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cket Nos. 50-361/50-362

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Mr. B. W. Gilman
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Gentlemen:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATING TO THE STAFF REVIEW OF THE SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

As a result of our review of the Final Safety Analysis Report for the San Onofre Nuclear Generating Station, Units 2 and 3, we find that we need the additional information listed in the Enclosure. Most of the enclosed questions have been informally transmitted to your staff during the past few weeks. If you have any questions about the requested information, please contact us.

We note that three of our previous questions were inadvertently mis-numbered. To correct this, we request that you re-number the first three 040-series questions in our July 28, 1980 letter to be 040.77, 040.78, and 040.79.

We have completed our review of your emergency plan submittal dated August 1980, which relates to improvement of emergency preparedness associated with the San Onofre Nuclear Generating Station. Based on our review, we conclude that your revised emergency plan meets the present requirements of 10 CFR 50, Appendix E, and the regulatory positions of Regulatory Guide 1.101 and NUREG-0610. Your plan was further reviewed against the criteria stated in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Plans and Preparedness in Support of Nuclear Power Plants" which addresses the planning standards set forth in the revised 10 CFR 50.47 (45 FR 55402 August 19, 1980) which becomes effective November 3, 1980; however, our review indicated that additional information and commitments are required before we can conclude that your onsite emergency preparedness program meets these criteria. Our comments on your program are covered in the 432-series questions in the enclosure. Your emergency plan should be revised to address these comments in accordance with the new rule and the schedule outlined in 10 CFR 50.

As stated in paragraph 50.47(a)(2), of the new rule, the NRC will base its findings on the adequacy of your emergency plan on a review of the Federal Emergency Management Agency (FEMA) findings and determinations as to whether State and local emergency plans are adequate and capable of being implemented, and on FEMA's assessment as to whether the applicant's onsite emergency plans are adequate and capable of being implemented. In addition, an emergency response exercise with State and local government designed to test the integrated capability of the emergency preparedness plans must be conducted before issuance of an Operating License.

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In our view, your August 1980 emergency plan reflects improvement over your previous plan and gives a greater margin for public health and safety. You should therefore begin to implement this revision.

For additional clarification of our criteria regarding TMI-related requirements, please see NUREG-0737, "Clarification of TMI Action Plan Requirements".

Sincerely,

/s/

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

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In our view, your August 1980 emergency plan reflects improvement over your previous plan and gives a greater margin for public health and safety. You should therefore begin to implement this revision.

Sincerely,

Robert L. Tedesco, Assistant Director
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In our view, your August 1980 emergency plan reflects improvement over your previous plan and gives a greater margin for public health and safety. You should therefore begin to implement this revision.

Sincerely,

Frank J. Miraglia, Acting Chief
Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation

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NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369 cc: w/enclosure

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001.0 LICENSING BRANCH #3

001.1 Provide an evaluation which demonstrates that San Onofre 2 and 3 comply with each of the regulations contained in Title 10, Code of Federal Regulations, Parts 20, 50, and 100. Any areas of non-compliance with these regulations should be identified and justified.

010.0 AUXILIARY SYSTEMS BRANCH*

010.67 (9.1.3) For the maximum spent fuel storage case involving storage of 800 assemblies, it is stated in the FSAR that, with two spent fuel pool trains in service, the maximum fuel pool temperature would be 150°F. Standard Review Plan (SRP) 9.1.3 states that the pool temperature should be kept at or below 140°F for the maximum heat load with normal cooling systems in operation. Therefore, demonstrate that fuel pool operation at 150°F for extended time periods will not result in degraded safety conditions due to the effect of the higher temperatures on the effectiveness of the spent fuel pool cleanup system ion exchanger and filter, the effect on fuel handling building ventilation systems including the charcoal filters, and the effect on operator access to the spent fuel storage facility to perform safety related operations.

010.68 (9.1.3) For the maximum fuel storage case discussed in request 010.64, assuming failure of one fuel pool cooling train, state what the pool temperature would be and its effects on safe fuel pool operation, considering the following alternatives:

a. Backup cooling systems are utilized. In this connection, provide the flow paths utilized and demonstrate the feasibility of utilizing these systems.

b. Backup cooling systems are unavailable.

*Unless otherwise specified, the numbers in parenthesis beneath the question numbers refer to the FSAR section that the question applies to.

010.69 (9.1.4) The FSAR does not contain sufficient information to demonstrate that a spent fuel cask drop accident caused by a failure of the cask handling system cannot result in unacceptable conditions because of damages to the spent fuel or excessive spent fuel pool water loss. Utilizing the guidelines in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" July, 1980, including the analysis methodology in Appendix A, provide the results of an analysis that, along with detailed drawings and sketches as necessary, demonstrates either that such an accident is very unlikely or that the consequences are within allowable limits.

010.70 (9.2.2) In our request 010.13 regarding the adequacy of the component cooling water system (CCWS) and related instrumentation systems to provide assured cooling of the reactor coolant pump (RCP) seals and motor bearings, it was our position that if the CCWS supply and return lines for the RCPs did not meet the single failure criterion, we would require that you demonstrate that the RCPs could operate for about 30 minutes without the loss of function, and that safety grade instrumentation must be provided to detect the loss of CCW to the RCPs and to alarm the operator in the control room. The entire instrumentation system, including audible and visible status indicators for loss of CCW must meet the requirements of IEEE Standard 279-1971/1974. Therefore, demonstrate the adequacy of these instrumentation systems to effect safe shutdown in the event of CCWS failure.

010.71 Provide the design bases and characteristics for the main steam isolation and relief valve enclosure blowout panels, and demonstrate their effectiveness for (1) enclosure overpressure protection and (2) tornado missile protection.

010.72 The steam supply line to the auxiliary feedwater pump turbine drive is routed in a trench covered with heavy grating. Since this pump is needed for safe shutdown, provide the results of an analysis which demonstrates that this steam line will not be incapacitated by tornado missiles.

010.73 It is our position that the FSAR contain a statement to the effect that the exhaust air from the fuel pool area be routed through the clean-up filters whenever fuel handling operations are in progress in this area (Reference: Standard Review Plan 9.4.2). Therefore, revise FSAR Section 9.4.3.1, "Fuel Handling Building Ventilation System" to incorporate this statement.

015.0 FIRE PROTECTION SECTION, CHEMICAL ENGINEERING BRANCH

015.45 Your response to Q015.3 was incomplete. Verify that all HVAC wrap and piping insulation have a structural base of noncombustible material (Item 7a) and a potential heat value not exceeding 3500 Btu/lb. in the form in which it is used. Also verify that all interior finishes have a flame-spread rating of not greater than 25 on any surface that would be exposed by cutting through the material on any plane. Identify any materials which do not comply with these NFPA 220 criteria for limited combustibility material and their estimated weights.

015.46 With regard to your response to Q015.4, submit the results of your pressure surge analysis of your fire water system, along with evidence that the valve manufacturer concurs with any finding that there would be no effect on the deluge valve, including the possibility of a severe water hammer opening fire protection valves not tripped by fire detection systems.

