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

DEC 23 1980

Docket Nos.: 50-329/330

NOTE TO: R. Bosnak
S. Pawlicki
W. Butler
F. Schauer

FROM: F. Miraglia

SUBJECT: REVIEW SCHEDULE FOR MIDLAND VESSEL HOLDDOWN MODIFICATIONS

The attached letter of December 10, 1980, in conjunction with several previous submittals forwarded to you, comprise the complete package of materials for a revised design concept for the reactor vessel support modification resulting from the preservice failure of holddown studs on Midland Plant, Unit 1. NRR is therefore in a position to complete its review of the revised support concept as requested by I&E's Task Interface Agreement RCI-80-01 dated May 14, 1980. NRR approval is tied to the applicant's construction schedule by virtue of its commitment to I&E that "... actual plant modifications to compensate for the defective bolts will not be started on Unit 1 until approval of the design concept is received from NRR." The applicant's schedule is such that review completion of the attached report is needed by the middle of January 1981. This will provide two weeks for meetings to resolve any staff concerns such that final NRR approval can be issued by February 1, 1981, the date by which the construction hold needs to be lifted if the construction schedule is not to be impacted.

The form of the staff approval to be issued February 1, 1981 will be a letter report to the applicant addressing the results of the staff review of the revised support concept in a format similar to that used to indicate approval of topical reports. The matters to be addressed should correspond to those of the Standard Review Plan applicable to the revised support concept which are typically reviewed prior to issuance of a construction permit. A construction permit level of review is deemed sufficient for purposes of lifting the construction hold. Staff approval of final detailed analyses will be made as part of our continuing FSAR review.

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The previous stud failures and revised concept are for Unit 1. The situation is different for Unit 2 and NRR is requested by the applicant to confirm the acceptability of the Unit 2 studs as is and without any fix. Nevertheless, it is the applicants present intent if practicable to modify Unit 2 to be similar to the Unit 1 support design. If practical, staff confirmation for Unit 2 should be indicated in the same report to be issued February 1, 1981 for the Unit 1 modification.

Please contact Darl Hood (28402) should you encounter difficulties with these schedules or require additional background reports. The more significant background reports are the six references listed in Enclosure 1, Enclosure 1 itself, and three technical reports by Teledyne Engineering Services.

Frank Miraglia

Frank Miraglia, Acting Chief
Licensing Branch 3
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Enclosure:
CPCo letter 12/10/80

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SUBJECT: Interim deficiency rept re reactor vessel Anchor bolt failure, completing design package of proposed mods. Prestress levels of studs have been lowered. Next rept will be sent by 810331.

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Consumers Power Company

James W Cook
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December 10, 1980

Mr J G Keppler, Regional Director
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MIDLAND PROJECT -
UNIT NO 1, DOCKET NO 50-329
UNIT NO 2, DOCKET NO 50-330
UNIT NO 1, REACTOR VESSEL BROKEN ANCHOR BOLT -
FILE 0.4.9.35 UFI 73*10*01, 02111(S), 21114(E) SERIAL 9787

References:

1. S H Howell Letters to J G Keppler; Midland Nuclear Plant; Unit No 1, Docket No 50-329, Unit No 2, Docket No 50-330; Unit No 1 Reactor Vessel Broken Anchor Bolt;
   a. Howe-311-79; dated December 14, 1979
   b. Howe-267-79; dated October 12, 1979
   c. Howe-51-80; dated March 3, 1980
   d. Howe-80-80; dated April 30, 1980
2. J W Cook letter to J G Keppler; Midland Nuclear Plant; Serial 8971; dated May 16, 1980
3. J W Cook letter to J G Keppler; Midland Nuclear Plant; Serial 9330; dated July 24, 1980
4. J W Cook letter to J G Keppler; Midland Nuclear Plant; Serial 8809; dated August 1, 1980
5. NRC (D S Hood) letter to CP Co, dated July 7, 1980, Subject: Summary of May 23, 1980 Meeting on Preservice Failure of Three Reactor Vessel Hold-Down Studs
6. J G Keppler letter to S H Howell, Docket No 50-329 and 50-330, dated August 18, 1980

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References 1, 2 and 4 were interim 50.55(e) reports, as is this letter, concerning broken anchor bolts in the Unit 1 reactor vessel support skirt. Reference 3 provided interim technical information concerning the reactor pressure vessel support modification and the schedule for the accomplishment of that modification. In Reference 5, the NRC requested a detailed description of the analytical techniques being used to assess the modified NSSS support system. Enclosure 1 to this report provides the requested information.

Enclosure 2 provides the status of actions taken to resolve this condition. Another 50.55(e) report, either interim or final, will be sent on or before March 31, 1981.

Reference 6 transmitted the NRC investigation report regarding the reactor vessel anchor bolt failures. Further, Reference 6 specified that "...actual plant modifications to compensate for the defective bolts will not be started on Unit 1 until approval of the design concept is received from NRR." Reference ~~4~~ 5 to this letter generally stated NRR staff concurrence with the design concept, and also alluded to the schedule and type of further information submittals. The attached report and our previous submittals comprise the complete package of materials describing the design concept. Based on the current procurement and fabrication schedule underway, we request that the Staff complete their review of the attached report by the middle of January. Immediately following the review, it is the Company's intent to meet with NRR to resolve any staff concerns, and thereby obtain formal recognition that the condition specified in your letter of August 18 (Reference 6) has been met. The final NRR approval is required by February 1, 1981 in order to support our construction schedule. If the Staff has any concerns from our previous report (Reference 4), we would appreciate being notified as soon as possible so that they can be resolved.

James W. Cook

WRB/cl

Enclosures: (1) Report entitled, "Reactor Pressure Vessel Support Modification for Midland Nuclear Power Plant, Midland, Michigan, Report No 2," dated October 1980

(2) MCAR-37, Interim Report #4, dated November 5, 1980, entitled, "Broken Reactor Vessel Anchor Studs in Unit 1"

CC: Director, Office of Inspection & Enforcement
Att Mr Victor Stello, USNRC (38)

Director, Office of Management
Information & Program Control, USNRC (1)

RCook, USNRC Resident Inspector

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REACTOR PRESSURE VESSEL

SUPPORT MODIFICATION

FOR

MIDLAND NUCLEAR POWER PLANT

MIDLAND, MICHIGAN

REPORT NO 2

DECEMBER 1980

CONSUMERS POWER COMPANY

JACKSON, MICHIGAN

rp1280-0036a112

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SUPPORT MODIFICATION

FOR

MIDLAND NUCLEAR POWER PLANT

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1.0 INTRODUCTION

This report provides a description of the analytical techniques that will be used in the analyses of the Midland Unit 1 Reactor Vessel modified support system. This report is a continuation of the report submitted to the NRC in July, 1980 entitled, "Reactor Pressure Vessel Support Modification for Midland Nuclear Power Plant, Midland, Michigan, Preliminary Report No 1."

2.0 REACTOR VESSEL SUPPORT DESIGN CRITERIA

The design criteria for the reactor vessel support system are those stated in the previous report, therefore please refer to Section 2.0 of the July 1980 Report for the discussion on this topic.

3.0 ANALYTICAL PROCEDURES

3.1 GENERATION OF SUPPORT LOADS

3.1.1 TECHNICAL BASIS

The methodology used to generate the design loads for the modified Nuclear Steam Supply System (NSSS) supports will utilize the same analytical techniques and computer codes as used in developing the B&W's Owners Group Report entitled, Effects of Asymmetric LOCA Loadings, BAW 1621 B&W 177-FA, (Reference 2) which has been submitted to the NRC for review in July 1980.

Modifications will be made to the existing mathematical models of the NSSS and its supports to incorporate the upper lateral support spring rates, reactor vessel anchor stud spring rates, internal wall structures, and boundary conditions at the reactor coolant pumps and steam generators specific to the Midland Plant. The seismic forcing functions are Midland specific, however the LOCA forcing functions (ie, cavity pressurization, and reactor internal differential pressures) used to determine the support loadings are based on larger breaks than those specifically applicable to Midland.

The analyses will incorporate techniques (described herein) which insure that all components supporting, and attached to, the reactor vessel will receive a full review for structural integrity under the modified support design.

3.1.2 MATHEMATICAL MODEL

It is assumed that the initial loads to which the Reactor Vessel (RV) and its supports are subjected will not produce component yielding. Therefore, model construction and subsequent analyses are based on linear analytical techniques. The validity of these assumptions is assured by comparing the linearly derived dynamic stresses to allowable stresses for a linear analysis.

In describing the mathematical model which will produce the final loads on the NSSS supports it is convenient to discuss the model as three integrated components; the NSSS, the internal wall structures, and the NSSS supports attached to the internal wall structures.

3.1.2.1 NSSS MODEL

Because of the complexity of the RV loading conditions and the number of attachments to the vessel, a detailed isolated model of this component is constructed. This model is a complete representation of the reactor vessel and its appendages (eg, control rod drive mechanisms, service support structure, and reactor internals). It also incudes the hot legs extending to the steam generators and the cold legs extending to the pumps for loops A and B. Boundary conditions are imposed at the ends of the pipes where they connect to the components to simulate the remainder of the NSSS. The isolated model is shown in Figures 12 through 15.

The isolated portion of the NSSS is modeled utilizing finite beam-element and lumped mass representations of each component. Finite element methods are used where necessary to define the structural characteristics of components such as the fuel and plenum assemblies. Once determined by finite element techniques, the structural characteristics of components are used to generate the equivalent finite-beam element and lumped mass representations: The criteria for developing the equivalent structural representation is that component stiffness and frequency must be retained.

The various components that make up the total RV and its internals are identified in Figure 16. By comparing Figure 16 with the lumped-mass model shown in Figure 13, the correlation between the components and the model elements representing them can be seen.

In addition to the structural representation of the components, the NSSS mathematical model incorporates the effects of fluid coupling between components into the overall structural response of the system. This is accomplished by develping a mass matrix using the height of concentric cylinders, the distance between the cylinders, and various parameters describing the fluid between the cylinders. The mass matrix which is generated is combined with the diagonal mass matrix terms defining component mass distribution to generate a full system mass matrix.

3.1.2.2 INTERNAL WALL STRUCTURES

The internal wall structural model properties included, are the area, shear area, area moments of inertia, modulus of elasticity, and Poisson's ratio for different elevations in the wall. Lumped masses at different elevations define the mass distribution and mass resistance of the wall structure. The internal wall structure is modeled to the center of the concrete basemat and the boundary conditions at that point are fixed such that no relative rotation or translation is allowed. The internal wall structure model is shown in Figure 17.

3.1.2.3 <u>NSSS SUPPORTS</u>

For the isolated RV model, the NSSS supports can be described as the boundary conditions imposed on the cold leg piping at the pumps and the hot leg piping at the steam generators, the reactor vessel skirt support, and the upper lateral supports near the RV flange.

The boundary conditions imposed on the reactor coolant piping at the pumps and steam generators consist of stiffness matrices that represent the characteristics of the structures to which the pipes are attached. They are obtained from a full system model by disconnecting the pipes at the component nozzles and computing a stiffness matrix of the remaining component with its supporting structures and other attached piping.

The reactor vessel skirt support is modeled as a boundary condition at the base of the RV skirt support in the form of a set of springs. The boundary conditions reflect the flexibility of the anchor studs, localized concrete flexibility, and overall flexibility of the RV pedestal from the RV skirt support to the center of the basemat.

The Upper Lateral Support (ULS) tie the RV to the internal wall structures. ULS structural properties are incorporated into equivalent beams with end conditions reflecting the axial load carrying ability of the supports and appropriate cross sections properties to reflect the support flexibility.

Localized concrete deformation is included in the considerations of the support flexibility. The ULS equivalent beams are shown in Figures 12, 13, and 17 as they connect the RV with the internal wall structures.

3.1.3 LOAD CASES ANALYZED

The isolated model will be subjected to four load cases in the process of determining the design loads on the supports. Two sets of seismic analyses will be performed; one for the Operating Basis Earthquake (OBE) and the other for the Safe Shutdown Earthquake (SSE). Two Loss of Coolant Accidents (LOCA) cases will be considered; a guillotine at the hot leg outlet of the RV and a guillotine at the cold leg inlet to the RV. The support system is designed such that the ULS receive no deadweight or thermal loads from the RV. Deadweight and thermal loads for the RV lower support have been previously computed and will not be affected by the support modifications.

3.1.4 METHOD OF ANALYSIS

3.1.4.1 SEISMIC FORCING FUNCTIONS

The seismic forcing functions that will be applied to the mathematical model consist of response spectra curves for SSE at damping values from 1% to 5%. Response spectra is supplied for earthquakes in five directions, North-South, East-West, vertical, rotation about North-South, and rotation about East-West. The rotation is applied as occurring about the geometric center of the RV at the elevation of the basemat.

