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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
FSAR Revisions Concerning
Section 3.9; Mechanical Systems and Components

Attached are revised FSAR sections 3.9.1.2, 3.9.1.4.7 and 3.9.3.1.2.3.1. These changes reflect the use of Bechtel computer program "ME 602" for merging response spectra curves and an update of applicable codes and standards.

HL&P believes that conclusions reached previously in the SER remain valid. If you should have any questions on this matter, please contact Mr. J. S. Phelps at (713) 993-1367.

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Deputy Project Manager

JSP/yd

Attachment: Revised FSAR Section 3.9

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3.9.1.2.2.21 CE035-BASEPLATE II - A description of the program is provided in Appendix 3.8.A. Documentation of the verification is maintained in the Bechtel Information Services Library.

3.9.1.2.2.22 CE413-WELD - The WELD program is used to size fillet welds for connections of wide flanges, tubes, pipes, angles, and channel. The program computes weld sizes based on AISC, NF, B31.1 and minimum weld for minimum heat transfer. The program is verified by hand calculations. 51

3.9.1.2.2.23 RELAP5/REPIPE - Thermal-Hydraulic Transient Analysis - RELAPS5/REPIPE is used for analysis of fluid transients in the piping system equations of conservation of mass, energy, and momentum are solved in one dimension for steam and/or water flow. The effects of noncondensible gas on steam/liquid flow are considered in the equations. REPIPE is the post processor which gives the forcing function for use in ME101. The program verification report is on file with Bechtel Data Processing.

3.9.1.2.2.24 ME150 FAPPS - FAPPS (Frame Analysis Program for Pipe Supports) is an inter-active computer program for the analysis and design of pipe supports. It optimizes member sizes, welds, baseplates and embedments based upon various user-specified design limitations. The program allows load combination by algebraic, absolute, or SRSS methods. The program has been verified against Bechtel Standard Structural Analysis Program CE901 (STRUDL) and hand calculations.

3.9.1.2.2.25 ME035 BASEPLATE - ME035 is a finite element-computer program for the analysis and design of baseplate. The program has important features like automatic mesh generation, availability of standard attachments, multiple plate thicknesses, and different printout options. The program has been verified against CDC Baseplate II (Bechtel CE035).

3.9.1.2.2.26 ME225 ANCHORPLATE ME225 is used to analyze and design interface anchors between non-seismic piping and seismically designed piping. Program has been verified by manual calculations.

ADD → 3.9.1.2.2.27 ME 602 SPECTRA MERGING & SIMPLIFIED SEISMIC ANALYSIS (See attached sheet)

3.9.1.3 Experimental Stress Analysis. Experimental stress analysis method has not been used for any seismic Category I ASME B&PV Code, Section III mechanical system or equipment. 50 Q21.39

3.9.1.4 Considerations for the Evaluation of the Faulted Conditions.

3.9.1.4.1 Stress Criteria for Class 1 Components and Supports in Nuclear Steam Supply System Scope: The structural stress analyses performed on the (NSSS Components and Components Supports) consider the loadings shown in Table 3.9-2.1. These loads result from thermal expansion, pressure, weight, OBE, SSE, design basis LOCA, and plant operational thermal and pressure transients.

3.9.1.4.2 Analysis of the Reactor Coolant Loop and Supports: The loads used in the analysis of the reactor coolant loop piping are described in detail below:

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3.9.1.2.27 ME602 SPECTRA MERGING & SIMPLIFIED SEISMIC ANALYSIS - This program merges response spectra curves, makes a neutral plot of these curves, and produces data files for ME101 Seismic Analysis. In addition the program calculates seismic span. Documentation of program verification is maintained in the Bechtel Information Services Central Library.

