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

MAY 16 1985

Docket Nos.: 50-498
and 50-499

Mr. J. H. Goldberg
Group Vice President - Nuclear
Houston Lighting and Power Company
Post Office Box 1700
Houston, Texas 77001

Dear Mr. Goldberg:

Subject: South Texas Project, Units 1 and 2 - Request for Additional Information

The NRC staff has determined that additional information is required for the review of the South Texas Project operating license application. Enclosed are the following Request for Additional Information (RAIs):

1. Mechanical Engineering Branch: The questions are designated 210.18 to 210.65. A draft version of most of these questions was transmitted to you earlier to facilitate preparation toward a meeting, preferably during the week of June 24, 1985.

2. Containment System Branch: The questions are designated 480.1 to 480.28.

3. Auxiliary Systems Branch: The questions are designated 410.01 to 410.19.

4. Reactor Systems Branch: The enclosed are additional questions which follow the series of questions 440.14 to 440.72 transmitted to you previously.

5. Environmental and Hydrologic Engineering Branch: There are two subsets of questions under this category related to the environmental review. The first subset is related to hydrologic engineering and designated 240.01 to 240.14. The second is related to water quality and terrestrial ecology impacts and designated 290.5 to 290.15 and 291.10 to 291.25.

The staff is available to discuss the above RAI's as may be required to provide any necessary clarification. Please inform us as to your schedule for responding to the RAI's. Please contact the Project Manager if you have any questions.

Sincerely,

George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
As stated

cc: See next page

The staff is available to discuss the above RAI's as may be required to provide any necessary clarification. Please inform us as to your schedule for responding to the RAI's. Please contact the Project Manager if you have any questions.

Sincerely,

/S/

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
As stated

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SOUTH TEXAS 1 AND 2 SER QUESTIONS

SECTION 3.6.2

210.18 Paragraph 3.6.2.1.1.4 of the FSAR stated that "In the absence of an ASME Code Class 1 stress analysis, breaks are postulated at all fittings, valves or welded attachments." Identify all the Class 1 piping systems for which an ASME Code Class 1 stress analysis is not performed.

210.19 Provide assurance that the guidance stated in BTP MEB 3-1, Section B.1.C (1)(d)(iii) concerning changes of new highest stress locations as a result of piping reanalysis will be used in STP high energy line break location postulation.

210.20 In order to assure that the pipe break criteria have been properly implemented, the Standard Review Plan requires the review of sketches showing the postulated rupture locations and of summaries of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. The required sketches and tables for some high energy piping systems have not been provided at this time in the FSAR. Provide a schedule for submission these data.

210.21 FSAR Section 3.6.2.1.3.2.1(a) and (b) described the criteria used for determining types of breaks for high energy piping other than RCL piping. These criteria do not comply with the criteria specified in BTP MEB 3-1, Section B.3.a(1) and B.3.b.(1). Revise your FSAR to conform to the SRP criteria.

210.22 BTP MEB 3-1, Section B.3.b.(3) specifies that for longitudinal breaks, axial splits should be oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions causes out-of-plane bending of the piping configuration. Provide assurance that this guidance has been used.

210.23 Discuss how jet impingement effects on target piping systems and components were evaluated, specifically, the criteria used in determining the acceptability of the target piping systems and components.

210.24 The staff finds that there is insufficient information describing the jet expansion model used for evaluation of jet impingement effects of steam, saturated water or steam-water mixtures. Provide additional information to assure that the criteria described in SRP 3.6.2 Section III.3 have been met for analysis of jet impingement forces.

210.25 SRP 3.6.2 states that rise times for jet thrust not exceeding one millisecond should be used unless justified. Provide assurance that this justification will be included in the FSAR.

210.26 Provide assurance that the criteria described in FSAR Section 3.6.2.1.1.7 relative to a structure separating a high energy line from an essential component has been used for both inside and outside containment.

210.27 Provide a listing of those postulated pipe breaks where limited displacements have been used to reduce break areas.

210.28 Is there any unrestrained whipping pipe inside containment? If so, discuss how pipe whip and jet impingement effects were determined for those postulated breaks in high energy piping that are not restrained (unrestrained whipping pipe). Provide the acceptance criteria for the impacted safety-related structures, systems, and components.

210.29 Provide the loads, load combinations, and stress limits that were used in the design of pipe rupture restraints. Include a discussion of the design methods applicable to the auxiliary steel used to support the pipe rupture restraint. Provide assurance that the pipe rupture restraint and supporting structure cannot fail during a seismic event.

210.30 Provide the design criteria used for pipe rupture restraints that also support piping.

210.31 SRP 3.6.2 Section III.2.a. states that the rated energy dissipating capacity shall be taken a not greater than the area under the essentially flat portion of the load deflection curve for crushable materials. Provide assurance that this guidance has been used.

210.32 No discussion could be found in the FSAR regarding design stress limits for Class 1 piping in the break exclusion zone. If there are any Class 1 lines in the break exclusion zone, provide the required design limits.

210.33 The criteria in the FSAR for designating the break exclusion zones on piping in the containment penetration areas require further justification. Identify all the branch lines which are considered as part of the break exclusion zone. Provide drawings and/or other information quantifying the lengths of pipe for all systems including branch lines defined by criteria of Section 3.6.2.1.1.5 of the FSAR.