015.47 Provide an automatic water suppression system for Room 425 of auxiliary building, EL. 70'-0".

015.48 It is our position that, in addition to the cable tray deluge system, you must provide an area water suppression system in the cable spreading rooms and cable riser galleries (Zones 12, 29, 30, 41, 42, and 67) to protect against exposure fires. Reference Q015.7a(1).

015.49 Your response to Q015.7a(2) is incomplete. Verify that you have evaluated the hydraulic capability of the water system given simultaneous operation of adjacent fire suppression systems in areas not separated by fire-rated barriers, e.g., cable tunnels and the cable spreading room.

015.50 Indicate the size of the fire truck pump and water tank, and the location where the truck will be housed. Provide a diagram showing the existing system layout and proposed modifications, including routings through the plant and locations of all hose connections. Reference Q015.9.

015.51 Provide diagrams of containment indicating circuit routing for safe shutdown systems. The drawings should identify the circuit functions and clearly indicate each circuit's physical relationship to other safe shutdown circuits. Reference Q015.12.

015.52 Since your one-hour fire rated walls were tested and found to be acceptable as two-hour fire rated walls and given that one-hour rated fire walls did not have to have fire rated dampers per NFPA 90A, verify that all newly defined two-hour duct penetrations of safety-related area barrier walls are provided with listed fire dampers. Indicate all locations where duct penetrations are not provided with rated fire dampers, or where less than a three-hour rated fire damper is provided in the penetration of a three-hour rated barrier. Also, for those areas where your FHA identified walls as one-hour rated walls, and your subsequent tests have demonstrated a two-hour rating, verify that the one-hour doors will be upgraded to coincide with the two-hour wall ratings.

Indicate the material used for penetration seals and reference a specific design or test method used to qualify the seal for its stated fire rating. Verify that anchored angular steel or other supports will be installed at penetration seals in a manner similar to that used in any tested assemblies. Reference Q015.15.

015.53 It is our position that breathing apparatus for fire brigade use be reserved only for fire brigade use. Additional units should be provided for other plant personnel. Verify that a minimum of five self-contained breathing units will be maintained for the exclusive use of fire brigade members during a fire emergency. Reference Q015.17.

015.54 The underground fire water system has insufficient valves to isolate hydrant laterals from essential interior suppression systems. We require that hydrants numbered 1N, 2N, 7N, 8N, 1S, 2S, 3S, and 8S be equipped with isolation valves to avoid the possibility of having important interior fire suppression systems being put out of service because of hydrant maintenance. Reference Q015.22.

015.55 It is our position that, because of the potential fire exposure to the control room, an automatic suppression system be provided for the turbine lab area, the instrument repair areas, and the storage areas in the control room support area. In addition, all other control room support areas should be provided with automatic fire detection. Reference Q015.25.

015.56 It is our position that you provide an engineered oil containment and collection system for the reactor coolant pumps to protect against a pressurized oil spray igniting and affecting other safety related equipment or pumps. The installation must satisfy Reg. Guide 1.29, paragraph C.2. Reference Q015.30.

015.57 It is our position that you provide standpipe hose stations for all areas of the plant, including Zones 28 and 45, in accordance with NFPA 14 requirements. Reference Q015.31.

015.58 It is our position that Zone 30 of the electrical tunnels be provided with standpipe hose stations in accordance with NFPA 14 requirements, considering a maximum of 100 ft. of hose per hose station. Reference Q015.41.

015.59 Revise the combustible loading calculations given in the FHA to include the cable loadings which you indicate are in the zone. Reference Q015.43.

015.60 It is our position that all areas which contain redundant safe shutdown systems which are not separated by three-hour fire rated barriers should be provided with an automatic, wet-pipe sprinkler system designed to cover the entire area as well as an early warning smoke detection system. In addition, to allow for possible thermal lag or failure of the suppression system, in those areas where the redundant systems are separated by less than 20 ft. of clear, open air space, an ASTM #E119 rated fire barrier which will completely enclose one of the redundant systems should be provided. The barrier should protect the circuit integrity/equipment availability of that system for one hour under fire test conditions. Areas where such protection is required include the following fire zones:

12 Cable Riser Gallery

13A Emergency HVAC Unit Room 309A

15 Rooms 308A and B, ESF Switchgear Rooms

22 Auxiliary Feedwater Pump Room

23 Spent Fuel Pool Heat Exchanger Room

29 Cable Riser Galleries

30 Electrical Tunnel Elev. 30'-6"

32B Fan Room - 233, 234 - Train B

36 Spent Fuel Pool Pump Room

42 Cable Riser Galleries

44 Intake Structure

48 CCW Heat Exchangers and Piping Rooms, Elev. 8'-0"

63 Corridor, Elev. 50'-0", Control Building

67 Cable Riser Galleries, Radwaste Area, Elev. 63'-6"

72 Corridor 442, Elev. 70'

78 Corridor Room 105

83 Salt Water Cooling Tunnel, Train A, Train B

84 Safety Equipment Building, Elev. 8', A/C Room No. 017

In lieu of the one-hour fire rated barrier, an alternate shutdown system can be provided.

Where safe shutdown capability cannot be assured by barriers, suppression and detection systems, it is our position that an alternate shutdown system should be provided. Such areas include the following fire zones:

5 Cable Riser Gallery

31 Control Room Complex

41 Cable Spreading Room

The alternate shutdown system should be completely independent of the area for which it is being provided such that a fire in either area which damages redundant systems will not affect the shutdown capability from the other area. Reference Q015.44a.

015.61 Your response to Q015.44b is adequate for the concern regarding the control room and cable spreading room separation from the remote shutdown panels. However, you have not addressed remote shutdown for loss of circuits in the areas identified in Question 015.44a. It is our position as stated in Question 015.44a that alternate shutdown systems be provided for areas of the plant in addition to the control room and cable spreading room.

In addition, you have not demonstrated that adequate personnel will be available to perform the necessary shutdown functions in addition to the five man fire brigade. This should be demonstrated for all shifts at minimum staffing levels. Also indicate how communications would be established between the remote shutdown locations to coordinate emergency shutdown procedures.

## 031.0 EQUIPMENT QUALIFICATION BRANCH

031.1 The following request relates to the environmental qualification information provided for the 600 volt power cables, 480 volt load and motor control centers, diesel driven electrical generating sets and containment building fan motors.

a. Identify the qualified life, for each of the six items, if less than 40 years, provide the documentation method and the reporting plan for replacement after the qualified life.

b. Clearly state the acceptance criteria for the environmental qualification for each of these items.

031.2 Provide the following information for the 480 volt load centers, 480 volt motor control centers and the diesel driven electrical generating sets.

a. Provide the equipment qualification plans as outlined in Section 5.3 of IEEE Standard 323-1971 (Refer to Table 040.50-1 and Section 3.11-2 of the FSAR). The use of previous operating experience and history may be acceptable for environmental qualification, however, this information must be complete (especially with regard to service conditions and equipment performance) and presented in an auditable form.

b. Provide a date by which the environmental qualification test results will be available for these items. Also, if this date is subsequent to the expected plant operation date provide an interim bases for plant operation.

031.3 State the complete model and/or manufacturers identification number(s) for the 600 volt power cables and the containment building fan motors.