3.1.4.2 <u>LOCA FORCING FUNCTIONS</u>

LOCA forcing functions are composed of three sets of time histories which are applied simultaneously to individual degrees of freedom. The forcing functions are the result of blowdown into the cavity between the RV and the primary shield wall, and pressure wave propagation inside the RV due to the break in the reactor coolant pressure boundary.

<u>Core Bounce</u>

The vertical response of the reactor internals and Fuel Assemblies (FA) result in a time varying force composed of the structural response to differential pressures. Core bounce is the terminology given to this response phenomina. The nonlinear structural response reflecting holddown springs and vertical gaps is calculated in a decoupled analysis. The FA core and reactor internals are simulated with a planar model consisting of beam elements, nonlinear axial springs, and lumped masses. The ANSYS code is used to calculate the vertical reactions of the core, which are then used as applied force time histories on the reactor vessel in the system dynamic analysis. The core bounce LOCA forcing functions are the result of the worst case possible double end guillotine pipe breaks at the RV nozzel.

Thermal Hydraulics and Linear Dynamic Response

The pressure waves through the RV produce several reactions that are not considered in the core bounce forcing functions and which can be applied directly to a linear dynamic system.

For the reactor vessel, the horizontal pressure gradient results in horizontal forces on the RV, core support cylinder, thermal shield, and the plenum cylinder. The vertical gradient results in vertical forces on the RV.

The integration of the pressure-time history defines the time history forces which are applied to discrete mass joints of the mathematical model.

The thermal hydraulic loadings applied directly to the linear dynamic model are the result of a hot leg pipe rupture and a cold leg rupture.

Asymmetric Cavity Pressures

Pipe ruptures which occur in the cavity between the RV and the wall result in differential pressures across the RV in a time varying manner. The differential pressures, when integrated across the area of the RV, produce time varying forces which are applied to discrete mass joints on the RV. The cavity pressure loadings on the RV for these analyses are produced by the Architect-Engineer and are the result of mass and energy data from single ended pipe guillotine ruptures.

3.1.4.3 COMPUTER CODES USED FOR NSSS ANALYSIS

The two analytical computer programs and the four data reduction codes used in the seismic and/or LOCA analyses for the support design loads are described herein.

Structural Analysis Codes

1. HYDROE - A computer code used in calculating the hydrodynamic mass coupling of concentric cylinders.

2. STALUM - A computer program for analyzing three-dimensional, finite segment systems consisting of uniform or nonuniform bar/piping segments, closed-loop arrangements, and supporting elements. STALUM performs both static and dynamic structural analyses undergoing small linear, elastic deformations. The

static analysis is based on the matrix displacement method. The static loadings are static mechanical forces, thermal, and/or support displacement loadings. The dynamic analysis is based on lumped-mass and normal-mode extraction techniques. The dynamic input loadings can be response spectra or time history forcing functions.

The essential input to the program consists of the physical properties of the system, the boundary conditions, and/or the loading information; the essential output consists of the resultant joint displacements, rotations, forces, moments at both ends of each segment, and stresses at various locations in each segment.

Data Reduction Codes

1. FTRAN - A computer code used for Fourier analysis of forcing functions to determine the frequency content of the forcing function.

2. S1235 - A post-processor program used to tabulate forces, moments, displacements, and rotations in a specification format.

3. INTFCE - A program used to convert pressure-loading data to force-loading data acceptable for use by the structural analysis codes.

4. LOPL - A post-processor program used to provide time history tabulations and plots of spring forces and resulting loads and displacements.

3.1.5 SEISMIC ANALYSIS

Utilizing the geometric and structural properties of the mathematical model shown in Figures 13 thru 15, and,17, the STALUM code is used to determine the structural frequencies and mode shapes of the isolated NSSS, the internal wall structures,and the NSSS supports as a coupled system. Each degree of freedom (DOF) in the model is assigned a damping value based on the location and type of component the DOF represents. Strain energy damping is used to determine a composite damping for each mode. The modal accelerations are applied to the model dynamically to reflect the structural amplification. Equivalent static forces for each mode are determined and applied to each DOF to give resulting modal displacements and member forces. The modal responses for each individual earthquake will be combined, and the individual member responses will be combined by taking the square root of the sum of the squares (SRSS) results of all six components. Figure 18 shows the flow diagram for the seismic analysis.

RV Support Anchor Loads

The seismic loads on the RV support are taken directly from the seismic analyses and are the forces and moments from the combined five earthquakes at the base of the RV skirt. These centerline loads are resolved into support loads for the stress evaluation described in Section 3.2.

ULS Loads

The combined five earthquake ULS load is distributed in a worst case manner to obtain a maximum load for an individual support member for which each is designed. The combined earthquake dynamic load on the equivalent beams representing the ULS in the mathematical model is given as the total horizontal primary shield wall load.

3.1.6 LOCA ANALYSIS

The geometric and structural properties of the mathematical model are used to determine the mode shapes and frequencies of the structure in the same manner as in the seismic analysis. The four sets of LOCA forcing functions are applied simultaneously to individual DOF's to represent the structural loadings to the components during the LOCA event. Modal displacement and member force responses are determined for each mode and the modal results are combined by direct algebraic summation. The resulting displacements and member forces and moments are stored such that time for time or peak results are available for any member or joints.

RV Support Loads

The peak forces and moments, regardless of their time of occurrance, will be obtained from the time history LOCA analysis output, and used as the total centerline load imposed by the RV on the support.

ULS Loads

The LOCA loads are determined in a fashion similar to the seismic loads. The peak LOCA horizontal dynamic load is distributed in a worst case manner to determine the peak individual ULS load for which each will be designed. The total horizontal force on the equivalent beams representing the ULS will be given as the maximum load on the primary shield wall.

3.1.7 PRELIMINARY DESIGN LOADS

B&W has performed preliminary analysis using the upper lateral support along with a conservatively assumed zero pretension loaded anchor studs. The load cases analyzed were SSE and a B&W identified worst case LOCA involving a hot leg guillotine at the RV. The analysis was done assuming the upper lateral supports in contact with the reactor pressure vessel. The loads transmitted from the RV to the support system at the RV skirt and the upper lateral support are given below.

RV SKIRT

	F (kips)(1)	F (kips)	M (ft-kips)(2)	M (ft-kips)(1)
SSE	114	233	147	1,646
LOCA	1,003	3,347	3,529	1,113

UPPER LATERAL SUPPORT

(RADIAL LOADS)

	Total Wall Load (kips)	Maximum Individual ULS Load (kips)
SSE	166	55
LOCA	3,377	1,126

(1) Treated as a shearing load on the shear pins and keys provided in the RV skirt to pedestal connection.

(2) M is in effect, the overturning moment.

3.2 ANALYSIS OF THE RV SUPPORTS FOR THE FINAL LOADS

3.2.1 ANALYSIS OF ANCHOR STUDS

The RV anchor stud stress analysis has assumed that the studs would resist the tensile forces in the base that result from vertical uplift forces and from overturning moments. Horizontal shears and the torsional moments are transferred from the RV skirt flange to the 5-1/2 inch thick sole plate by 48 shear pins, and then transferred by shear lugs to the concrete pedestal (See Figures 8 thru 11).

The determination of the stud stresses for the final loads will be performed by means of a finite element analysis. The finite element model will include the RV skirt and flange represented by shell elements, along with boundary spring elements to simulate the anchor stud tensile stiffness, compressive stiffness of the concrete, and the shear pins embeded in the sole plate. The broken studs in Unit 1 will be accounted for by omitting the tension boundary springs at their corresponding node point locations. The reactions from the vertical uplift forces, overturning moments, and horizontal shears will be resolved into discrete nodal loads at the top of the RV skirt model. The applied reaction forces will be oriented such that the maximum tensile stresses in the studs will occur in the neighborhood of the broken studs. The stud prestress forces will be simulated by equivalent compressive forces applied to the base nodes representing the stud locations.

The finite element program being used to assess the stud stresses produces only linear solutions. The analysis will require a number of iterations to achieve a balanced solution. The analysis will be initiated with the neutral axis coinciding with the geometric center of the RV skirt flange. After the loads are applied, the boundary node point stresses will be checked to verify that neither the studs have exceeded their prestress force nor the concrete bearing stress has exceeded its allowable. If either of these conditions are not true, then adjustments will be made to the position of the neutral axis by either declaring more studs with tension loads above the prestress, and/or smaller areas of concrete capable of resisting bearing loads. This iterative process will be continued until the boundary stresses are balanced.

3.2.2 ANALYSIS OF UPPER LATERAL SUPPORTS

The ULS bracket analysis assumes that the bracket would resist both the compressive loads from seismic and LOCA forces on the RPV and the bending loads from upward pressurization of the shield plugs. The preliminary analysis indicates that the maximum anticipated pressurization load applied to the brackets in addition to the preliminary axial load produce stresses well within the range of allowable stresses. The trapezoidal ULS bracket will be assessed by taking sectional properties at several locations along the length. The allowable yield stress for the steel will be reduced at each section to account for the higher temperature according to the AISC 1971 code edition, that will occur from having the bumper in contact with the RPV. The thermal analysis assumes an RPV surface temperature of 580 F and 16,000 cfm airflow at 130 F. The results of the temperature analysis of the ULS indicate that the exposed edge of the primary shield wall

(Point A on Figure 6) will be 248 F, and the concrete behind the ULS embedment (Point B on Figure 6) will be heated to 159 F. The material used to strengthen the bracket, according to the current preliminary design, will be the same material used to fabricate the bracket, which is ASTM A516 Grade 70 steel. The shim material will be ASTM A240 Type XM-19 stainless steel.

4.0 STATUS OF ANALYSIS AND DESIGN

4.1 FINAL SUPPORT LOAD GENERATION

The analysis by B&W incorporating the final mathematical representation of the modified boundary conditions to simulate the ULS and the reduced stud prestressing is in progress. Results verifying the design will be submitted to the NRC upon completion of the analyses.

4.2 ULS DESIGN AND FABRICATION

The preliminary upper lateral support design has been completed and the structural drawings are being prepared to procure the material and proceed with fabrication.

The final design of the ULS has not started but will begin shortly after the loads have been developed. It has been anticipated that the final loads will be less than the capacity of the bracket since their design is based upon a conservatively estimated set of preliminary loads.

The existing brackets, which will be a part of the ULS design, were originally designed to support the cavity annular shield plug at El 632'. The layout and details of the ULS brackets are shown in Figures 3 thru 7.

As shown in Figures 5 and 6, the additional stiffness required by the ULS will be obtained by adding steel plates to the bottom flange and to either side of the top flange. The clear distance between the brackets and the RV varies between 1-1/4 and 6-1/2 inches. This gap will be shimmed tight with both the RV and the ULS in the hot operating condition. A shimming procedure is currently under development to measure the thermal displacements of the ULS and RV in order to establish the required shimming distance. A method of measuring the change in the gap between the Reactor Pressure Vessel (RPV) and the bracket end that will work in the extreme environmental conditions of the hot functional test is being developed for use.

5.0 STUD DETENSIONING STATUS

The Unit 1 studs were detensioned in order to preclude further failures and are currently at a nominal stress level of about 6 ksi as recommended in TES Report TR-3887-2, Rev 1 (Reference 1). The detensioning procedure is also being evaluated to ensure that the limits of accuracy of the measured stud stress levels are compatible with the criteria of Reference 1.

The stud detensioning procedure that was used required that the liftoff values be recorded. These values are shown in Table 1 and Figure 2, and exhibit a certain amount of scatter. A consultant specializing in the field of tensioning behavior and tensioning systems is being retained to establish the possible reasons for this scatter as well as to comment on the procedure used to tension and detension the studs to assure that the 6 ksi prestress design allowable will not be exceeded. The recommended criteria for establishing an allowable short-term stress was established in TR-3887-2, Rev 1 (Reference 1) and included in Section 3.2.2 of the first report of this subject. With this criteria in mind, the studs that had a recorded liftoff stress of less than 75 ksi were proof-test tensioned to 75 ksi for several minutes so that a value of half of the indicated tensile load, 37.5 ksi, could be used as an allowable short-term stress.