210.34 Provide assurance that 100% volumetric inservice examination of all pipe welds in the break exclusion zone will be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

210.35 Discuss how high energy leakage cracks were considered.

210.35 BTP MEB 3-1, Section B.2.C specifies criteria for postulating through-wall leakage cracks for moderate energy fluid systems in areas other than containment penetrations including ASME Code Class 1, 2, 3 and non-safety class piping both inside and outside containment. FSAR Section 3.6.2.1.2 states cracks in moderate energy ASME Code Class 1 piping are not postulated. Provide justification for not postulating cracks in moderate energy Class 1 piping.

210.37 Provide a schedule for submission of response to NRC Question 110.6.

SECTION 3.9.1

210.38 Justify not considering the following primary system transients for normal conditions listed in FSAR, Section 3.9.1.1.6.
1) Reactor coolant pumps startup and shutdown
2) Reduced temperature return to power

210.39 FSAR Section 3.9.1.3 states that an experimental stress analysis method has been used for essential cooling water underground aluminum bronze piping. Provide the acceptance criteria used for the test program described in the FSAR.

210.40 Identify components for which inelastic analysis has been used. If any, provide details of methods used.

SECTION 3.9.2 and 3.7.3

210.41 SRP 3.9.2 requires a list of systems for which visual inspections and measurements (as needed) will be performed during the pre-operational piping testing program. Provide a list of systems to be included in the pre-operational testing program.

210.42 Provide the acceptance criteria that will be used to determine if the vibration levels observed or measured during the pre-operational testing are acceptable. Specifically address how the vibration amplitudes will be related to a stress level and what stress levels will be used for both steady-state and transient vibration.

210.43 It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or startup testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

210.44 Discuss how floor response spectra curves are broadened for NSSS and BOP scope.

210.45 Discuss how closely spaced modes are combined for BOP scope.

210.46 On page 3.7-21 of the FSAR it is stated that in certain cases, such as with auxiliary piping connected to the reactor coolant loop, multiple spectra have been used to reduce the excessive conservatism in applying enveloped spectra over the entire length of piping. Discuss how multiple spectra are used.

210.47 SRP Section 3.9.2.III.2.a.(2)(c) states that to obtain an equivalent static load of equipment or component which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response. FSAR Section 3.7.3B.1.7 does not comply with this guidance. Provide justification for not using a factor of 1.5.

210.48 Provide additional information to justify the use of a multiplication factor of 1.0 in the equivalent static load method for design of cable tray hangers and heating, ventilating and air conditioning (HVAC) duct supports.

210.49 SRP 3.9.2.II.2.h. specifies criteria for using constant vertical static factors. The use of constant vertical static factors is acceptable only if it can be justified that the structure is rigid in the vertical direction. Provide assurance that this guidance has been used.

210.50 FSAR Section 3.7.3A.2 states that a minimum value of five cycles per seismic event (one SSE and five OBEs) is selected for BOP seismic Category I systems and components. With respect to the NSSS scope, FSAR Section 3.7.3B.2 states that a time history study has been conducted to arrive at a realistic number of maximum stress cycles per OBE occurrence. As a result of this study, 10 maximum stress cycles for flexible equipment and five maximum stress cycles for rigid equipment for each OBE occurrence are used for fatigue

evaluation of Westinghouse systems and components. However, FSAR Table 3.9-8, Summary of Reactor Coolant System Design Transients, lists 400 cycles for OBE and 1 occurrence for SSE. Provide justification for the differences of earthquake cycles listed in the above referenced FSAR sections and table. Include in your discussion the nonconformance to SRP Section 3.9.1.II.2.b criteria of a minimum of 10 maximum stress cycles per seismic event (one SSE and five OBE).

210.51 Provide the basis used for the design of piping anchors which separate seismically designed piping and non-seismic Category I piping. Include in your discussion, the loads and load combinations used and how the local pipe wall stresses are considered.

210.52 FSAR Table 1.3-1, Comparison with Similar Facility Design, states that the new design of the reactor vessel head closure system and lower internals are different from the Comanche Peak plant. Provide additional information which describes the differences in lower internals design between STP and Comanche Peak. Specifically, describe any changes in the reactor internals design which may have resulted from utilization of the Rapid Refueling concept at STP. If such changes exist, discuss the effects of these changes on the response of the reactor internals to flow-induced excitation and provide the basis for meeting the guidelines of Regulatory Guide 1.20 and maintaining Indian Point, Unit 2 as the prototype plant for STP.

SECTION 3.9.3

210.53 FSAR Section 5.3.1.7 describes the Roto-Lok reactor vessel head closure system which is used for the STP Units 1 and 2 reactor vessel head. It also states that a prototype Roto-Lok closure system has been tested to verify this closure design. Results of these tests are presented in the WCAP-8447, December, 1974. However, Section 7 of WCAP-8447 states that, "Also, it should again be noted that the program described in this report was for development hardware and testing only. The final design and analysis for a particular vessel is performed by the vessel supplier when the Roto-Lok is actually applied to production vessels." The staff's review of the WCAP-8447 as provided in the letter from J. F. Stoltz to C. Eicheldinger dated September 2, 1977 determined that WCAP-8447 provides an acceptable basis for the preliminary design of the Roto-Lok closure system. Furthermore, in that evaluation, the staff required that for the first reactor vessel to use this closure system (South Texas Plant) the results of final design and analysis of the closure system be provided in the FSAR. The applicant is requested to provide this information. Include in your discussion how the assumptions presented in the WCAP-8447 are applicable to the STP Units 1 and 2 plant specific reactor vessels.

210.54 The staff finds that there is insufficient information describing the design of safety-related HVAC ductwork and supports. Provide the design basis used for qualifying the HVAC ductwork and support structural integrity.