031.4 Address the following which relate to the environmental qualification information provided for the 600 volt power cables.

a. Justify the use of these cables for the San Onofre Station since the test results provided show that when these cables are thermally and radiation aged they have substantial deterioration of the jackets which in one case was repaired before the steam and chemical spray was applied and again during the high potential test (Refer to Cable B10.). Further, provide justification for not maintaining an electrical load on Cable B10 throughout the steam and chemical exposure test.

b. Provide supporting data which clearly indicates that the LOCA environmental qualification conditions equals or exceeds the maximum calculated MSLB environmental qualification conditions.

031.5 Provide information which clearly states that the $10^6$ Rads documented in the FSAR is enveloped by the qualification plan for the diesel driven electrical generating sets.

031.6 Provide the following information for the Containment Building Fan Motors.

a. In addition to the qualification parameters (i.e., thermal aging, seismic testing, LOCA testing, etc.) provide the test results of the same type or a similar type motor that uses the insulating materials listed in the Joy Report X-604 subjected to radiation aging (cumulated dose 5 x $10^7$ Rads plus margin as stated in the FSAR).

b. Identify the measured motor insulation resistance before the LOCA testing and justify the acceptability of this motor since the motor insulation resistance was zero after testing. Also, state the acceptance criteria for the insulation resistance of this motor and identify the fan motor electrical loading (to include margin) during the LOCA testing.

c. Explicitly identify where the environmental qualification testing was completed considering only LOCA environmental conditions and provide supporting information which demonstrates for any such case that the LOCA environment exceeds or are equivalent to the maximum calculated MSLB conditions.

d. Provide supporting information which clearly indicates that the design and testing conditions for this fan motor envelopes the worst case environmental conditions in the containment.

031.7 Address the following items which relate to the transmitters.

a. Provide the test report for the transmitters in the balance of plant list that could be subjected to the limiting harsh environmental conditions in the plant. If this transmitter is to be associated with the auxiliary feedwater flow indicator then clearly state that it is environmentally qualified to $10^6$ Rads as indicated in the FSAR.

b. State more precisely the installed plant location and define the normal and accident environmental conditions to which the transmitter is to be qualified.

c. Identify the installed and service life of the transmitter and any component part for which the service life is less than the installed life. Also, if the installed and/or service life of this transmitter is less than the 40 year design life, provide the documentation method and the reporting plan for replacement of the transmitter or appropriate component parts after their service life.

031.8 For the Electric Motor Valve Actuators, state the acceptance criteria for the valve actuator switch contact chatter and verify that this equipment satisfies this acceptance criteria.

032.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

032.39 Section 7.3.1 of the FSAR states that the discharge valves of the emergency feedwater system are automatically clo sed to secure excess feedwater flow when the steam generator water level returns above the low level set point. Provide a detailed description of the operation of these valves, including logic and electrical schematic diagrams. Identify all valves involved in this operation.

## 112.0 MECHANICAL ENGINEERING BRANCH

112.41 Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

a. Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

b. Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

112.42 There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance and etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

121.0 MATERIALS ENGINEERING BRANCH

121.28 Provide the following information regarding the reactor containment pressure boundary:

a. Identification of the fabrication codes (edition and addenda) and the specific paragraphs in these codes that specify the fracture toughness requirements and acceptance criteria (for weldments and base metals). Codes and code paragraphs should be identified for all materials which constitute part of the containment boundary (e.g., piping penetrations, personnel airlocks, equipment hatch).

b. The materials test data that certify that the fracture toughness acceptance standards have been met for each of the identified materials in the containment pressure boundary.

c. Lowest service metal temperature of reactor containment pressure boundary materials.

d. As-built dimensions and materials of construction of flued head of hot line penetration shown in FSAR Figure 3.8-11.

121.29 Provide the following information for each LP turbine:

a. Turbine type

b. For each disc:

(1) type of material including material specifications
(2) tensile properties data
(3) toughness properties data including Fracture Appearance Transition Temperature and upper energy and temperature
(4) keyway temperatures
(5) critical crack size at operating and design overspeed
(6) crack growth rate
(7) calculated bore and keyway stress at operating and design overspeed
(8) calculated $K_{1c}$ data
(9) minimum yield strength specified for each disc

121.30 Question deleted.

121.31 Indicate the turbine discs that will have sufficient moisture in the hub to cause a propensity for stress corrosion cracking.

121.32 Indicate whether an analysis and evaluation regarding turbine missiles have been performed for your plant and provided to the staff. If such an analysis and evaluation has been performed and reported, please provide appropriate references to the available documentation. In the event that such studies have not been made, consideration should be given to scheduling such an action.

121.33 The staff position concerning IWC-1220 exemption criteria, as permitted by the 1974 Edition of Section XI, for Class 2 welds in the emergency core cooling system, the residual heat removal system, and the containment heat removal system, is that a representative sample of welds in these systems must be subjected to inservice volumetric and/or surface examinations. Welds in these safety related systems cannot be completely exempted from volumetric or surface inspection based upon the requirements of 50.55a(b) in 10CFR50, General Design Criteria 36 & 39, and the Summer 1978 Addenda to the 1977 Edition of Section XI. Your ISI program should include a representative sampling of welds and the proposed methods of examination for the ECCS, RHRS, and CHRS welds previously exempted for chemistry control, pressure/temperature conditions, or line size. Identify the lines and welds exempted from examination in the preservice inspection by IWC-1220 criteria.

121.34 The preservice inspection program lists Class 1 components exempted from examination by IWB-1220 of Section XI, 1974 Edition including Addenda through Summer 1975. Provide the calculations and assumptions made in determining line sizes exempted under IWB-1220(b)(1) based on reactor coolant makeup capacity.

121.35 The San Onofre 2 & 3 PSI program indicates that steam generator and pressurizer nozzle to vessel welds and branch pipe connection welds on lines exceeding 6 inches in diameter will not be examined to the full extent required by the code due to inaccessibility and geometry. Provide the following additional information for our evaluation:

a. The identification of each weld for which this relief request applies.

b. The percentage of the code required examinations performed in the preservice inspection.

c. The construction code examinations performed on these welds.

d. Any supplemental or alternative examinations.

We will require that all areas in the branch pipe connection welds which were not subjected to a volumetric examination be examined by a surface method.

121.36 Standard Review Plan 3.6.1 requires that 100% volumetric examination of high energy fluid system piping welds between containment isolation valves be completed each interval. These augmented inservice inspection requirements exceed Section XI requirements. In order to evaluate the degree of compliance with the augmented ISI requirements in SRP 3.6.1, we require the following information:

a. Describe the preservice examinations performed on these welds.

b. Provide a list of the welds in high energy fluid system piping between containment isolation valves that are not being completely examined and a technical justification.

121.37 The PSI program states that ultrasonic examinations of components not covered by Appendix I of the 1974 Edition of the code or Appendix III of the 1977 Edition will have indications greater that 50% of the reference level recorded. The governing specifications for these components is Article 5 of Section V of the ASME Code, which specifies that indications greater than 20% must be investigated. Provide the justification to support this deviation from the code in a relief request.

Bolting examination requirements in the 1977 Edition through the Summer 1978 Addenda of the code for your preservice inspection program must meet all of the requirements in the later Edition and Addenda.