6.0 CONCLUSION

This report has described in detail the modeling techniques being used in the analyses of the modified reactor vessel support system for the Midland Nuclear Power Station. These methods represent the standard techniques utilized by the NSS suppliers for primary system analysis under the various design conditions. The design modification is mandatory for Unit 1 because of the anchor stud failures experienced. Based on the investigations conducted, the Company originally recommended using the Unit 2 reactor vessel support design in its original condition although this matter is still under review with the NRC staff. However, it is the Company's intent if practicable to modify the Unit 2 design with upper lateral supports to be similar to the Unit 1 support design. Analyses for Unit 2 with upper lateral supports will also be carried out using the techniques described in this report with the appropriate changes being made to the input data to properly represent the Unit 2 configuration.

This report provides information regarding the detailed analytical techniques which fulfill the Company's understanding of the material necessary for final NRC review and concurrence of the reactor vessel support design modification concept. The design of the upper lateral supports has proceeded using preliminary design loads as described in the report. The supports are conservatively designed with respect to these preliminary loads and will be able to withstand loads in excess of those anticipated from the final analyses. The confirmation of the adequacy of the design will be made upon receipt of the final support loads. Appropriate status reports and final analytical results will be submitted in the future to document the completion of the detailed design.

7.0 REFERENCES

1. Teledyne Engineering Services Report, TR-3887-2, Rev 1, "Acceptability for Service of Midland RPV Anchor Studs," May 20, 1980.

2. BAW 1621 B&W 177-FA Owners Group, "Effects of Asymmetric LOCA Loadings", Phase II Analysis, July 1980.

3. Consumers Power Company, "Reactor Pressure Vessel Support Modification for Midland Nuclear Power Plant, Midland, Michigan, Preliminary Report No 1," July 1980.

TABLE 1

DETENSIONING DATA

UNIT 1 REACTOR VESSEL ANCHOR STUDS

Stud Number (2)				Hydraulic	
Sequence	B&W	Teledyne	Date	(psig) 1	Pressure Bolt Stress to Nearest ksi
1	01 in	37 in	4-08	13,000	88
2	02 in	13 in	4-23	11,900	81
3	03 in	01 in	4-25	13,400	91
4	04 in	25 in	5-19**	9,300	63*
5	01 out	37 out		8,000	54*
6	02 out	13 out		12,500	85
7	03 out	01 out		10,800	73*
8	04 out	25 out	5-12	8,400	57
9	05 out	43 out	5-13	12,500	85
10	06 out	19 out	5-13	12,500	85
11	07 out	07 out	5-13	13,400	91
12	08 out	31 out	5-14	13,800	94
13	05 in	43 in	5-14	12,300	83
14	06 in	19 in	5-14	11,500	78
15	07 in	07 in	5-15	12,000	81
16	08 in	31 in	5-15	11,400	77
17	09 in	40 in	5-16	12,300	83
18	10 in	16 in	5-16	11,700	79
19	11 in	04 in	5-19	13,700	93
20	12 in	28 in	5-19	12,400	84
21	09 out	40 out	5-20	12,200	83
22	10 out	16 out	5-20	12,500	85

TABLE 1 (Continued)

	Stud Number (2)			Hydraulic	
Sequence	B&W	Teledyne	Date	Pressure (psig) 1	Bolt Stress to Nearest ksi
23	11 out	04 out	5-20	13,000	88
24	12 out	28 out	5-21	12,300	83
25	13 out	46 out	5-21	12,800	87
26	14 out	22 out	5-21	11,500	78
27	15 out	10 out	5-21	12,300	83
28	16 out	34 out	5-22	12,600	85
29	13 in	46 in	5-22	11,100	75
30	14 in	22 in	5-22	12,100	82
31	15 in	10 in	5-23	9,300	63*
32	16 in	34 in	5-23	13,100	89
33	17 in	38 in	5-23	11,600	79
34	18 in	14 in	5-27	9,500	64*
35	19 in	02 in	5-27	13,300	90
36	20 in	26 in	5-27	9,600	65*
37	17 out	38 out	5-28	12,500	85
38	18 out	14 out	5-28	12,300	83
39	19 out	02 out	5-29	14,000	95
40	20 out	26 out	5-29	12,100	82
41	21 out	44 out	5-30	12,200	83
42	22 out	20 out	5-30	12,300	83
43	23 out	08 out	6-17	12,300	83

TABLE 1 (Continued)

Sequence	Stud Number (2) B&W	Stud Number (2) Teledyne	Date	Hydraulic (psig) 1	Pressure Bolt Stress to Nearest ksi
44	24 out	32 out	6-18	12,300	83
45	21 in	44 in	6-18	12,800	87
46	22 in	20 in	6-18	10,900	74*
47	23 in	08 in	6-19	12,300	83
48	24 in	32 in	6-19	12,400	84
49	25 in	41 in	6-20	12,200	83
50	26 in	17 in	6-20	11,800	80
51	27 in	05 in	6-20	13,000	88
52	28 in	29 in	6-23	12,800	87
53	25 out	41 out	6-23	12,500	85
54	26 out	17 out	6-24	12,700	86
55	27 out	05 out	6-24	8,900	60*
56	28 out	29 out	6-25	12,500	85
57	29 out	47 out	6-25	10,200	69
58	30 out	23 out	6-25	12,200	83
59	31 out	11 out	6-26	12,200	83
60	32 out	35 out		B R O K E N	
61	29 in	47 in	6-26	11,900	81
62	30 in	23 in	6-27	12,400	84
63	31 oin	11 in	6-27	11,800	80
64	32 in	35 in	6-27	11,600	79
65	33 in	39 in	7-02	11,700	79

TABLE 1 (Continued)

Sequence	Stud Number (2) B&W	Stud Number (2) Teledyne	Date	Hydraulic Pressure (psig) 1	Bolt Stress to Nearest ksi
66	34 in	15 in	7-02	11,700	79
67	35 in	03 in		B R O K E N	
68	36 in	27 in	7-03	12,300	83
69	33 out	39 out	7-03	12,100	82
70	34 out	15 out	7-03	12,300	83
71	35 out	03 out	7-07	12,000	81
72	36 out	27 out	7-07	10,300	70*
73	37 out	45 out	7-07	12,600	85
74	38 out	21 out	7-08	12,500	85
75	39 out	09 out	7-08	12,200	83
76	40 out	33 out	7-08	13,600	92
77	37 in	45 in	7-09	13,000	88
78	38 in	21 in	7-09	11,500	78
79	39 in	09 in	7-09	12,200	83
80	40 in	33 in	7-10	13,200	90
81	41 in	42 in	7-10	11,800	80
82	42 in	18 in	7-10	12,500	85
83	43 in	06 in	7-11	10,200	69*
84	44 in	30 in	7-11	12,300	83
85	41 out	42 out	7-11	12,200	83
86	42 out	18 out	7-14	10,400	71*
87	43 out	06 out	7-14	11,800	80

TABLE 1 (Continued)

Sequence	Stud Number (2) B&W	Teledyne	Date	Hydraulic Pressure (psig) 1	Bolt Stress to Nearest ksi
88	44 out	30 out	7-14	11,700	79
89	45 out	48 out	7-15	13,100	89
90	46 out	24 out	7-15	10,400	71*
91	47 out	12 out	7-15	11,700	79
92	48 out	36 out		B R O K E N	
93	45 in	48 i.	7-16	12,500	85
94	46 in	24 in	7-16	11,900	81
95	47 in	12 in	7-16	12,100	82
96	48 in	36 in	7-17	11,700	79

NOTES:

1) Ram area of tensioner = 27.134 sq in, bolt area = 4.00 sq in.

2) Refer to Figure 1 of Reference 3 for the locations of the studs.

*) Proof loaded to 75 ksi after detensioning.

**) Tensioner run up to 14,200 psig/96 ksi on initial attempt without being able to rotate nut. Lift-off data shown are results of detensioning attempt after 20th in sequence.