210.55 Provide the basis for assuring that ASME Code Class 1, 2 and 3 piping systems are capable of performing their safety function under all plant conditions. Describe the methodology used to assure the functional capability of essential piping systems when service limits C or D are specified.

210.56 The staff review of FSAR Section 3.9.3.3 finds that the design and installation details for mounting of pressure-relief devices requires further clarification. Provide the following information for our review:

(1) Clarify whether it is the intention of Section 3.9.3.3.2 to address BOP supplied components.
(2) Clarify whether all the NSSS scope safety and relief valves transients are evaluated using detailed dynamic analysis techniques. Provide assurance that the most severe potential sequence of discharges, i.e., the maximum values of forces and moments are considered for multiple-valve discharges.
(3) Provide a discussion of the basis for assuring that the valve end loads are acceptable. Specifically, address how the applicable design loads will be correctly reflected in the valve design specification.

210.57 The staff review of FSAR Section 3.9.3.4 finds that there is insufficient information regarding the design of ASME Class 1.2 and 3 equipment and component supports. Per SRP Section 3.9.3, our review includes an assessment of design and structural integrity of the supports. The review addresses three types of supports: (1) plate and shell, (2) linear, and (3) component standard types. For each of the above three types of supports, excluding pipe supports, provide the following information (as applicable) for our review:

(a) Describe (for typical support details) which part of the support is designed and constructed as component supports and which part is designed and constructed as building steel (NF vs. AISC jurisdictional boundaries).
(b) Provide the complete basis used for the design and construction of both the component support and the building steel up to the building structure. Include the applicable [illegible]es and standards used in the design, procurement, installati[illegible]xamination, and inspection.
(c) Provide the loads, load combinations and stress limits used for the component support up to the building structure.
(d) Provide the deformation limits used for the component support.
(e) Describe the buckling criteria used for the design of component supports.

Specifically, describe how the "A" term used in the response to NRC Question 110.19 was defined.

210.58 Valve discs are considered part of the pressure boundary and as such should have allowable stress limits. Provide these limits for our review.

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

Pre-Service Examination

A pre-service examination should be made on all safety-related snubbers. 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 minimum 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 specifications.
(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, and 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.

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 discrepancies 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.

210.60 Does the design criteria for component supports in systems categorize the stresses produced by seismic anchor point motion of piping and the thermal expansion of piping as primary or secondary? It is the staff's position that for the design of component supports, and stresses produced by seismic anchor point motion of piping and the thermal expansion of piping should be categorized as primary stresses. The application of this position is most critical for those supports which would be subjected to large deformations.

210.61 Describe what actions have been taken to address the staff concerns regarding stiff pipe clamps as described in IE Information Notice 83-80.

210.62 The staff's review of your component support design finds that additional information is required regarding the design basis used for bolts.

(a) Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.
(b) Provide a discussion of the design methods used for expansion anchor bolts used in component supports.

SECTION 3.9.5

210.63 Clarify the statement on page 3.9-80 of the FSAR that, "The allowable stress limits during the Design Basis Accident used for the core support structures are based on the 1974 edition of the ASME Code for Core Support Structures, Subsection NC, and the criteria for faulted condition."

SECTION 3.9.6

210.64 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.

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, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute (GPM) for each valve 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 value would be indication of valve degradation from one test to another.

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.

210.65 Provide a schedule for completion of your program for inservice testing of pumps and valves including any request relief from ASME Section XI requirements.

REQUEST FOR ADDITIONAL INFORMATION
FOR CONTAINMENT SYSTEMS REVIEW OF
SOUTH TEXAS PROJECT FSAR

480.1 (6.2.1) Provide the following information concerning the passive heat sink models used in the containment functional analyses and the minimum containment pressure analysis for performance capability studies of ECCS:

a. Explain the differences between the passive heat sink model used for the containment functional analyses of the LOCA and the MSLB as listed in Tables 6.2.1.1-7 and 6.2.1.1-13, respectively. The structural heat sink model must be shown to conservatively maximize the containment temperature and pressure. Also, justify the exclusion of the air gap between the containment liner and the concrete in the passive heat sink model for the MSLB.

b. Explain the reason for the differences between the assumed thermophysical properties in the LOCA and MSLB passive heat sink models as shown in Table 6.2.1.1-8 and 6.2.1.1-14, respectively.

c. Describe the bases of the material thicknesses and surface areas listed in Table 6.2.1.5-4, "Passive Heat Sink Data for Minimum Post-LOCA Containment Pressure," and explain how they conservatively minimize the containment pressure.

480.2 (6.2.1) Provide an "accident chronology" for the design basis MSLB (1.4 ft^2 DER MSLB 102% Power Min CHRS) similar to the accident chronologies provided for LOCAs in Table 6.2.1.1-10. Also provide information to demonstrate that the assumed times for full operation of the containment spray system and reactor containment fan coolers (RCFCs) in the MSLB analyses are conservative.

480.3 (6.2.1) Discuss and justify the initial conditions used in the maximum external containment pressure analysis (Section 6.2.1.1.3.6) and minimum containment pressure analysis (Section 6.2.1.5). For example, assuming a spray water temperature of 65°F is not conservative since a minimum temperature of 37°F is specified in Technical Specification 3.3.5C for the refueling water storage tank. Also, assuming an initial containment pressure of 0 psig is not conservative since the minimum pressure specified in Technical Specification 3.6.1.4 is -0.2 psig.