121.38 Evaluation of examination results is covered by Articles IWC-3000 and IWD-3000 in the code for Class 2 and 3 components respectively. However, both of these Articles are in the course of preparation. Indicate the alternative evaluation procedures you propose to use.

121.39 Supply impact energy data for both the transition and upper shelf energy regions for the following weld seams:

a) 3-203A, 3-203B, 3-203C, and 9-203 of San Onofre Unit No. 3, and

b) 9-203 of San Onofre Unit No. 2.

121.40 Identify all reactor vessel beltline weld seams and weldment test specimens by the following:

a) weld wire and heat number,

b) flux and lot number, and

c) welding process.

If weldment test specimens were not taken directly from excess vessel shell course materials and welds, identify, in addition to the above, the base metal combinations.

121.41 Revise Tables 121.24-1, 2, 3, and 4 to include identification of the reactor vessel beltline weld seam that the surveillance program weld metal represents.

121.42 Identify the orientation of the Charpy V-notch test specimens (Table 121.26-1) used to establish the upper shelf energy levels of the pump flywheel plate material.

121.43 As required by Paragraph C.1.c of Safety Guide 14, demonstrate that the minimum fracture toughness of the flywheel plate material, ASTM 543, Grade I, Type B, is equivalent to a dynamic stress intensity factor ($K_{IC}$ dynamic) of at least 100 ksi $\sqrt{in}$ at the normal operating temperature of the flywheel by either 1) justifying that the normal operating temperature is 212°F (Table 121.26-1) or 2) that the material has greater than 50 ft-lbs absorbed energy at the normal operating temperature.

212.0 REACTOR SYSTEMS BRANCH

212.159 In response to staff's request 010.58 concerning the possible complete loss of CCW supply to the RCPs due to a postulated single active failure in the CCW supply system, you stated that tests have been completed to demonstrate that the San Onofre Units No. 2 and 3 RCPs are capable of continued operation for a minimum of 30 minutes without shaft seizure or excessive seal leakage if CCW is lost. We have reviewed your test results presented in the FSAR, in ASME paper No. 80-C2/PVP-28 and in the Byron-Jackson test report GS-1520 and find that we require additional information for evaluation as follows:

a. The San Onofre 2 & 3 design incorporates a single CCW supply and return line to all four RCP motor bearing and pump seal heat exchangers in each unit. A single active failure of the isolation valves in either the CCW supply line or the return line will cause complete loss of CCW supply to both motor bearing and pump seal heat exchangers of all four RCPs. Your tests were performed separately for the motor bearing and pump seal on loss of CCW supply. Explain why your tests did not simulate a complete loss of CCW supply to all heat exchangers simultaneously. Justify that the results of the separate tests for the motor bearing and pump seal are applicable to the integral effects which would result from a postulated complete loss of CCW.

b. The test report does not address the pump motor speed during the tests. State the motor test rpm and confirm that these motor speeds are compatible to the rpm associated with the design RCP flow or otherwise justify why the test results are applicable.

c. Describe whether the motor was tested under "load" or "no load" conditions. Explain how the test results are applicable to a loss of CCW under normal plant operating conditions.

d. Describe the RCP inlet and discharge pressures during the tests and confirm that these pressures are compatible to normal plant operating conditions or otherwise justify why the test results are applicable.

e. The test report indicates that pump shaft vibration was noted during the tests. Explain why prolonged vibration (~30 minutes) is acceptable.

212.160 During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

a. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

   This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

b. Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.

c. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

d. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

e. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation.

(1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

(2) Provide a drawing showing the location of the drain sump relative to the containment sumps.

(3) Provide the following information with your evaluation of debris:

(a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

(b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

(c) For each type of thermal insulation used in the containment, provide the following information:

(i) type of material including composition and density,

(ii) manufacturer and brand name,

(iii) method of attachment,

(iv) location and quantity in containment of each type,

(v) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

(d) Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

212.161 As the result of our review of your response to our question 212.127 and the "Final Report on Hydraulic Model Studies of Containment Emergency Sump Recirculation Intakes" for SONGS 2 & 3, we have the following specific questions:

a. What is the influence of north sump operation on south sump performance? Flow straightening by trash racks does not resolve concerns associated with resultant flow stratification.

b. Are there any high pressure pipes in the vicinity of the sumps; if so, how is jet impingement accommodated by the sump design?

c. Are there any drain holes in the ceiling in the vicinity of the sumps; if so, how was the potential for air entrainment accommodated in the design?

d. Address the influence of flow path "C" on the north sump; why isn't the north sump modeled when a failure of pumps in the south sump could lead to counterclockwise rotational patterns from paths B, C and D in the north sump? If this is because of symmetry, show that the tests envelop rotational velocities.

e. Section 5.2 of the sump pump test report indicates that the NPSH required for the spray pump is 24.0 ft. The data you provided in response to our question 212.133 show that the NPSH required for the spray pump is 13.0 ft. Clarify the discrepancy and confirm that all HPSI pumps and spray pumps have sufficient margin in NPSH during the recirculation mode.

212.162 In your response to question 212.157, you have agreed to perform a natural circulation test to demonstrate the capability to cool down to SDCS initiation conditions within 7 hours under minimum cooldown capability. This test will also verify that adequate boron mixing can be achieved using natural circulation. We request that you submit the details of your test procedure for review. We also request that you address the prototypicality of this test to a natural circulation cooldown from full power conditions. In particular, you should address the capability to cooldown to SDCS conditions in 7 hours in light of present knowledge regarding the St. Lucie cooldown event. (They are presently recommending cooldown rates to SDCS conditions in excess of 7 hours in order to avoid vessel voiding.)

212.163 At a meeting on August 15, 1980, the staff informed you that your response to question 212.152 was unsatisfactory. The Standard Review Plan (NUREG 75/087) Section 15.4.6 requires that redundant alarms not subject to a single failure be provided to alert the operator of an unplanned dilution event. The staff requests that you describe in detail the redundant alarms which will signal an unplanned dilution during all modes of operation including cooldown.

212.164 The staff has reviewed the shutdown cooling system design of San Onofre 2 and 3 for compliance to Reactor Systems Branch Technical Position 5-1 (as to be implemented for Class 2 plants). We have concluded that your present design does not meet that part of BTP 5-1 which requires the operator to be able to bring the plant from normal operating conditions to SDCS entry from the control room. It is our understanding that at least nine (9) valves in the SDCS train need to be manually repositioned from outside of the control room in order to realign from the safety injection to the SDC mode of operation.

It is the staff position that the SDCS design of San Onofre 2 and 3 be revised to comply with the above. We request that you submit the appropriate documentation of your design revision for staff approval prior to installation. Included in your submittal should be an evaluation which demonstrates that the modifications made do not significantly reduce the reliability of ECCS.

Because of the extent of the modifications necessary for compliance, we do not require that compliance be completed prior to your scheduled OL issuance. Rather, we will accept an extended schedule for completing the necessary design revisions. We propose that an acceptable schedule for completing the necessary design revisions is by the end of your first refueling outage.

