423.0 **INITIAL TEST PROGRAM**

423.26 (RSP) Your response to item 423.20 parts (c) and (d) is not totally acceptable. The NRC Standard Review Plan, NUREG-75/087, Section 14.2, requires that we examine the acceptance criteria presented in the test descriptions to assure that the functional adequacy of structures, systems, and components will be demonstrated. Your response (which describes in general terms how acceptance criteria for startup tests are "typically" determined) does not enable us to complete our review. Modify each of the test descriptions listed below to provide the following information:

(1) The acceptance criteria to be used for Unit 2;

(2) The acceptance criteria to be used for Unit 3 if different from those for Unit 2; and

(3) A description of how the acceptance criteria assure that the plant will operate in accordance with design predictions and will operate within the bounds of your accident analysis throughout plant life. This description should quantify the acceptable range of monitored parameters and describe how this range was determined.

Test descriptions:

14.2.12.83	Isothermal Temperature Coefficient Test
14.2.12.84	Shutdown and Regulatory CEA Group Worth Tests
14.2.12.85	Differential Boron Worth Test
14.2.12.86	Critical Boron Concentrations Test
14.2.12.87	Pseudo Dropped and Ejected CEA Worth Test
14.1.12.90	Unit Load Transient Test
14.1.12.91	Control Systems Checkout Test
14.1.12.93	Turbine Trip Test
14.1.12.94	Unit Load Rejection Test
14.1.12.99	Pseudo Rod Ejection Test
14.1.12.100	Dropped CEA Test

423.27 (RSP) Your description of how your CEA worth acceptance criteria will be applied to Unit 3 (as presented in response to item 423.20) is not acceptable. It is our position that if the follow-on acceptance criteria are not satisfied for the Unit

3 regulating CEA groups, that the worth of the shutdown groups also be measured. It is also our position that if the Unit 2 measured shutdown group worth is used in the Unit 3 shutdown margin calculation, then appropriate measurement uncertainties should be subtracted. Provide a commitment to these staff positions.

423.28 Modify your description of the Unit Load Rejection Test to provide a description of how the generator output breaker is tripped (reference item 423.15) and to provide acceptance criteria for grid frequency and voltage during the transient.

423.29 We have concluded that Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" (Revision 1, July 1978) and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" (Revision 1, August 1977) are applicable for your facility. Modify your PSAR to describe how your initial test program will conform to Regulatory Guide 1.68.2 and Regulatory Positions C.2.a and C.2.b of Regulatory Guide 1.108 or describe how you will provide for equivalent alternative testing.

423.30 Table 14.2-2A indicates that all tests to be performed "post 80% plateau" will be conducted at 80% power. This is not consistent with information presented elsewhere in the FSAR. Modify the table to remove this inconsistency.