480.4 (6.2.1) In Section 6.2.1.2.2.1 of the FSAR, it is stated that for the reactor cavity subcompartment analysis, the postulated pipe rupture occurs in the inspection toroid and only a small fraction of the blowdown enters the reactor cavity. Justify that only a small fraction of the break flow enters the cavity, i.e., discuss how the break flow is prevented from entering the cavity, and why it is not appropriate to postulate a break inside the cavity. Provide

appropriate plan and elevation drawings of the reactor cavity showing the inspection toroid, piping, pipe restraints, postulated break location and vent paths (flow area) to the reactor cavity and steam generator compartment, including the blowout panel for venting to the lower reactor cavity.

480.5 (6.2.1) The reactor cavity model described in Section 6.2.1.2.3.2 indicates that "one hundred and eighty-degree symmetry was assumed...". Explain and justify the use of the one hundred and eighty-degree model versus modeling the entire reactor cavity and inspection toroid.

480.6 (6.2.1) Concerning the blowout panel in the heating, ventilating, and air conditioning ducting leading from the loop compartment subpedestal space to the lower reactor cavity (i.e., junction 110 in Table 6.2.1.1-4):

a. Justify the constant vent area of 4.05 square feet given for this vent path in Table 6.2.1.2-4.

b. Provide the dynamic analysis of the blowout panel that gives the vent area as a function of time after the break.

c. Provide drawings showing details of the blowout panel and surrounding areas.

d. With regard to possible generation of missiles, describe the potential for damage to safety-related systems by the blowout panel during a loss-of-coolant accident within the reactor cavity/inspection toroid.

480.7 (6.2.1) For the reactor cavity analysis, provide justification that vent areas will not be partially or completely plugged by displaced objects (e.g., insulation). Of particular concern is the rationale for not considering the blockage of the vent paths through the restricted clearance spaces around the primary piping nozzles in the reactor cavity/inspection toroid analysis.

480.8 (6.2.1) In Table 6.2.1.2-1, it is stated that the short term mass and energy release rates "include a 10% margin not used" in the subcompartment analysis. Explain the origin of this 10% margin and justify why it is not to be used in the subcompartment analysis. Present the mass and energy release rates actually used in the subcompartment analyses.

480.9 (6.2.1) Provide the results of the nodal sensitivity studies performed for the steam generator subcompartment analysis referenced in Section 6.2.1.2.3.3. The concern results because of the gross nodal modeling, particularly in nodes 1 through 8, which does not account for flow restrictions and variations around piping and other obstructions.

480.10 (6.2.1) Provide the rationale for connecting node 3 in the surge line model (shown in Figures 6.2.1.2-4 and 6.2.1.2-12 of the FSAR) to node 15 (containment) directly and thereby ignoring the intervening pressurizer compartment.

480.11 (6.2.1) Reference 6.2.1.5-1 is referred to in Section 6.2.1.5 of the FSAR, but has been left out of the reference list of Section 6.2. Provide the missing reference in the list.

480.12 (6.2.1) The response to Q 022.2, regarding the subcompartment analysis, is incomplete since it refers to a future amendment. Provide additional information to complete the response.

480.13 (6.2.2) In Figure 6.2.1.5.2, the Containment Fan Cooling Heat Removal Performance Curve is only shown for containment atmosphere temperatures up to 275°F. Expand this curve to cover the maximum expected range of containment atmosphere temperature in postulated accidents (for example, the peak containment atmosphere temperature from a MSLB is 323°F).

480.14 (6.2.2) The FSAR states in Section 6.2.2.2.1 that the RCFC performance is not affected by flooding following a LOCA, as the discharge points of the supply duct are located above the flood level. However, as shown in FSAR Figure 6.2.2-5, portions of the supply duct from the RCFCs lie below the containment flood level. Justify that water would not accumulate in the supply duct as a result of a LOCA or MSLB and thereby cause an unacceptable increase in the discharge head of the RCFCs and consequent decrease in RCFC flow. Also verify that the supply duct is leak-tight and able to withstand the maximum postulated differential pressure resulting from submergence following the design basis LOCA or main steam line break.

480.15 (6.2.2) Table 6.2.2-4 of the FSAR indicates an available pump NPSH of 17 feet. Submit a NPSH analysis in sufficent detail to permit the staff to assess the adequacy of the analysis. Also, provide the numerical values of each term in the NPSH equation for the response to Q 22.14.

480.16 (6.2.4) The containment isolation provisions for each fluid line penetrating containment must conform to the requirements of General Design Criteria 54, 55, 56 or 57, as appropriate. Those containment penetrations whose isolation provisions do not satisfy the explicit requirements of the general design criteria, but which are

acceptable on some other defined basis, should be discussed line by line, with the deviation identified and the specific "other defined basis" justified. For example, penetrations associated with the main steam lines (M1, M2, M3 and M4) are not identified as satisfying any GDC requirements.

480.17 (6.2.4) a. FSAR Figure 9.4.5-3 indicates that the supplementary containment purge exhuast piping upstream of the inside containment isolation valve, which takes suction at the top of the containment dome, is non-nuclear safety and non-seismic. Therefore, this piping cannot be relied upon to remain intact following a LOCA. Thus, provisions (i.e., debris screens) are required to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam (reference BTP CSB 6-4 B.1.g.). The same concerns apply to the supplementary containment purge supply piping downstream of the inside containment isolation valve, which directs outside air to the RCFC ring duct.

It is the NRC position that debris screens are required for the supplementary containment purge subsystem. Provide engineering drawings showing the materials and dimensions of the supplementary containment purge subsystem debris screens, and demonstrate compliance with the following criteria:

(1) The debris screen should be Seismic Category 1 design and installed about one pipe diameter away from the inner side of the inboard isolation valve.