Your response should acknowledge your acceptance of the staff position and either the acceptability of our proposed implementation schedule or a justifiable alternate schedule.

212.164 Your response to TMI-related requirement item II.B.1 is not sufficient. Provide all necessary information for your proposed Reactor Coolant System Vents including a detail system description, results of analyses, P&IDs, operating procedures and technical specifications as required in the attached clarification for this item.

## II.B.1 REACTOR COOLANT SYSTEM VENTS

### Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:*

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

### Changes to Previous Requirements and Guidance

(1) The probability of a valve failing to close, once opened, should be minimized.

(2) Establishes environmental qualification (Commission Order, May 23, 1980).

(3) Establishes provisions for testing.

(4) Delete requirements of September 27, 1979 letter from Vassallo to applicants stating that vents shall be safety grade and shall satisfy single-failure criteria of IEEE-279. Vent systems are not required to have redundant paths. A degree of redundancy should be provided by powering different vents from different emergency buses.

(5) Documentation date changed to July 1, 1981 and implementation date to July 1, 1982.

Clarification does not change NRC concept of requirement, but provides more detail on scope. The dates have been revised to provide time for procurement and installation.

*It was the intent of the October 30, 1979 letter to delete the requirement to meet the criteria of 10 CFR 50.44 and SRP 6.2.5 for beyond-design-basis events. The analysis requirements of Position 2 in the September 13, 1979 letter are therefore unnecessary.

Clarification

A. General

(1) The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling.

(2) Procedures addressing the use of the reactor coolant system vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant being able to maintain core cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10 CFR 50.44 or 10 CFR 50.46.

(3) The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which may be considered, is to specify a volume of noncondensible gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

(4) Where practical, the reactor coolant system vents should be kept smaller than the size corresponding to the definition of LOCA (10 CFR 50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed. with the vent

(5) A positive indication of valve position should be provided in the control room.

(6) The reactor coolant vent system shall be operable from the control room.

(7) Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary, all requirements for the reactor pressure boundary must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures, may be a viable option to meet the single-failure criterion. For vents larger than the

LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

(8) The probability of a vent path failing to close once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

(9) Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

(10) The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE 344-1975 as supplemented by Regulatory Guide 1.100, 1.92 and SEP 3.92, 3.43, and 3.10. Environmental qualifications are in accordance with the May 23, 1980 Commission Order and Memorandum (CLI-80-21).

(11) Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

(12) It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

B. BWR Design Considerations

(1) Since the BWR owners' group has suggested that the present BWR designs have an inherent capability to vent, a question relating to the capability of existing systems arises. The ability of these systems to vent the RCS of noncondensible gas generated during an accident must be demonstrated. Because of differences among the head vent systems for BWRs, each licensee or applicant should address the specific design features of this plant and compare them with the generic venting capability proposed by the BWR owners' group. In addition, the ability of these systems to meet the same requirements as the PWR vent system must be documented.

(2) In addition to RCS venting, each BWR licensee should address the ability to vent other systems, such as the isolation condenser which may be

required to maintain adequate core cooling. If the production of a large amount of noncondensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

(1) Each PWR licensee should provide the capability to vent the reactor vessel head. The reactor vessel head vent should be capable of venting noncondensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths).

(2) Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

(3) Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

Installation should take place by July 1, 1982. Until staff approval is obtained, installation may proceed; but operating procedures should not be implemented and valves should be placed in a condition so as to minimize the potential for inadvertent actuation (e.g., remove power).

Type of Review

A preimplementation review will be performed prior to authorizing use of the vent.

Documentation Required

By July 1, 1981, the licensee shall provide the following information on the reactor coolant vent system for staff review:

(1) The information requested in items 1 and 2 under "Position";

(2) A discussion of the design with respect to conformance to the design criteria discussed under "Clarification," including deviations, if any, with adequate justification for such deviations; and,

(3) Supporting information including logic diagrams, electrical schematics, piping and instrumentation diagrams, test procedures, and technical specifications.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

Letter from D. G. Eishenut, NRC, to all Operating Nuclear Power Plants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 13, 1979.

Letter from H. R. Denton, NRC, to all Operating Nuclear Power Plants, Subject: Discussions of Lessons Learned Short-Term Requirements, dated October 30, 1979.

U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660, Vols. 1 and 2, May 1980.

Commission Orders, May 23, 1980 (CLI-80-21).

221.0 CORE PERFORMANCE BRANCH, THERMAL-HYDRAULICS SECTION

221.21 Provide a description of the in-core thermocouple system. Include a description of the primary and backup means of monitoring in-core thermocouple temperature and readout/printout capability. State the time required to complete thermocouple mapping.

221.22 Provide complete "Information Required on the Subcooling Meter" defined in the October 30, 1979 letter from H. Denton (NRC) to All Operating Nuclear Power Plants.

221.23 Provide your schedule for the procurement, testing and installation of reactor vessel water level instrumentation at San Onofre 2 and 3.

231.0 CORE PERFORMANCE BRANCH, FUELS SECTION

231.33 Provide the data requested in the forms given in Appendix A so that we may complete our review of the generic methods and the plant-specific audit being conducted of the structural analysis of fuel element assemblies for combined seismic and LOCA loads.

231.34 The NRC staff has been generically evaluating three materials models that are used in ECCS evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) discussed our evaluation with vendors and other industry representatives (Reference 1), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Reference 2), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (References 3 and 4).

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we will require that the ECCS analyses in your FSAR be accompanied by supplemental calculations to be performed with the materials models of NUREG-0630. For these supplemental calculations only, we will accept other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above.

Please provide the supplemental calculations described above.

References

1. Memorandum from R. P. Denise, NRC, to R. J. Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979. Available in NRC PDR for inspection and copying for a fee.

2. D. A. Powers and R. O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980. Available from the NRC Division of Technical Information and Docket Control.

3. Letter from D. G. Eisenhut, NRC, to all Operating Light Water Reactors, dated November 9, 1979. Available in NRC PDR for inspection and copying for a fee.

4. Memorandum from H. R. Denton, NRC, to Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979. Available in NRC PDR for inspection and copying for a fee.

312.0 ACCIDENT ANALYSIS BRANCH

312.44 (15.7.3.4 9.4.3.1 15B.6.2) The post-accident air cleanup system for the fuel handling area is designed as a full flow recirculation system with redundant filter units. The system is not designed to produce and maintain a negative pressure in the building. Your model for the analysis of the radiological consequences from a fuel handling accident in the fuel building includes the following two assumptions: (1) The activity released from the fuel pool surface diffuses instantaneously to uniformly occupy the fuel building volume; and (2) There is no unfiltered leakage from the building to the environment.

a. With respect to the first assumption we note that the openings in the air intake and return ducts of the system are located approximately at the 110 feet elevation close to the roof of the building and approximately 50 feet above the surface of the spent fuel pool. The return duct openings are located as close as seven feet from the intake openings. The current design and operation of the system potentially can short-circuit the intended airflow and mixing of the atmosphere and therefore may not provide for an effective air cleanup, i.e., removal of radioiodine released from the pool surface during the accident.