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3.9.3.1.2.1 Design Loading Combinations - The design loading combinations for ASME Code Class 1 BOP components and component supports are given in Tables 3.9-2.3 and 3.9-2.4. The design loading combinations for ASME Code Class 2 and 3 BOP components and component supports are given Tables 3.9-2.3A and 3.9-2.4. 53 52

3.9.3.1.2.2 Design Stress Limits - The design stress limits established for components are sufficiently low to ensure that violation of the pressure-retaining boundary does not occur and that the components operate as required. Stress limits for Class 2 and 3 components for each of the loading combinations are component-oriented and are presented in Tables 3.9-3, 3.9-4, 3.9-4A, 3.9-6A, and 3.9-7A for tanks, inactive pumps, active pumps, valves, and piping, respectively. Active pump and valve operability are discussed in Section 3.9.3.2. The component supports are designed in accordance with ASME Code, Section III, Subsection NF, and are discussed in Section 3.9.3.4. The stress limits for component supports are given in Table 3.9-7C. 53 41

3.9.3.1.2.3 Applicable Codes And Standards - The following codes and standards are used as a basis for the piping design which includes piping stress analysis and the design, fabrication, construction and testing of the pipe supports. Different issue dates of these documents may be used provided they meet the minimum requirements stated herein. Code cases and other standards are given in project design criteria and design specifications and in response to Q210.07N.

1. American Society of Mechanical Engineers (ASME) - ASME B&PV Code, Section III Subsections NA, NC, ND and NF, 1974 Edition, including Winter 1975 is used for piping stress analysis and piping support design. Other addenda used for piping stress analysis of Class 2 and 3 systems are: Paragraph NC/ND-3611.2 of Winter 1976 Addenda, NC-3652.3 of 1977 Edition of code, NC/ND-3622.5 of Winter 1978 Addenda and NC/ND-3658.3 of Summer 1979 Addenda., Containment mechanical penetrations have been designed in accordance with the requirements of NC-3200 and NC-3600 of the ASME Code, Section III, 1974 Edition with Addenda through Summer 1976 and NC-3217 of the Winter 1976 Addenda. 52 53

2. American National Standards Institute (ANSI) - ANSI B31.1 Power Piping Code, 1973 Edition including Winter 1975 Addenda.

3. American Institute of Steel Construction (AISC)

   A. "AISC - Specification for the Design, Fabrication and Erection Structural Steel for Buildings", 1969 Edition, including supplements 1, 2, and 3.

   B. "Specification for Structural Joints using ASTM A325 or ASTM A490 Bolts," 1969, including 1976 Addendum.

   C. "Code of Standard Practice for Steel Buildings and Bridges," 1972.

4. Manufacturers Standardization Society (MSS) MSS-SP-58-1975, "Hangers and Supports-Materials, Design and Manufacture."

5. MSS-SP-69-1976, "Pipe Hangers and Supports - Selection and Application."

NC-3652 thru NC-3655 of Winter 1981 Addenda, ND-3652 thru ND-3655 of Summer 1984 Addenda.

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of the reactor vessel and internals. The analysis is performed by numerically
intergrating the differential equations of motion to obtain the transient
response. The output of the analysis includes the displacements of the reac-
tor vessel and the loads in the reactor vessel supports which are combined
with other applicable faulted condition loads and subsequently used to calcu-
late the stresses in the supports. Also, the reactor vessel displacements are
applied as a time-history input to the dynamic reactor coolant loop blowdown 5
analysis. The resulting loads and stresses in the piping components and sup-
ports include both loop blowdown loads and reactor vessel displacements. Q22
Thus, the effect of vessel displacements upon loop response and the effect of Q11
loop blowdown upon vessel displacement are both evaluated. .1

3.9.1.4.6.6 Results of the Analysis - As described, the reactor vessel
and internals were analyzed for four postulated break locations. Table 3.9-12
summarizes the displacements and rotations of and about a point representing
the intersection of the centerline of the nozzle attached to the leg in which
the break was postulated to occur and the vertical centerline of the reactor
vessel. Positive vertical displacement is up and positive horizontal dis-
placement is away from and along the centerline of the vessel nozzle in the
loop in which the break was postulated to occur. These displacements were
calculated using a conservative break opening area for the postulated pipe
ruptures at the vessel inlet and outlet nozzles and double-ended ruptures at 41
the pump outlet nozzle and SG inlet nozzle locations. These areas are
estimated prior to performing the analysis. Following the reactor coolant
system structural analysis, the relative motions of the broken pipe ends are
obtained from the reactor coolant loop blowdown analysis. The actual break
opening area is then verified to be less than the estimated area used in the
analysis and assures that the analysis is conservative.