(2) The piping between the debris screen and the isolation valve should also be Seismic Category 1 design.

(3) The debris screen should be designed to withstand the LOCA defferential pressure.

b. Specify the maximum allowable leak rate of the purge isolation valves giving appropriate consideration to valve size and maximum allowable leakage rate for the containment (as defined in Appendix J to 10 CFR Part 50) (reference BTP CSB 6-4 B.5.d).

480.18 (6.2.4) NUREG-0737 Item II.E.4.2 pertains to containment isolation dependability. Describe specifically how each paragraph of this NUREG-0737 item is satisfied. Concerning Position (6) and Clarification (7), verify that the normal containment purge isolation valve (HA007, HA008, HA009, and HA010) will be sealed closed as defined in SRP Section 6.2.4 Item 11.f, during the operational modes of power operation, startup, hot standby, and hot shutdown, and that these valves will be verified to be closed at least every 31 days.

480.19 (6.2.4) Valves FV9647, FV9698, FV9696, are listed as containment isolation valves for the normal containment purge subsystem in Table 3.6-1 of FSAR Chapter 16 but are not listed in Figure 6.2.4-1 and are not shown on Figure 9.4.5-2. Provide a piping and instrumentation diagram showing the location of these valves relative to the normal containment purge subsystem penetrations (M-41 and M-42) and the previously identified containment isolation barriers associated with these penetrations. Also, provide the design information prescribed by Section 6.2.4.2 of Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," for these valves.

480.20 (6.2.4) Verify that the containment isolation valves in Table 3.6-1 of FSAR Chapter 16 are consistent with those in Figure 6.2.4-1 of the FSAR.

480.21 (6.2.5) Demonstrate compliance with the acceptable assumptions for evaluating the production of combustible gases following a loss-of-coolant accident listed in Table 1 of Regulatory Guide 1.7, i.e., state your assumptions for fraction of fission product radiation energy absorbed by the coolant, hydrogen yield rate, oxygen yield rate, and fission product distribution model. Also verify that the assumption for the extent of metal-water reaction (1.5%) is the greater of 5 times the extent of the maximum calculated reaction under 10 CFR 50.46 or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.00023 inches.

480.22 Verify that the fission product decay energy used in the calculation
(6.2.5) of hydrogen from radiolysis of the emergency core cooling water and sump water is equal to or more conservative than the decay model given in Branch Technical Position ASB 9-2 in SRP Section 9.2.5 (Reference SRP Section 6.2.5 Item 11.2).

480.23 Table 3.2.A-1, Sheet 11 of 23 indicates that the containment hydrogen
(6.2.5) monitoring system hydrogen analyzer packages and piping and valves outside containment beyond the outer containment isolation valves are Safety Class 3. It is the NRC position that the combustible gas control system, including the containment hydrogen monitoring system, shall be designed, fabricated, erected, and tested to Group B quality standards (i.e., Safety Class 2), as recommended in Regulatory Guide 1.26 (Reference SRP Section 6.2.5 Item 11.7). Provide information on how you will comply with this position.

480.24 Identify containment penetrations that are proposed to be excluded
(6.2.6) from the local leak rate testing program. For each of these penetrations, justify that it does not constitute a potential containment atmosphere leak path under postulated accident conditions.

480.25 (6.2.2) The response to NRC Question 022.13, and Sections 6.2.2.1.2 and 6.2.2.2.3, Table 3.12-1, and Figure 6.2.4-2 of the FSAR, address compliance with Regulatory Guide 1.82 but do not demonstrate that every paragraph of the position section has been satisfied. Using engineering drawings and design descriptions as appropriate, describe specifically how paragraphs 4, 5, 9, 10, 11 and 13 of the Regulatory Guide 1.82, Rev. 0, position section have been satisfied.

480.26 (6.2.2) Provide plan and elevation drawings of the recirculation intake structures; show the level of water in the containment following a loss-of-coolant accident (LOCA) in relation to the structures. Describe how the screening is attached to the intake structures to preclude the possibility of debris bypassing the screening.

Discuss the design capability to withstand the differential pressure that may be imposed by the accumulation of debris.

480.27 (6.2.2) In Amendment 36 of the FSAR, it is stated that the emergency sump design complies with Appendix A to Regulatory Guide 1.82, proposed Rev. 1. However, no analysis has been presented to support this statement. Therefore, provide an analysis of debris generation (describe the types of debris that might be generated) and debris transport to the sump intake structure, in the event of a LOCA. Provide a NPSH analysis to show that adequate NPSH (i.e., with sufficient margin) is available to the recirculation pumps.

480.28 To provide additional assurance that long term cooling of the reactor
(6.2.2) 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. Containment cleanliness should be periodically assured; at a minimum this inspection should be performed at the end of each refueling outage.

B. Describe any changes deemed necessary to reduce vertical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions. 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.

C. Compare the size of opening in the fine screens with the minimum dimensions in the pumps which take suction from the sump, the minimum dimension in any spray nozzle 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. Estimate what effect debris particles, capable of passing through the fine screen, would have on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

REQUEST FOR ADDITIONAL INFORMATION
SOUTH TEXAS PROJECT, UNITS 1 AND 2
AUXILIARY SYSTEMS BRANCH

410.01 (3.5.1.1) The FSAR states that the auxiliary and main feedwater pump turbines are protected from overspeed by redundant overspeed trips and that neither turbine is considered to be a source of missiles. Regardless, provide the results of an analysis which shows safe shutdown will not be affected by such missiles.