We request that you provide an analysis of the air flow characteristics in the building that demonstrates the effectiveness of the system. Such analysis should take into consideration potential temperature gradients in the building that would inhibit natural convection flow. If your analysis shows that the existing system cannot assure the required mixing of the building atmosphere the relocation of the air intakes to within close proximity of the spent fuel pool would be an acceptable approach for providing an increased sweep action over the pool. Because such relocation would be limited by the required travel of the fuel handling bridge over the pool, you should consider a location of the intakes at the wall of the fuel building.

b. With respect to the second assumption, in your analysis the post-accident cleanup system is modeled as a once-through ventilation and filter system discharging directly to the environment as described in your response to our earlier question 312.38. While this model maximizes the offsite doses with respect to filtered leakage it does not consider the contribution from "actual exfiltration" which should be assumed to be unfiltered leakage. Such exfiltration could arise as a result of a pressure difference between the building internal pressure and the outside barometric pressure. Although the staff finds that the fuel handling building, in comparison with such buildings at other facilities, has been designed and constructed to greatly reduce such leakage we cannot conclude that it is a zero leakage building. We therefore request that you provide an analysis that defines the actual exfiltration rate under a slight overpressure (about 0.1 inches water gauge) in the fuel building. An acceptable approach would be the determination, by test, of the necessary air flow into the building that would produce and maintain the slight overpressure. Such test should be performed with the post-accident air cleanup system in full operation.

312.45 (15.6.3.3.5.1.2) Your analysis of the radiological consequences resulting from continuous post-LOCA leakage from ESF components located outside containment is based on the leakage sources listed in Table 15.6-19. These leakage sources include the valve stems and pump seals of the high and low pressure injection pumps and the containment spray pumps. It is our understanding that the valves listed in the table are located in various rooms within the ESF building. We request the following information:

a. Provide a listing and identify the location, by room, of the valves in each of the ESF systems.

b. Describe the potential leakage path(s) to the outside environment from each of the locations in (1) above.

c. Provide the bases for the leak rates from valve stems and seals as listed in Table 15.6-19 that were used in your analysis.

d. Propose technical specifications and surveillance requirements for the valves and seals listed in Table 15.6-9 above to assure that the leak rates listed will not be exceeded.

312.46 Our review of your response to Q312.42 concludes that you have not shown that the explosion risks associated with transportation of hazardous materials past the site are sufficiently low to be acceptable. Therefore, it is our position that you should consider some mitigative measures which would provide a demonstrable and significant reduction of the explosion risk. For example, we believe the following considerations should be evaluated for their effectiveness in risk reduction:

a. Moving the railroad switch, which is currently situated near SONGS Unit 2, outside the exclusion boundary and well to the south of it.

b. Continuous and visual monitoring of the I-5 highway and ATSF railway within the exclusion boundary. Timely detection of traffic accidents or other hazardous events, followed by an appropriate emergency response, should be considered. A contingency plan, and accident response capability (e.g., fire fighting personnel and equipment, traffic control under accident conditions) should be developed.

c. The ATSF railway should be monitored periodically and necessary corrective steps implemented whenever track conditions are found to be defective or degraded.

d. The effectiveness of a barrier between the ATSF railway and the plant should be considered with respect to heavier than air vapor diversion, overpressure intensity reduction, and minimizing the potential for derailed cars approaching the plant structures.

Alternatively, you may wish to consider other possible mitigative steps beside the above suggested items. Upon receipt of this type of information we will review it and evaluate its potential for risk reduction.

312.47 With respect to your analysis of toxic gas hazards from transportation accidents, we are unable to verify the motor carrier accident rate which is presented in Section 6.4 of the FSAR. The value of $2 \times 10^{-10}$ accidents per mile used in Section 6.4 is about four orders of magnitude less than the truck accident rate based on nationally averaged statistics used in FSAR Section 2.2 analyses. Thus, the estimated need for control room operator protection may have to extend beyond the selected gases (chlorine, butane, and anhydrous ammonia). Our position is that you should substantiate the truck accident rate used in the toxic gas analysis or revise it accordingly.

## 321.0 EFFLUENT TREATMENT SYSTEMS BRANCH*

321.12 (II.F.1) In your TMI submittal, you stated that the effluent monitors will provide continuous recording of the effluent in the control room. It is our position that this should be done for each separate release pathway. Explain how you can accomplish this with a single monitor which can switch back and forth between continuous vent and containment purge exhausts.

321.13 (II.F.1) How will you quantify the containment purge exhaust releases under widely varying containment purge exhaust rates ranging from a low 50 CFM to a high 40,000 CFM?

321.14 (II.F.1) Will the area radiation monitors that you propose to install to monitor steam dump/safety valve releases provide a dose rate range equivalent to Xe-133 equivalent concentration range of $10^{-1}$ to $10^{3}$ uCi/cc in the discharge? How will you correct the readings of these external monitors for low energy gammas? Describe the procedures and calculational methods you will employ to convert the dose rate to concentrations and release rates.

321.15 (II.F.1) Describe how you will initially calibrate the monitors and also at what frequency you will calibrate them periodically.

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*The numbers in parentheses beneath the question numbers refer to the applicable section of NUREG-0660.

321.16 (II.F.1) Indicate how you will correct instrument readings for background effects when applicable.

321.17 (II.F.1) How will you determine iodine and particulate effluent releases via containment purge and continuous vent exhausts if you provide only a single sampler for both the vents?

321.18 (II.F.1) Will the 4" lead shield that surrounds the filters be equivalent to the basis set forth for shielding envelope by NRC in its September 5 letter to all licensees and applicants on "Preliminary Clarification of TMI Action Plan Requirements"?

321.19 (II.F.1) Describe how you will initially calibrate the sample volumes and analysis and also at what frequency you will check them periodically.

321.20 (II.F.1) Demonstrate that the flow controlling devices have the capability to maintain iso-kinetic conditions with variations in stack or duct design flow velocity of $\pm$ 20 percent.

321.21 (III.D.1.1) We require that you leak test in the immediate future (a) containment spray and safety injection systems which you have recognized may contain highly radioactive fluids following a postulated accident (b) post-accident reactor coolant and containment air sample lines (containment air return sample line up to stop valve that will be added to the waste gas header), and (c) other applicable systems that are unique to San Onofre, Unit Nos. 2 and 3. You should provide a summary description, together with the initial leak test results at least 4 months prior to issuance of full power operating license.

321.22 (III.D.1.1) We require that you leak test the CVCS and waste gas systems, since they may get contaminated with highly radioactive fluids prior to their isolation and/or may be used during the accident.

321.23 (III.D.1.1) Provide the details of immediate leak reduction measures you plan to implement.

321.24 (III.D.1.1) The statement in your August 1980 TMI response that you are evaluating leakage of systems located outside the containment to determine whether a leak reduction program is necessary is unsatisfactory. Provide information on the continuing leak reduction program you are required to implement. This information should include (a) frequency of the integrated leak tests, (b) method and summary of procedures for testing each system or subsystem, (c) steps that you will take for minimizing occupational exposures, and (d) details on the preventive maintenance steps to reduce leakage to as-low-as practical levels.

321.25 (II.B.3) Provide assurance that reactor coolant and containment atmosphere sampling during post-accident situations will not require an isolated auxiliary system to be placed in operation in order to use the sampling system.