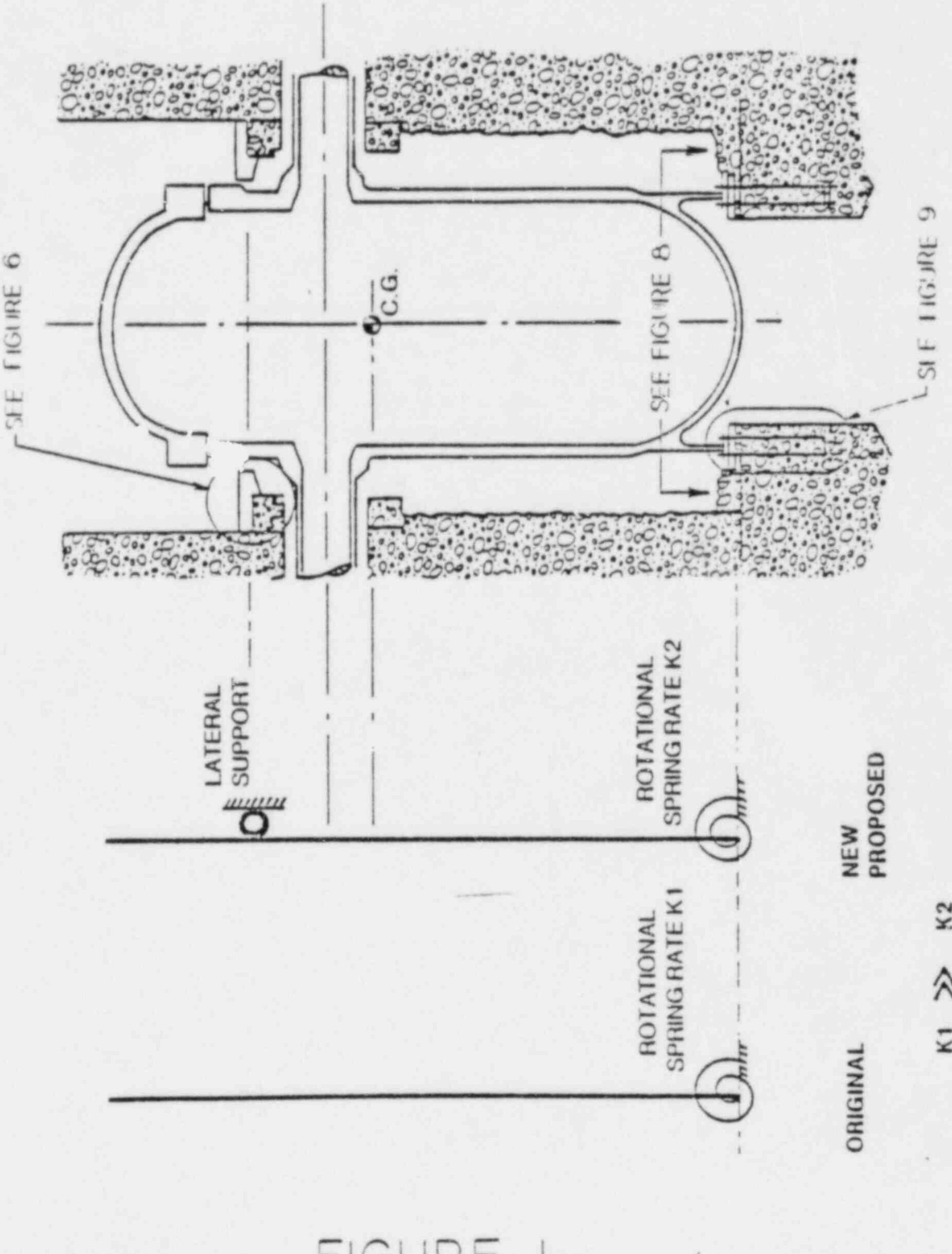


FIGURE 1

UPPER LATERAL SUPPORT CONCEPT

MIDLAND JOB 7220
UNIT-1
LIFT-OFF VALUES DURING
DETENSIONING
APRIL 8 1980 TO JULY 17, 1980
TOTAL: 93 STUDS

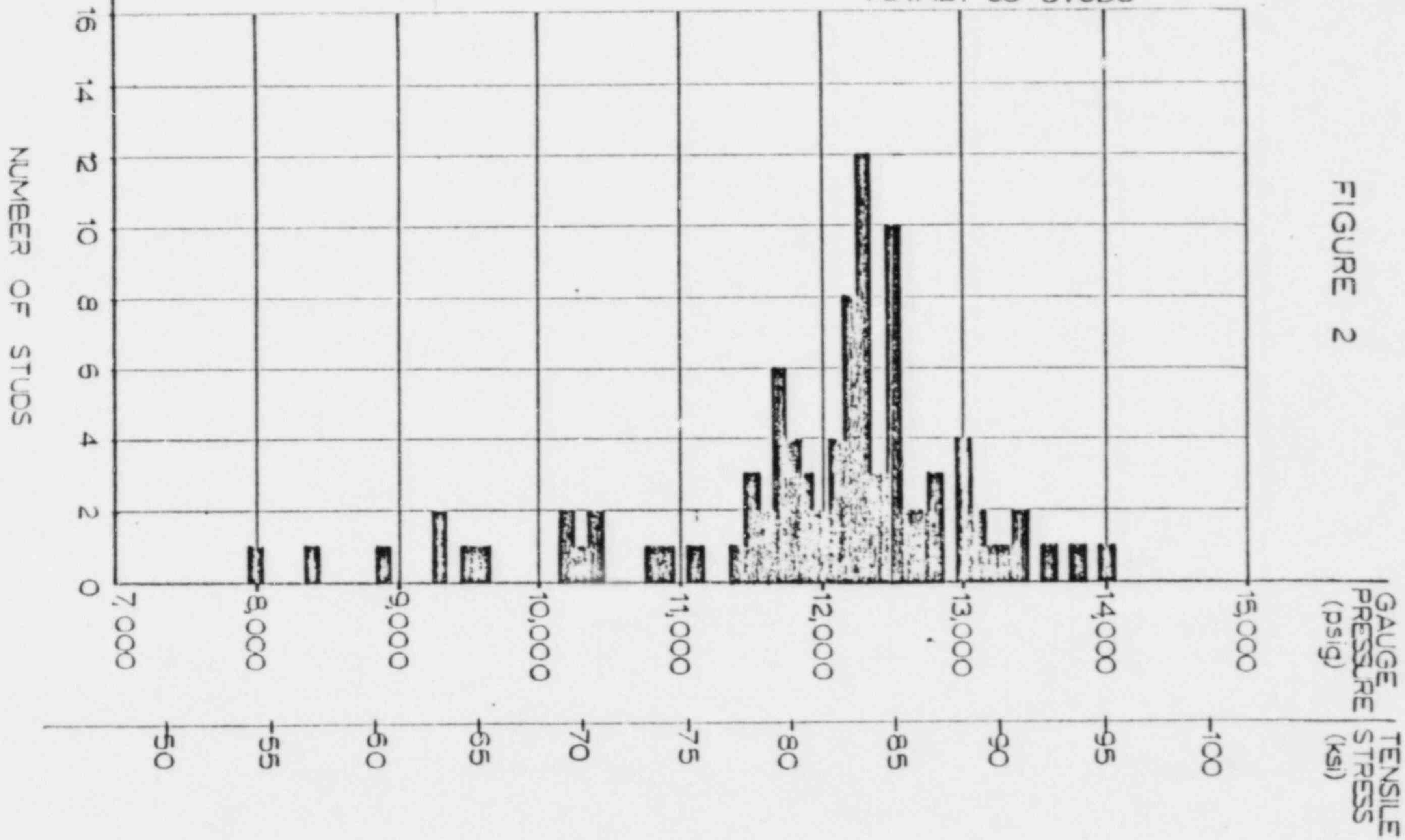


FIGURE 2

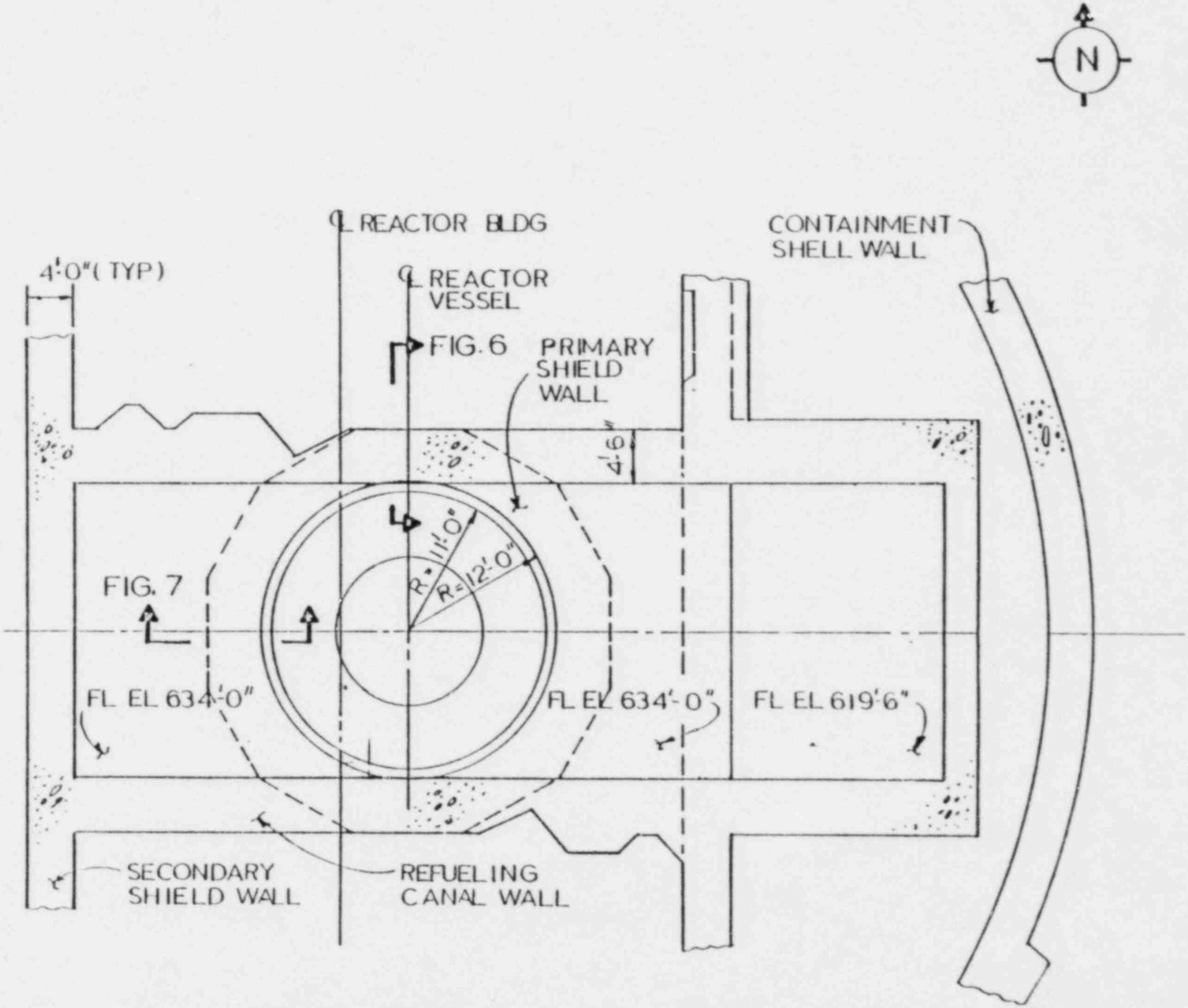
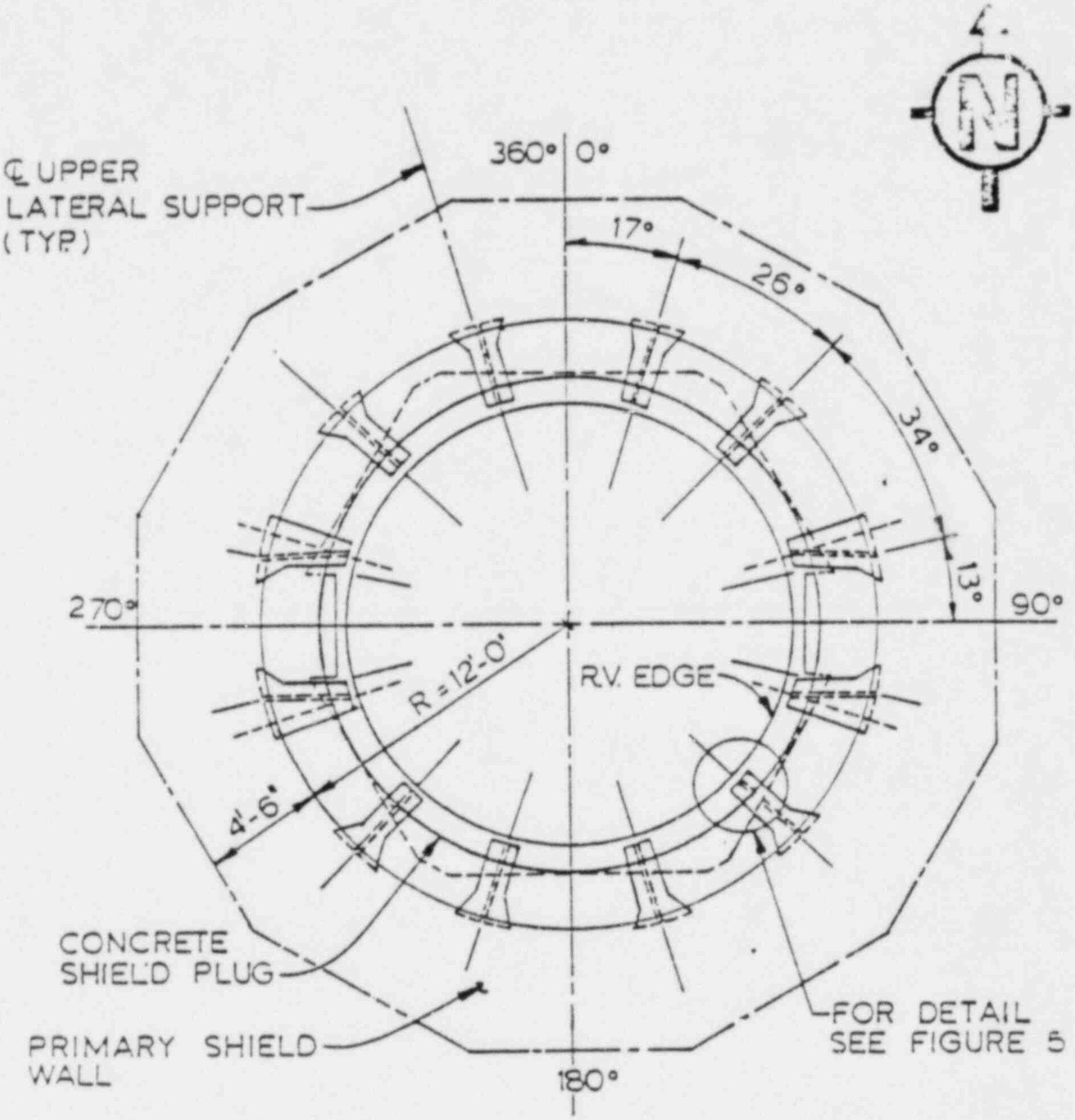
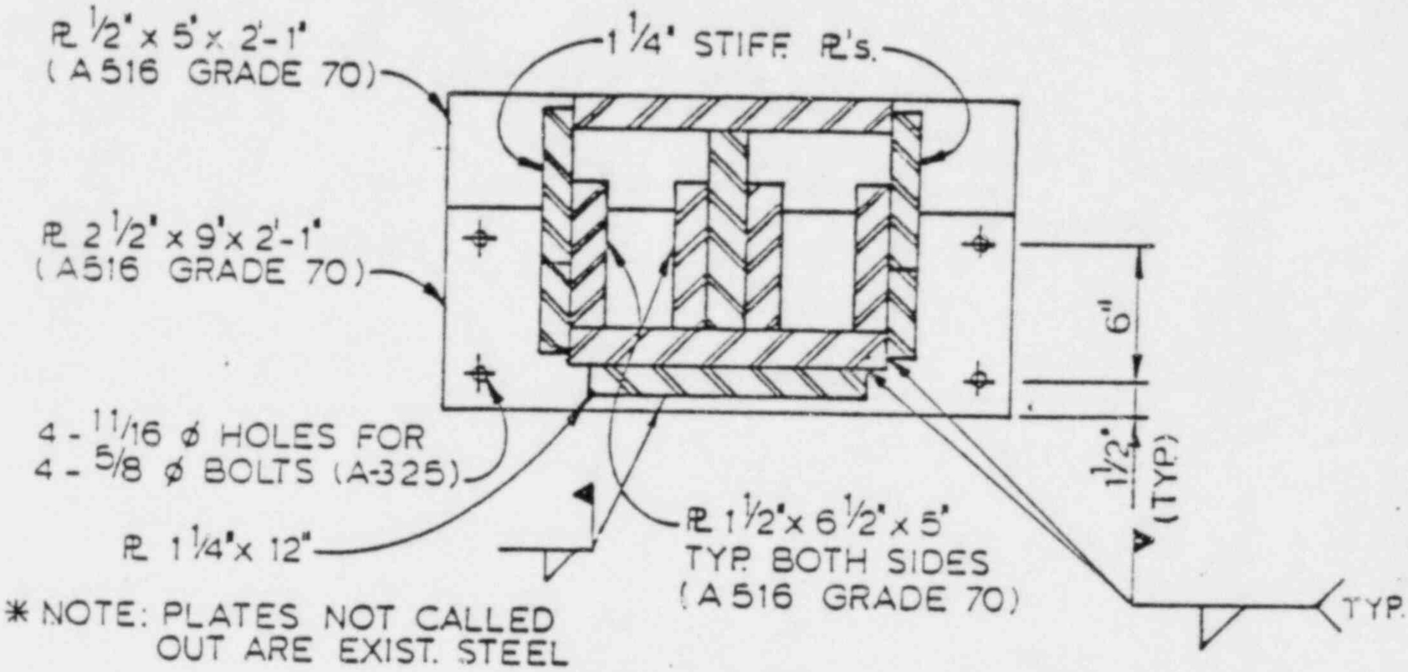