410.02 (3.5.1.1 and 3.5.1.2) Provide the results of an analysis, as per commitment in the FSAR, which provides information relating to missiles generated by postulated failures of pumps and fans. Note that your response should include single failure criterion.

410.03 (3.5.1.1 3.5.1.2) Verify that missiles produced from pressurized tanks and compressed air/gas cylinders have been evaluated and that they will not affect safe shutdown equipment.

410.04 (3.5.2) Provide the results of an analysis which shows that the diesel generator exhausts on the side of the building at elevation 65 ft. 8 in. above plant grade are not subject to the larger missiles of SRP 3.5.1.4 (utility pole); i.e., there is no elevation 35 feet or higher within 1/2 mile of the plant. Furthermore if blockage of exhaust opening by smaller size missiles such as 4 x 12 plank would prevent diesels from starting provide protection for the exhaust or perform a PRA to demonstrate that the probability of significant damage to the diesel generator exhaust piping due to tornado missiles causing a release of radioactivity in excess of 10 CFR Part 100 limits shall be less than or equal to a median valve (realistic) 10^{-7} per year or a mean valve (conservative) of 10^{-6} per year. The loss of offsite power should be assumed in the PRA.

410.05 (3.6.1) Provide the results of a high energy pipe break analysis that addresses the consequences of pipe whip, jet impingement, flooding and environmental effects on safety-related systems and components as indicated on the FSAR Table 3.6.2.2.

410.06 ((3.6.1) Provide the results of an analysis which considers the affects of a main steam line and feedwater line breaks in the valve room taking into account the possibility of superheated steam conditions occurring as a result of the break uncovering the steam generator tubes. The single active failure should be considered.

410.07 (5.2.5) Provide the leak detection sensitivity in GPM of the containment air humidity instrument and other pressure and temperature instruments in the containment.

410.08 (5.4.11) In considering the event that the rupture disks of the pressurizer relief tank are blown out and become missiles, describe the associated hazards to the safety-related equipment and whether missile protection features are provided.

410.09 (9.1.2) Verify that the fuel pool is not located in the vicinity of any high energy lines or rotating machinery to ensure physical protection for the spent fuel from internally generated missiles and the effects of high energy line breaks.

410.10 (9.1.3) Identify any portion of the spent fuel pool cooling and cleanup system that is designed to nonseismic requirements and verify that failure of the nonseismic portion of the system will not affect the operation of the cooling trains.

410.11 (9.1.4) It is stated in a letter dated January 30, 1985 that the kinetic energy of the tool lifted to the maximum height exceeds the kinetic energy of the tool and a fuel assembly lifted to the normal height and that it has been demonstrated by a Westinghouse analysis that exceeding the bounds of the tool and fuel assembly analysis causes no adverse safety impact. Provide a copy of/or reference to this Westinghouse analysis.

410.12 (9.3.1) Provide the followng information for the compressed air system (a) A description of the means provided to verify that proper instrument air quality will be maintained over the plant life to assure the safety function of the system (i.e., air operated valves will perform their safety function). Include the air quality limits which should not be exceeded in order to assure the safety function, (b) Verification that the failure of any air operated valve to assume its fail safe position will not prevent the function of a safety-related system or compromise the ability to safely shutdown.

410.13 (9.3.3) Regarding internal flooding of safety related areas verify that adequate protection has been provided for safety-related equipment assuming the worst case flooding resulting from failures in high or moderate energy piping or postulated failure in nonseismic components (such as tanks). This protection cannot assume credit for nonseismic Category I sump pumps. Your response should include the time required for operator action if necessary to provide protection of essential equipment once indication from the Class 1E level switches is given.

410.14 (9.4.0) Describe the effects on the safety function of the essential HVAC systems in the event of a single failure of a fire damper. Such a failure should not compromise the safety function of the HVAC system. Adequate accessibility should be assured if credit is taken for manual reopening of the damper.

410.15 (9.4.0) Describe measures provided for detecting and correcting dust accumulation on safety-related equipment in order to assure their availability when needed. List any outside air intake for safety-related equipment which is less than 20 ft. from grade elevation.

410.16 (9.4.3) Discuss the charcoal filtration units in the mechanical auxiliary building HVAC (supplementary exhaust subsystem) comply with the guidelines of Positions C.1 and C.2 of Regulatory Guide 1.140.

410.17 (10.4.7 and 10.4.9) Verify that your preoperational test program will include tests to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Procedures for these tests should be provided for review.

410.18 (10.4.9) Provide a response to the staff's March 10, 1980 letter to near-term operating license applicants concerning your AFW system design (TMI-2 Task Action Plan, NUREG-0737, Item II.E.1.1). This response should include the following:

(a) A review of the AFW system design against Standard Review Plan Section 10.4.9, and Branch Technical Position ASB 10-1.

(b) A review of the AFW system design, Technical Specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter.

(c) The design basis for the AFW flow requirements and verifications that the AFW system will meet these requirements (refer to Enclosure 2 of the March 10, 1980 letter).

410.19 (10.4.9) The FSAR states that the AFW system reliability analysis will be provided later in Appendix 10A. This analysis when provided should be consistent with that described in NUREG-0611.

ADDITIONAL QUESTIONS - RSB

5.4.7 - RHRS

(1) What relief provisions are provided to accommodate thermal expansion of the water trapped between the two RHR inlet valves during heatup?