321.26 (II.B.3) Clarify what you mean by the statement that you have included provisions to measure total dissolved gas concentrations up to approximately 2,000 cc/KG.

321.27 II.B.3) Describe the sample room exhaust filters referred to in your August 1980 TMI submittal. Your description should include filter efficiencies for all forms of gaseous iodine and particulates.

321.28 (II.B.3) Provide assurance that backup sampling thorugh grab sampling will be provided for systems using in-line monitoring for samples. Give the frequency of such grab sampling.

331.0 RADIOLOGICAL ASSESSMENT BRANCH

331.20 Provide additional information concerning the two high range containment monitors required by our letter of November 9, 1979, implementing the Lessons Learned item 2.1.8.b of NUREG-0578. The supplemental information should include:

a. location of and type of readout (continuous and recording);
b. energy response (sensitive 60 kev);
c. calibration frequency and methods (refueling frequency);

The location of the monitors should be shown on plant layout drawings. The monitors should be located in a manner as to provide a reasonable assessment of radiation levels inside containment. Monitors should not be placed in areas which are protected by massive shielding.

331.21 Provide a summary of the shielding design review results required by our letter dated November 9, 1979, implementing the Lessons Learned item 2.1.6.b of NUREG-0578, and provide a description of the results of this review. Include in your description:

a. source terms used in the evaluation (NUREG-0578 specified that source terms in Regulatory Guide 1.3, 1.4 and 1.7 be used).
b. a listing of systems assumed in your analysis to contain high levels of radioactivity in a post-accident situation including, but not limited to, containment, SDCS, safety injection systems, CVCS, containment spray recirculation system, sample lines, and gaseous radwaste systems. If any of these systems or others that could contain high radioactivity were excluded, explain why such systems were excluded from review;

c. a listing of plant areas where access is considered necessary for vital system operation after an accident. The control room, Technical Support Center, sampling stations and sample analysis area must be included among those areas where access is considered vital after an accident. Your evaluation to determine the necessary vital areas should include but not be limited to, consideration of the control room, Technical Support Center, Operational Support Center (see letter dated April 25, 1980, D. G. Eisenhut to all power reactor licensees which allows substitution of an onsite TSC with an offsite TSC), sampling and sample analysis areas, manual SDCS alignment area, motor control centers, instrument panels, emergency power supplies, security center and radwaste control panels. If any of these areas were not considered areas where access was necessary after an accident, explain whey they are excluded;

d. designation of the codes used for analysis, such as ORIGEN, ISOSHIELD, QAD or others;

e. the projected doses to individuals for necessary occupancy times in vital areas;

f. a brief description of the proposed plant modifications resulting from the design review and confirmation that these modifications will be complete by January 1, 1981 or full power, whichever is later.

These modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area. For other modifications, to allow access to areas where access would be useful but not vital, you should specify the anticipated modification and the scheduled completion date for modification. Note that the control room modifications required by our letter of May 7, 1980, must be completed by January 1983.

331.22 In addition to your response to Lessons Learned item 2.1.8.c (Improved In-Plant Iodine Instrumentation) of NUREG-0578 contained in your December 1979 status report, describe the sample analysis equipment type and location, the sample flushing methods, such as using compressed clean air for purging entrapped noble gases, and the procedures and training in the use of the systems. There should be sufficient samplers to sample all vital areas. Sample results should be available within 10 minutes after the sample is taken. The sample analysis equipment must be located in a low background area after January 1, 1981.

331.23 Section 13.1.2.3 (Operating Shift Crew) of the FSAR, implies that a HP technician will not be onsite during all shifts. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," requires that a health physics technician be onsite at all times. Show how you plan to comply with this criteria and revise appropriate portions of the FSAR.

331.24 In section 13.1.3.1 you state that the Supervision of Plant Chemistry and Radiation Protection, Assistant Chemical/Radiation Protection Engineer, and Chemical Radiation Protection Foreman will all meet the minimum requirements of Regulatory Guide 1.8, "Personnel Selection and Training" which references ANSI 18.1. Provide updated resumes for the personnel who have been chosen to fill these positions with a breakdown of their qualifications corresponding to Regulatory Guide 1.8/ANSI 18.1 requirements (education, training, experience). The experience referenced for all above personnel must be in the individuals speciality, which in this case would be radiation protection.

331.25 According to the draft document "Criteria for Utility Management and Technical Competence", the Radiation Protection Section shall be separate from the Chemistry Section. In addition the Radiation Protection Manager shall 1) report directly to the Plant Manager or Unit 2&3 Superintendent, 2) report at the same level as the Supervisor of Plant Operations, and 3) be a member of PORC. It is our position that you make the above changes and revise your FSAR and proposed Technical Specifications Accordingly.

331.26 In section 13.1.2.3 you state that the Watch Engineer is responsible for implementing the radiation protection program in the absence of the Supervisor of Chemistry and Radiation Protection or his designated alternative. It is our position that this authority be delegated to a person more qualified in radiation protection in the absence of the supervisor of Chemistry and

Radiation Protection. Such persons would include the Assistant Chemical Radiation Protection Engineer, the Chemical Radiation Protection Foreman, or the HP technician on shift.

331.27 Based on information contained in the draft document "Criteria for Utility Management and Technical Competence", it is our position that your organization chain contain a qualified health physicist to provide backup in the event of the absence of the Supervisor of Chemistry and Radiation Protection. The December 1979 revision of ANSI 3.1 specifies that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, 2 years experience in radiation protection, 1 year of which should be nuclear power plant experience, 6 months of which should be on-site. It is our position that such experience be professional experience. Provide an outline of the qualifications of the individual who will act as the backup for the RPM in his absence.

432.0 EMERGENCY PREPAREDNESS LICENSING BRANCH*

432.18 (A.1) Illustrate the interrelationships among the licensee, and other involved organizations (State, local governments, DOE, laboratories etc.).

432.19 (B.5) It is not clear whether the staffing level on p. 5-1 is for one unit or for all three units. Please clarify this and justify any differences between your staffing level and the requirements in Table B-1 of NUREG-0654.

432.20 (B.6) Specify the interfaces among the onsite functional areas of emergency activity, headquarters support, local services support, and State and local government organizations. Illustrate these interfaces with a block diagram.

Identify the organization that corresponds to the EOF as described in NUREG-0654. Use the same block diagram to illustrate the relationship among the TSC, the EOF, and the corporate organization.

432.21 (B.9) Specify the contractors and private organizations (laboratories, vendors, etc.) who may be requested to provide technical assistance to the emergency organizations.

*The numbers in parentheses beneath the question numbers correspond to items in Section II of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants".

432.22 (C.1) More detail is needed on the agreement made with DOE, as outlined in NUREG-0654. Also, the agreement letter, presumably contained in Appendix A, is missing .

432.23 (C.3) Identify radiological laboratories and their capabilities and expected response times.

432.24 (D.1 & D.2) All example conditions found in NUREG-0610 should be included in each emergency class. The specific instruments, parameters or equipment status shall be shown for establishing each emergency class.

432.25 (E.3) Establish, in conjunction with State and local organizations, the contents of the initial messages to be sent from the plant. Provide the list of items to be imcorporate in these messages.