The maximum loads induced in the vessel supports due to the postulated pipe
break are given in Table 3.9-13. These loads are per vessel support and are
applied at the vessel nozzle pad. It is conservatively assumed that the maxi-
mum horizontal and vertical loads occur simultaneously and on the same sup-
port, even though the time-history results show that these loads do not occur
simultaneously on the same support. The peak vertical and horizontal load
occurs for a vessel inlet nozzle break. Note that the peak loads are conser-
vative values since the break opening area for the vessel inlet and outlet
nozzle break (as obtained from the dynamic loop analysis) is actually less
than the estimated upper bound area used to generate the applied loads. If 41
additional analysis was performed using the lower break opening area, the
loads would be considerably reduced. Furthermore, the peak vertical load and
peak horizontal load do not occur on the same vessel support. The largest
vertical loads are produced on the supports beneath and opposite the broken
nozzle. The largest horizontal loads are produced on the supports which are
the most perpendicular to the broken nozzle horizontal centerline.

3.9.1.4.7 Stress Criteria for Class 1 Components and Component
Supports for BOP Scope of Supply: All Class 1 components and supports have
been designed and analyzed for the design, normal, upset, emergency and
faulted conditions as specified in the rules and requirements of the ASME B&PV
Code, Section III. Stress criteria for Class 1 BOP valves and piping are 53
outlined in Tables 3.9-5 and 3.9-7. Stress limits for Class 1 BOP component
supports are given in Table 3.9-7B.

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and Table NB-3681(a)-1 of 1983 Edition of ASME Section III. 53

The Class 1 piping has been designed and analyzed for the design, normal, upset, emergency and faulted conditions in accordance with the requirements of NB-3600 of the ASME B&PV Code, Section III, 1974 Edition through Winter Addenda of 1975, NB-3658 of Summer Addenda of 1977, NB-3650 and NB-3680 of Summer Addenda of 1979. When the stresses as determined by the methods given in NB-3630 exceed the limits thereof, the design can be accepted provided it meets the requirements of NB-3200. The rules of NB-3630 meet all the requirements of NB-3200. 41

3.9.1.4.8 <u>Evaluation of the Control Rod Drive Mechanisms</u> - The Control Rod Drive Mechanisms (CRDMs) are evaluated for the effects of postulated reactor vessel nozzle limited displacement breaks. A time-history analysis of the CRDMs is performed for the vessel motion discussed in Section 3.9.1.4.6. 41
A model of the CRDMs is formulated with gaps at the upper CRDM support modeled as nonlinear elements. The CRDMs are represented by beam elements with lumped masses. The translation and rotation of the vessel head is applied to this model. The resulting loads and stresses are compared to allowables to verify the adequacy of the system. 5 Q22.2 Q110 .1
The highest loads occur at the head adaptor, the location where the mechanisms penetrate the vessel head. The combined effect including seismic loads is shown to be less than the allowable loads at this location. 41

3.9.2 Dynamic Testing and Analysis

3.9.2.1 <u>Preoperational Vibration and Dynamic Effects Testing on Piping</u>. Piping vibration tests will be performed during the initial test program to comply with the recommendations of Regulatory Guide (RG) 1.68 and satisfy the requirements of ASME B&PV Code, Section III.

3.9.2.1.1 <u>Nuclear Steam Supply System Scope</u>: A preoperational piping vibrational and dynamics effects testing program will be conducted for the reactor coolant loop/supports system during startup functional testing of the plant. The purpose of these tests will be to confirm that the system has been adequately designed and supported for vibration as required by Section III of the ASME Code, Paragraph NB-3622.3. The preoperational piping vibration and dynamic effects test program for the primary coolant loop system (this includes the hot legs, cold legs, cross-over legs, reactor coolant pumps, steam generators, and reactor vessel) at South Texas Units 1 and 2 is as follows:

1. The primary coolant loop system as defined above will be instrumented with accelerometers to measure the dynamic response of the system during normal and transient operating conditions. In addition to normal steady state operation, the test conditions will include steady state operation with various combinations of reactor coolant pumps in operation and transient conditions due to the starting and tripping of the reactor coolant pumps. 41 Q110 .27