FIG. 3. WALLS AND SLAB PLAN AT ELEV. 636'-0"

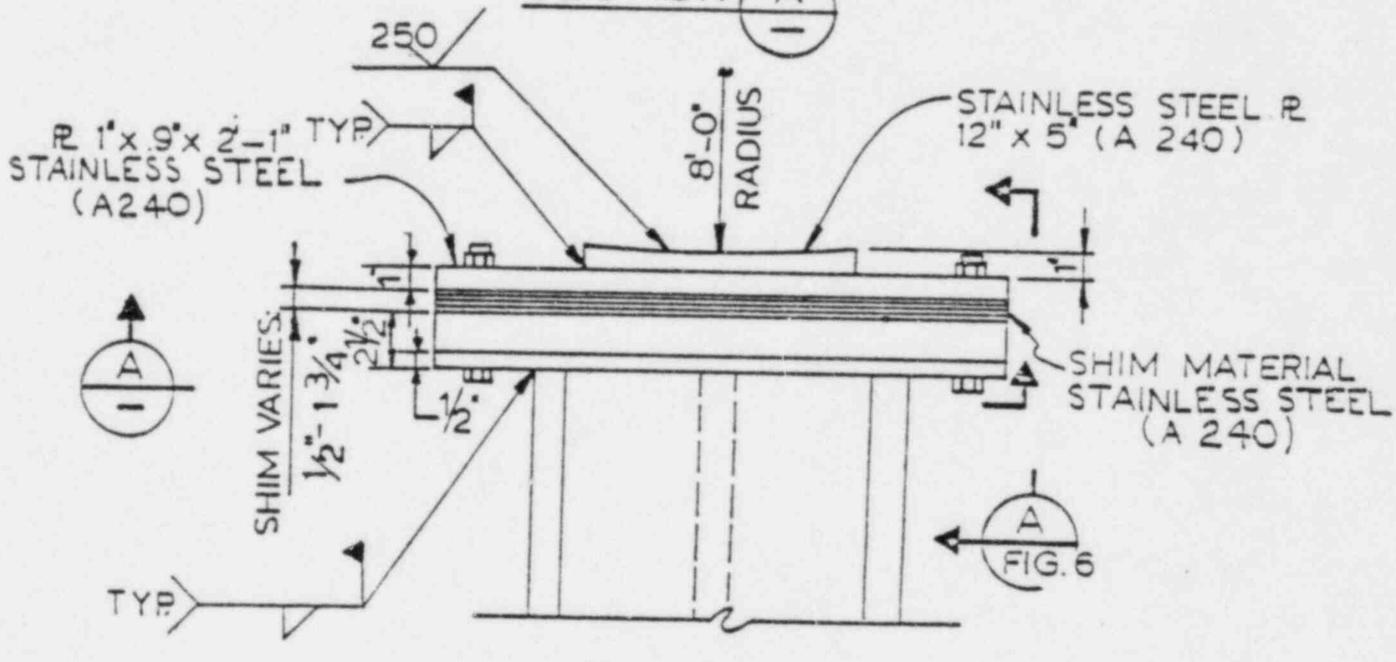


UPPER LATERAL SUPPORT PLAN
FIGURE 4

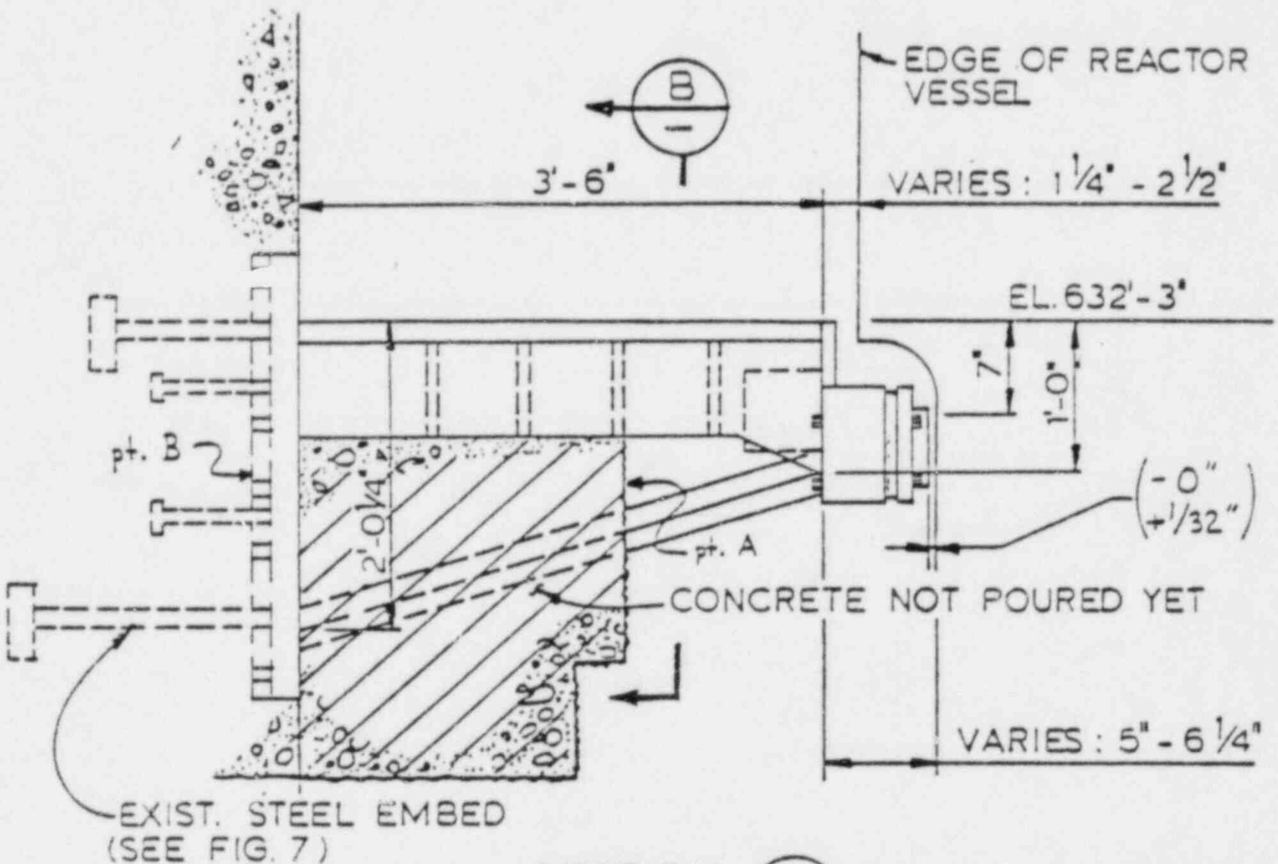
REACTOR PRESSURE VESSEL
UPPER LATERAL SUPPORT
FIGURE 5



SECTION A

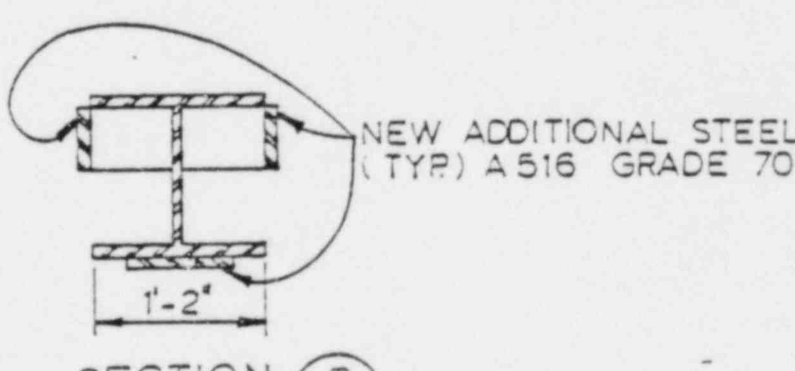


PLAN AT EL. 632'-3"