(2) What is the capacity of the RHR discharge line relief valves?

(3) Discuss the adequacy of the RHR pump bypass valve manual on-off controls. Other recent W plants have automatic recirculation control valve modulation based on total pump flow. Discuss the required operator actions.

(4) Will alarms be provided to indicate excessive RHR pump seal temperatures? (e.g. component cooling water outlet high temperature alarms?)

6.3 - ECCS

(1) The RWST volumes in Table 6.3-1 and Response 211.17 do not appear to be consistent. There appears to be a discrepancy of ~30,000 gallons in total volume. (561,000 gallons in Response 211.17 vs 532,000 gallons in Table 6.3-1). The "unusable volume" figures are also different. Clarify the discrepancies.

(2) With regard to switchover from injection to recirculation, how much time does the operator have to close the motor operated RWST isolation valve before pump NPSH is lost, assuming failure of the check valve in the RWST discharge line and the containment pressure shown in FSAR Figure 6.2.1.1-4 at time of switchover? Is this time period compatible with the recommendations of ANSI N660 (draft)?

(3) It is not apparent from Table 6.3-11 that all check valves that form the interface between the RCS and ECCS are leak tested. Some of the valve numbers in Figures 6.3-1 thru - 4 are illegible and thus can not be compared with the list in Table 6.3-11. Therefore we should go over this during the meeting at W.

SOUTH TEXAS PROJECT
HYDROLOGIC ENGINEERING SECTION
ENVIRONMENTAL REVIEW QUESTIONS

240.01 (ER) (2.5) Descriptions of floodplains, as required by Executive Order 11988, Floodplain Management, have not been provided. The definition used in the Executive Order is:

"Floodplain: The low land and relatively flat areas adjoining inland and coastal waters including at a minimum that area subject to a one percent or greater chance of flooding in any given year."

With this definition in mind, please provide the following:

1) Descriptions of the floodplain of Little Robins Slough for both pre-project and post-project conditions. On a suitable map, provide delineations of those areas adjacent to the plant, that will be inundated during the one percent (100 year) chance flood.

2) For Little Robins Slough, the Colorado River and West Branch Colorado River, describe how construction in the floodplain has affected the 100 year flood levels upstream of the site.

3) Identify, locate on a map and describe all structures and topographic alterations in the floodplains.

240.02 (ER) The average flow in the Colorado River at Bay City based on the period 1948 to 1970 was 2353 cfs. Please revise this value to include the period 1970 to present.

240.03 (ER) What is the status of the reservoir that has been proposed for construction near Columbus, Texas? Was the effect of this reservoir considered when you estimated the average flow in the Colorado River?

240.04 (ER) (3.3) What is the average amount of water that will be withdrawn from the Colorado River during normal operation? Of this amount, what percentage will be consumptively used?

240.05 (ER) (3.3) At the CP stage, the 7 day-10 year low flow was 1.0 cfs. Is this still a valid number?

The following questions are on the FSAR. However, responses to these are needed to enable the staff to prepare the Environmental Statements.

240.06 (FSAR) (2.4.8.2.2) You state that no significant erosion is expected in the spillway discharge channel due to flood flows. What is the expected velocity of flow in this channel? Provide assurance that the grassed channel will withstand this velocity.

240.07 (FSAR) (2.4.13.2.5) No sustained pumping (from the deeper aquifer) is permitted within a 4,000 ft. radius of the plant area. What is the purpose of this restriction? Is any pumping (other than sustained) permitted.

The staff is required to assess the impacts of a Class 9 accident. The following information is required to evaluate the liquid pathway effects of a core-melt accident.

240.08 In the event of a core melt accident, what would be the scenario as far as the relief wells and the cooling pond are concerned? Would the wells continue to operate and possibly introduce contaminated water to the surface or could the wells be sealed to prevent this?

240.09
(FSAR)
(2.4.13.3.2.1.1) The porosity and permeability of the lower shallow aquifer are given as 0.37 and 85 ft/day, respectively. How were these determined? Is the permeability value based on a pumping test or a lab test? Is it the average of several tests or is it the result of one test?

240.10
(FSAR) The porosity and permeability of the upper shallow aquifer are given as 0.35 and 14.17 ft/day, respectively. Please provide the same information as requested in question 240.09 above.

240.11
(FSAR)
(2.4.13.3.2.1.1
and
2.4.13.2.2.1.2) In determining groundwater velocity, you used total porosity. This is not correct. You should use effective porosity. Therefore, please provide the effective porosity values for the lower and upper shallow aquifers.

240.12
(FSAR)
(2.4.13.3.2.1.1 In determining groundwater velocity you used a gradient of 2.6×10^{-4} ft/ft for the lower shallow aquifer and 6.9×10^{-4} ft/ft for the upper shallow aquifer. How were these values determined.

240.13 (FSAR) (3.4.1.2)	You state that groundwater fluctuates between 2 ft and 10 ft below grade. How were these fluctuating levels considered in the gradients in question 240.10 above?
240.14 (FSAR)	Groundwater levels are shown in several FSAR figures. How will groundwater levels be affected by the relief wells? Will direction of groundwater flow be altered from what it presently is?

Environmental Engineering Section, EHEB

Requests for Additional Information

South Texas Project

Environmental Report - OL Stage

Draft Amendment 7

290.5 What is the source of Fig. 2.7-7 and what is the acreage of undeveloped prime farmland?

290.6 What agriculture or other management, if any, will be undertaken for prime farmland during operation?

290.7 What hunting, fishing, or other recreational use, if any, will be permitted on the site during operation?

290.8 How long will the special monitoring of alligators, eagles, deer, and waterfowl be continued? What changes, if any, in numbers or habits of these species have been noted, and to what extent are any changes attributable to the presence of the project? What additional changes are anticipated when the plant becomes operational and the cooling reservoir undergoes design temperature regimes?