432.26 (E.4) Make provisions for followup messages from the facility to authorities. Such messages shall contain information listed as items a. through n. in NUREG-0654.

432.27 (E.5) Establish administrative and physical means to notify and instruct the public within the plume exposure EPZ, as describes in Appendix 3 of NUREG-0654.

432.28 (E.7, G.1, G.2) Describe your program to the public on matters such as emergency action levels, protective actions and evacuation.

432.29 (F.1) Describe provisions to alert or activate your emergency personnel (onsite and offsite).

432.30 (F.2) Provide commitment to periodically test the communications systems.

432.31 (G.3) Designate the principle points of contact and physical locations for use by news media during an emergency.

432.32 (G.4) Designate a spokesperson who will have access to all necessary information.

432.33 (H.2) Indicate when the design of the EOF will be completed. Provide its proposed location.

432.34 (H.4) Provide information on the staffing level of the EOF.

432.35 (H.5 & H.6) Indicate which natural phenomena monitors listed in table 7-3 are to be phased offsite.

432.36 (H.7) Identify offsite meteorological capacity in the vicinity of your plant.

432.37 (H.8) Provide meteorological instrumentation and procedures which satisfy the criteria in Appendix 2, and provisions to obtain representative real-time meteorological information from other sources. (The NRC staff will establish a schedule for the implementation of the requirements, set forth in this Appendix).

432.38 (H.12) Establish a central point for the receipt and analysis of all field monitoring data.

432.39 (I.1) Identify plant system and effluent parameter values characteristic of a spectrum of off-normal conditions and all the example initiating conditions in NUREG-0610.

432.40 (I.2) Provide a map showing the location of onsite and offsite radiation monitors.

432.41 (I.3) Provide methods and techniques to determine the source term of release of radioactive material within plant systems, and the magnitude of the release of radioactive materials based on plant system parameters and effluent monitors.

432.42 (I.4) Establish the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions.

432.43 (I.5) (See H.8.) Also, there shall be provisions for access to meteorological information by emergency response centers.

432.44 (I.6) Establish the methodology for determining the release rate/projected doses if the instrumentation for such assessment are offscale or inoperable.

432.45 (I.7) Provide the sensitivities of your air samples.

432.46 (I.8) Describe the capability and resources for field monitoring within the plume exposure EPZ.

432.47 (I.9) Provide methods, equipment and expertise to make rapid assessments of the actual or potential magnitude and locations of any radiological hazards through liquid or gaseons release pathways. (This shall include activation, notification means, field team composition, transportation, communication, monitoring equipment and estimated deployment times).

432.48 (I.10) Establish means for relating the various measured parameters to dose rates for key isotopes and gross radioactivity measurements.

432.49 (U.4) Provide decontamination capability for evacuated personnel.

432.50 J.6) Provide, for individuals onsite, respiratory protection, protective clothing and radioprotective drugs.

432.51 (J.7 & J.10) Expand Section 6.4.2 of your plan to describe the mechanism for recommending protective actions to State and local officials. Such recommendation should be based on emergency action levels and Table S-1 of EPA-510/1-75-001, and taking into account factors such as evacuation time and local protection.

432.52 (J.10) Provide maps to show pre-selected radiological sampling and monitoring points, and population distribution. These shall be in a format described in Table J-1.

432.53 (K.2) Provide an onsite radiation protection program to be implemented during emergencies. It shall identify individuals, by position or title, who can authorize emergency workers to receive doses in excess of 10 CFR 20 limits.

432.54 (K.5) Specify the criteria for determining the need for personnel decontamination.

432.55 (K.6) Provide your criteria for permitting return of areas and items to normal use after their contamination (Expand Section 9.1).

432.56 (L.4) The recovery plan shall include a method for periodically estimating total population exposure.

432.57 (N.1 thru N.5) The plan should include a discussion of exercises and drills to be held periodically. The information you supplied in Section 8.1 and Table 8-1 is brief. The description of these drills and exercises should follow the general outline and reach the level of detail suggested by NUREG-0654.

432.58 (P.8) Provide an index which covers any State and local plans, and a cross reference between your plan and each criteria in NUREG-0654.

432.59 (P.9) Commit to conducting independent audits of your emergency plan at least once every two years. The audit shall include the plan, its implementing procedures, training, readiness testing and equipment. Management controls shall be implemented for evaluation and correction of audit findings.

630.0 LICENSEE QUALIFICATIONS BRANCH

630.1 NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" was published for interim use and comment in September 1980. Included in Item II.A.2.c of NUREG-0731 are guidelines for qualifications of the Shift Technical Advisor. Please amend your response to NRC Action Plan Item I.A.1.1 to include a discussion of how you intend to meet these guideline qualifications.

630.2 Your response to Action Plan Item I.A.1.3 referred to our July 31, 1980 letter in addressing overtime work. NUREG-0731 has modified certain of the overtime restrictions. Please amend your response accordingly.

630.3 Your response to Action Plan Item I.A.2.3 should include commitment to meet the NUREG-0660 instructor qualification requirements related to successful completion of senior operator examination and to requalification.

APPENDIX A

# DATA REQUEST FORM



## DRAWINGS

| | |
|---|---|
| FUEL ASSEMBLY | INCLUDE TOLERANCES ON ANY GAPS BETWEEN COMPONENTS AND INTERFACE DETAILS |
| CORE PLATE | |
| CORE BARREL OR SHROUD | |

## FORCING FUNCTIONS

| TYPE | DESCRIPTION | DESIRED FORM |
|---|---|---|
| LATERAL CORE PLATE MOTIONS | ACCELERATION TIME HISTORIES | PLOTS |
| AXIAL LOCA FORCES | PRESSURE TIME HISTORIES | PLOTS |

# DATA REQUEST FORM



## TEST DATA

| TYPE | AXIAL | LATERAL | DESIRED FORM |
|---|---|---|---|
| FORCE-DEFLECTION (STATIC) | | | PLOTS |
| (1st 7-10 Lateral) MODE SHAPES | | | (Amplitude at Grids) PLOTS |
| (1st 7-10 Lateral) FREQUENCIES | | | TABLE |
| DROP TEST FORCE-TIME (Plus Drop Height) | | | PLOTS |
| INTERNAL ROD TO GRID IMPACT STIFFNESS | | | TABLE |
| EXTERNAL GRID IMPACT STIFFNESS | | | TABLE |
| BEAM-COLUMN RESULTS | | | DISCUSSION TABLE OR PLOTS |
| GRID-ROD FRICTION | | | TABLE |
| GRID CRUSH STRENGTH (IMPACT) | | | TABLE |

# DATA REQUEST FORM



MATERIAL DESCRIPTION

| COMPONENT | WEIGHT | CENTER OF GRAVITY | TYPE MATERIAL | STIFFNESS AND/OR PRELOAD |
|---|---|---|---|---|
| FUEL ROD | | | | |
| SPACER GRID | | | | |
| TOP END BOX | | | | |
| BOTTOM END BOX | | | | |
| FUEL | | | | |
| HOLD-DOWN SPRING | | | | |
| FUEL ASSEMBLY | | | | |