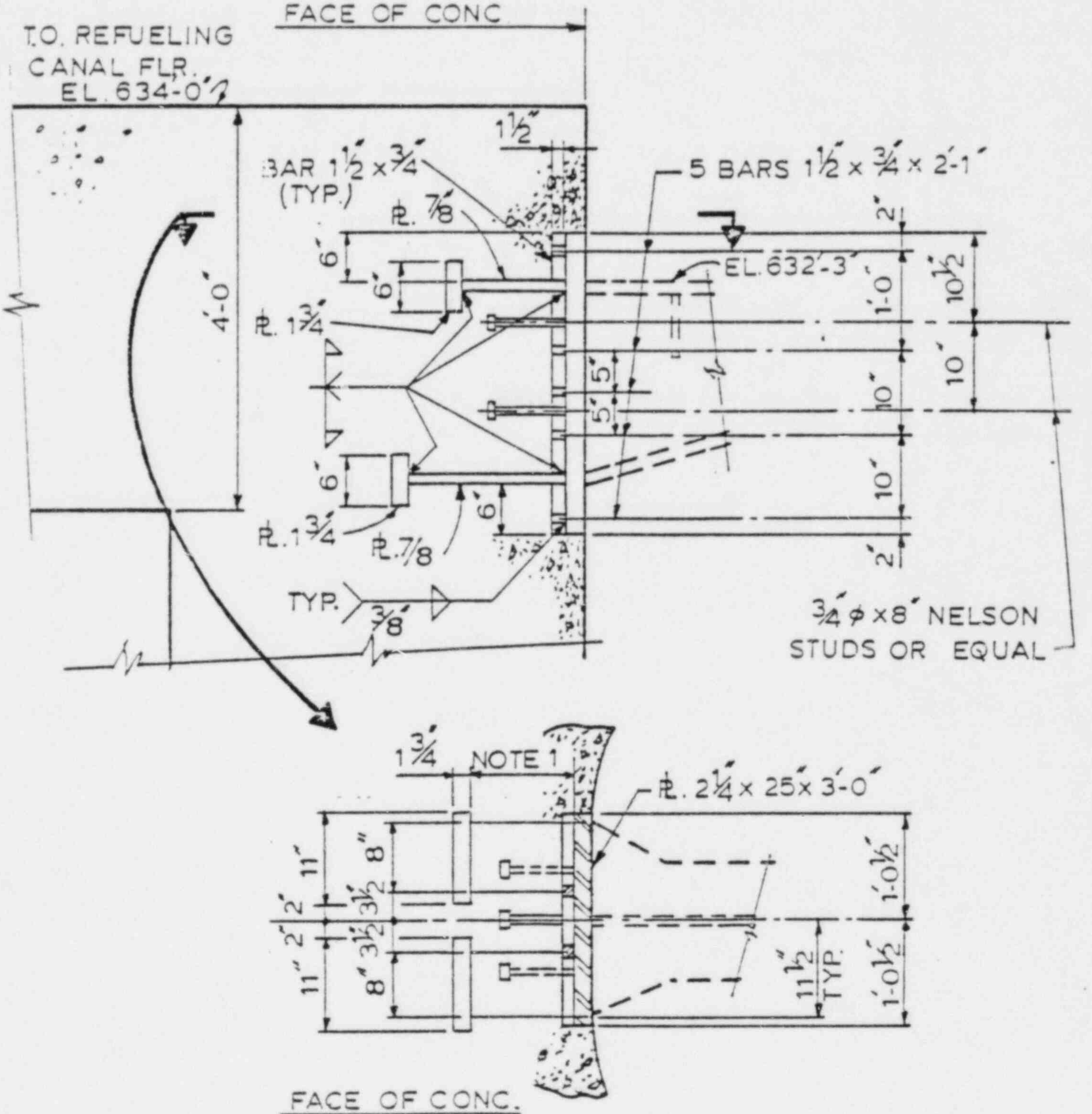


SECTION A FIG. 6

UPPER LATERAL SUPPORT DETAIL
FIGURE 6

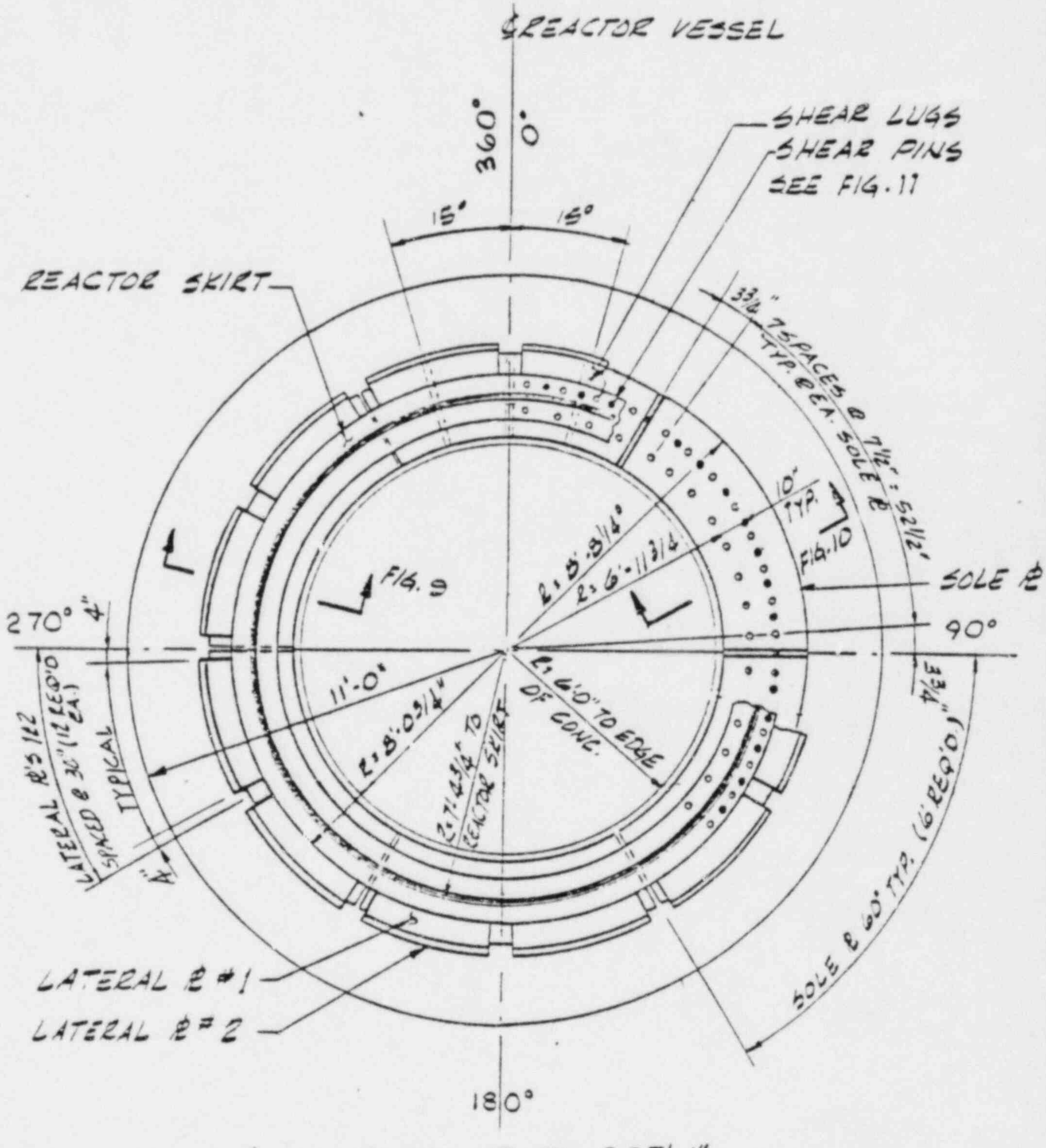


SECTION B



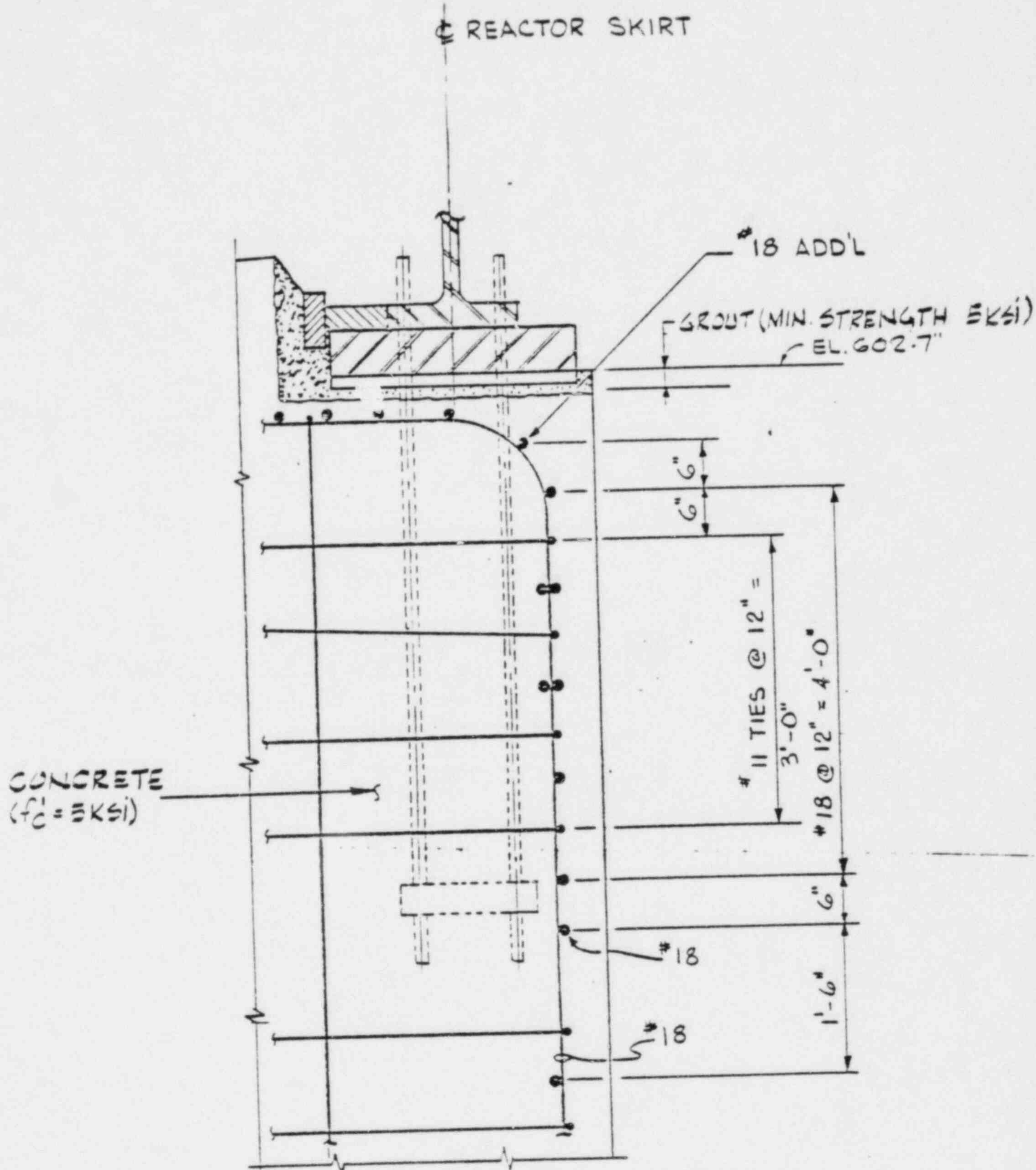
NOTE 1
THIS DIMENSION
1'-0" FOR TOP ANCHOF
1'-6" FOR BOT. ANCHOR

FIGURE 7



PLAN AT EL 603'-1"

FIGURE 8



CONCRETE PEDESTAL DET.

FIGURE 9

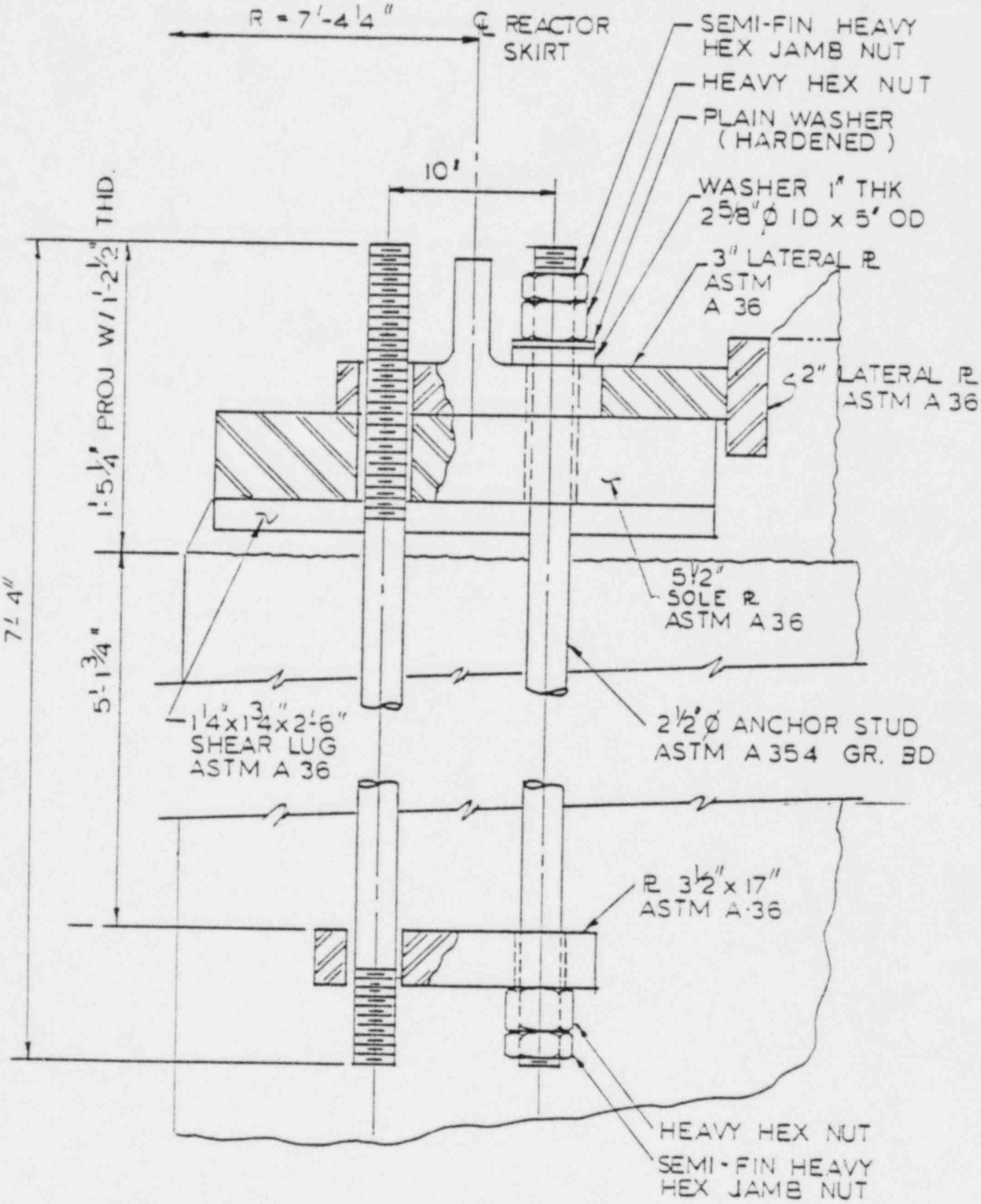
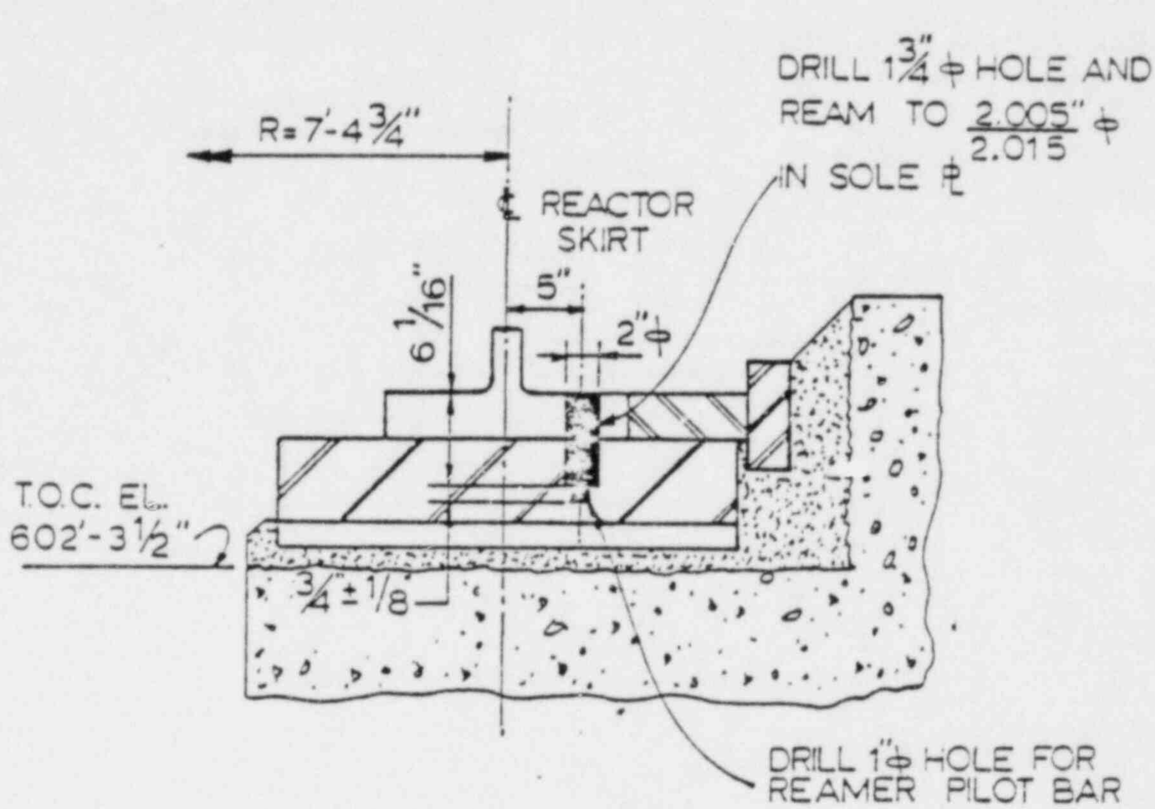
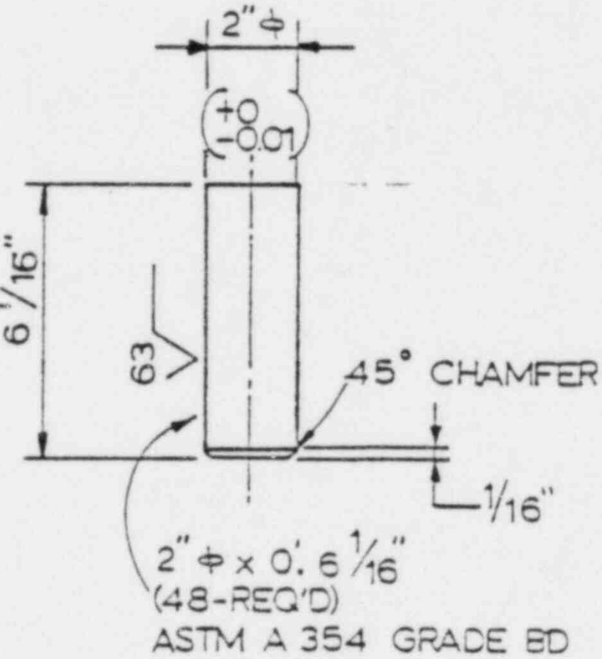