290.9 What management, if any, will be undertaken for the Natural Lowland Habitat on the east side of the site? Will cattle grazing or herbicide use be permitted in this area?

290.10 What other onsite wildlife management or mitigation activities, if any, will be practiced during operation?

290.11 What changes, if any, have been noted in LRS water level, overall vegetation, indicator species, and salinities since 1978, and to what extent can any changes be attributed to the presence of the project?

290.12 How long will monitoring of the LRS be continued?

290.13 Please verify that transmission line maintenance procedures, particularly with regard to herbicide use, remain as described in the ER-OLS, Amendment 2, Section 5.6.1.

290.14 Please verify that the routes are as described in Section 3.9 of the ER-OLS. How near completion is the system?

290.15 Please document the national industry standards to which towers and other equipment are designed with reference where applicable to corona, leakage, interference, raptor protection, and other potential impacts. What grounding precautions have been or will be implemented for fences etc. under or near the lines?

291.10 (ER-OL Sec. 3.4.1.5) Provide estimates of the annual volume of sediments dredged from the intake settling basins and specify how the dredged material will be stabilized in the vicinity of the cooling reservoir makeup.

291.11 (ER-OL Sec. 3.6.1.1) Provide information on kind and quality of chemicals added to adjust wastes in the makeup demineralizer neutralization pit from a pH of 10.0 to 11.0 to pH 6-8.

291.12 (ER-OL Sec. 3.6.1.2) Identify the water quality standards that apply to discharge of the flushing water to the reservoir.

291.13 (ER-OL Sec. 5.4.4-1) Provide information on concentrations of constituents in the blowdown discharged from the cooling reservoir to the Colorado River.

291.14 Please provide the following references:

a. Sharik, T. L., P. V. Morgan, and R. D. Groover. 1974. An ecological study of the Lower Colorado River - Matagorda Bay area of Texas, Cyrus Wm. Rice Division, NUS Corp. (Pittsburg, PA, 1974). (Ref. 2.7-1).

b. Groover, R. D., T. L. Sharik, and P. V. Morgan. 1974. A report on the ecology of the Lower Colorado River - Matagorda Bay area of Texas, June 1973 through July 1974, R-24-05-09-1756, NUS Corp. (Rockville, MD, October 1974).

c. Groover, R. D., and P. V. Morgan. 1976. Final report - Little Robbins Slough aquatic ecological studies, April 1975-March 1976, R-32-00-12/76-656, Ecological Sciences Div., NUS Corp. (Houston, TX, December 1976). (Ref. 2.7-3).

d. Groover, R. D. and P. V. Morgan. 1976. Final report - Colorado River entrainment monitoring program, Phase One Studies - April 1975 - March 1976, R-32-00-12/76-676, Ecological Sciences Div., NUS Corp. (Houston, TX, December 1976). (Ref. 2.7-4).

e. Sharik, T. V., P. V. Morgan, and R. D. Groover. 1974. A report on the ecology of the Lower Colorado River - Matagorda Bay area of Texas, June 1973 through July 1974. Cyrus Wm. Rice Div., NUS Corp. (Pittsburg, PA, 1974). (Ref. 2.5-2).

291.15 What is the disposal method for trash removed from the trash rack by the trash rake? Is it discharged to the sluiceway? What is the intake velocity across the trash rack? How is the trash rack structure designed to minimize impingement of fish?

291.16 Page 3.4-2 define "excessive amounts of debris" and "excessive numbers of fish," i.e., what is the criteria for determining the discharge/disposal mode for trash and fish?

291.17 The ER-OL p. 3.4-2 states that the intake structure complies with criteria of the EPA. Specify and cite the EPA "criteria" for the intake structure.

291.18 Is there a permit for discharge of trash/fish to the river? Will there be any attempt through visual inspection of the traveling screens and screen rotation to minimize mortality of impinged organisms, e.g., fish?

291.19 Section 3.6.1 states that the makeup demineralizer water system (MDWS) is supplied by well water and will need no pretreatment. Section 3.3 states that the well water is treated with sulfuric acid prior to demineralization. Please resolve these differences.

291.20 Is the 49.500 gal of waste water produced when each train is regenerated an aggregate of the cation exchange unit, anion exchange unit, and the mixed-bed unit, or is this the volume of waste produced per unit? If an aggregate, specify the constituent streams and the frequency of occurrence.

291.21 What is the composition, the amount, and the concentration of the chemicals in the alkaline cleaning solution, including chelant, inhibitor, and surfactant? What is the composition and quantity of passivating solution in the chemical cleaning wastes (p. 3.6-3)?

291.22 What are the Texas Dept. of Water Resources standards for the Lower Colorado River?

291.23 Sects. 3.6.2 and 5.4.3. Is the chlorine residual remaining in the condenser effluent being discharged to the cooling reservoir total residual chlorine or free residual chlorine? What is the combined concentration of the chlorine residual discharged to the cooling reservoir from the condenser effluent and from the 20-minute, periodic chlorination of the circulating water intake structure and essential cooling water system intake structure?

291.24 What are the present projected concentrations of constituents in the blowdown di charged to the cooling reservoir?

291.25 What will be the area of the mixing zone with respect to the cross-sectional area of the river?