FIG. 10 ANCHOR STUD DETAIL

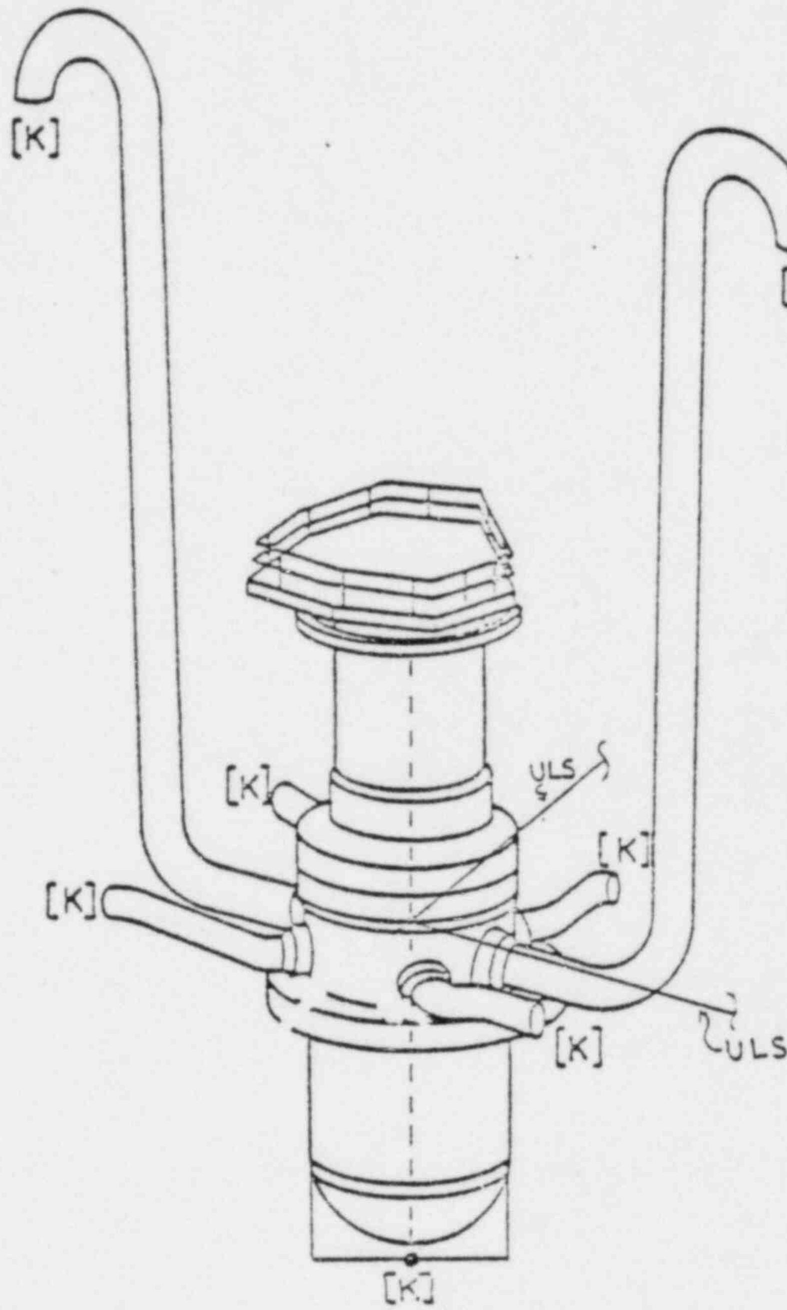




SHEAR PIN DETAIL

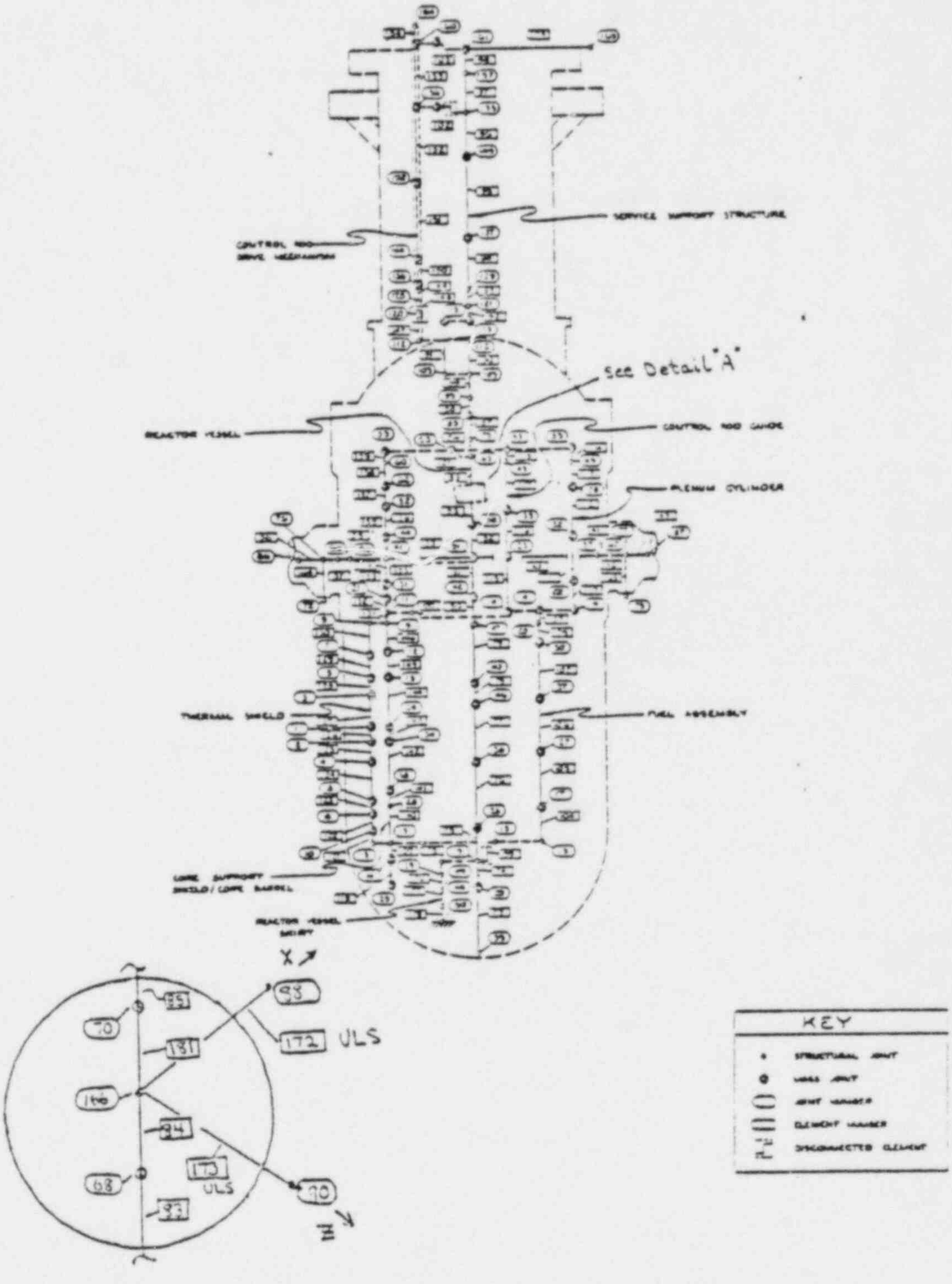
FIGURE 11

FIGURE 12 Reactor Coolant System Boundaries



[K] = Stiffness matrix

FIGURE 13 RV Isolated Model, Reactor Internals and SSS



Detail "A"

FIGURE 14 RV Isolated Model, Skirt-Supported Plant, Plan View

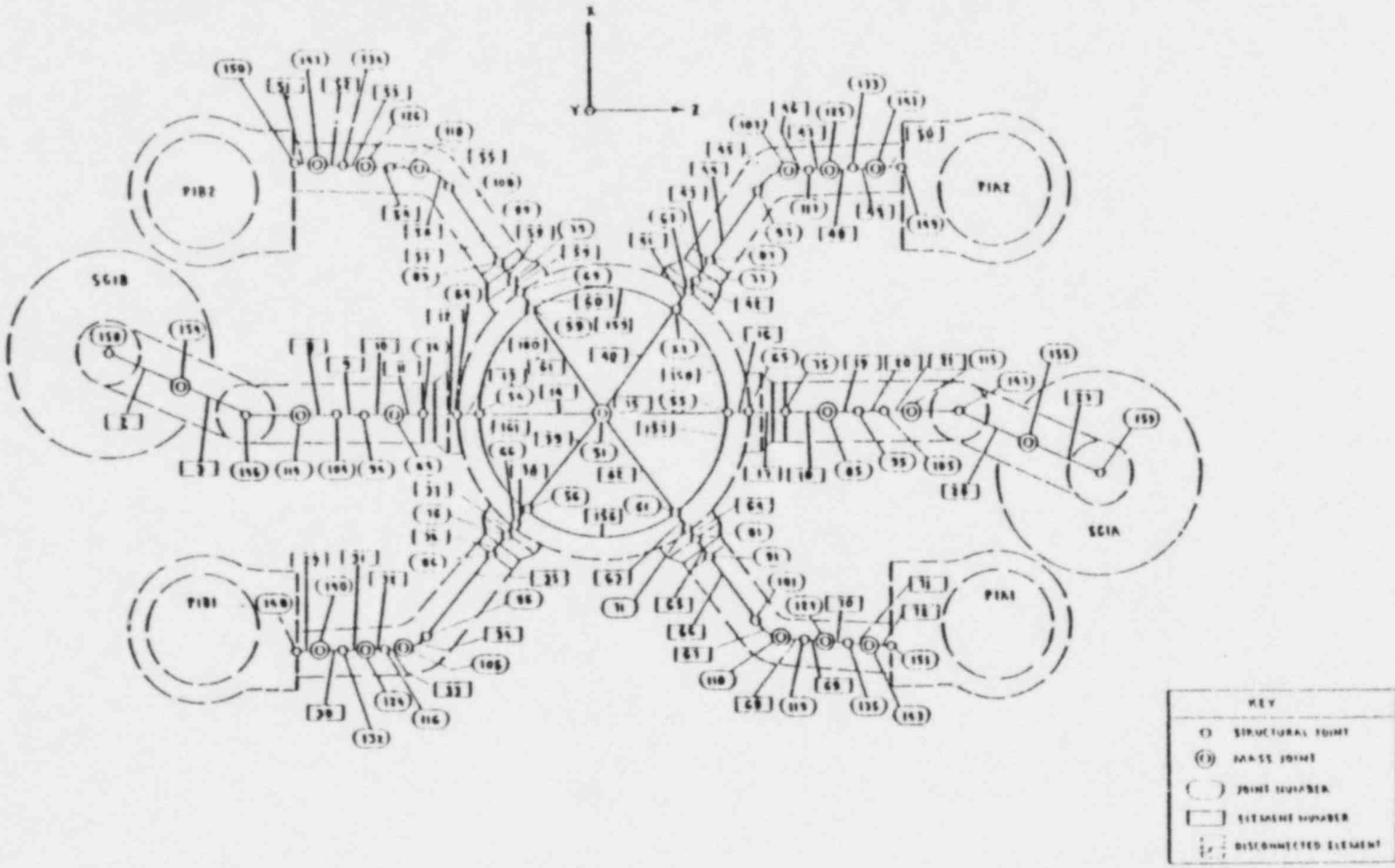


FIGURE 15 RV Isolated Model, Elevation
View A-A, Hot Leg

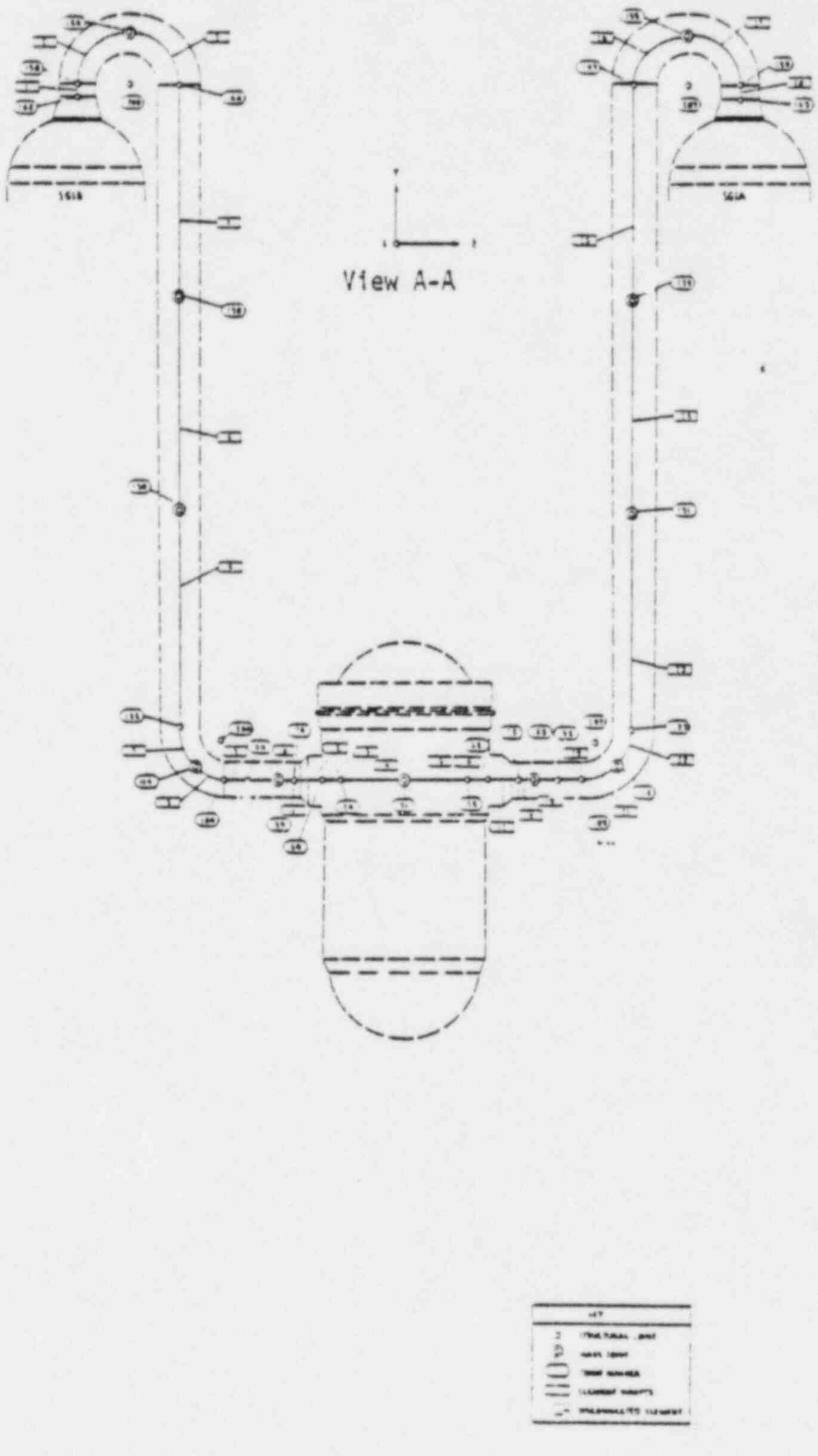


FIGURE 16 Reactor Internals and Service Support Structure

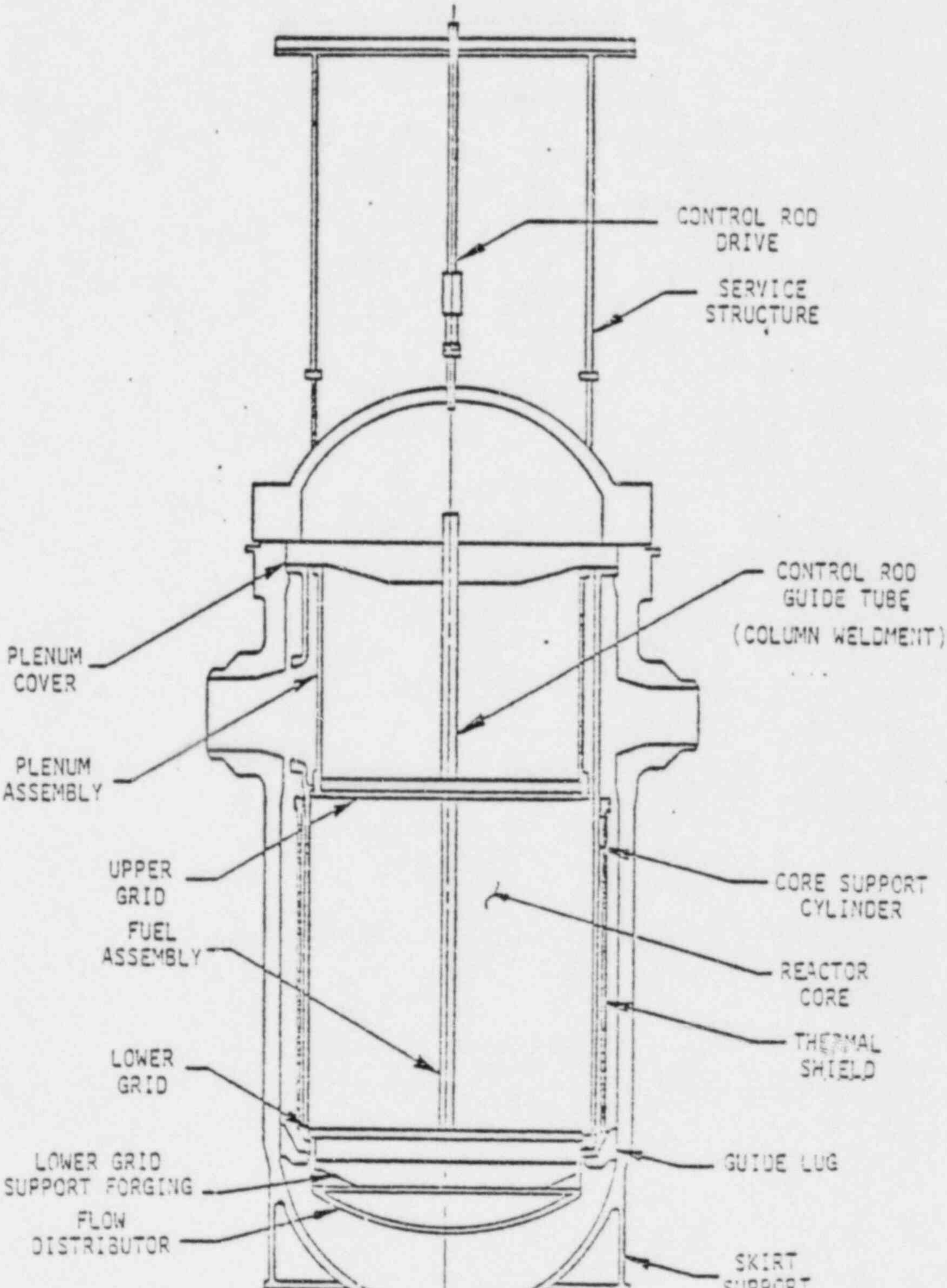
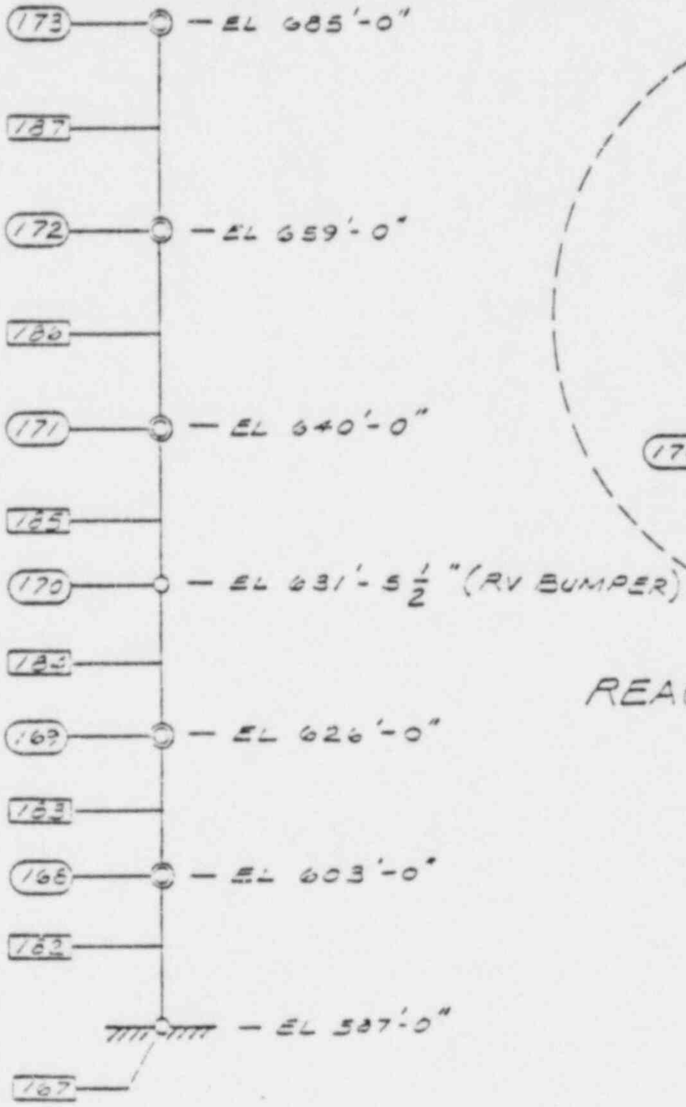
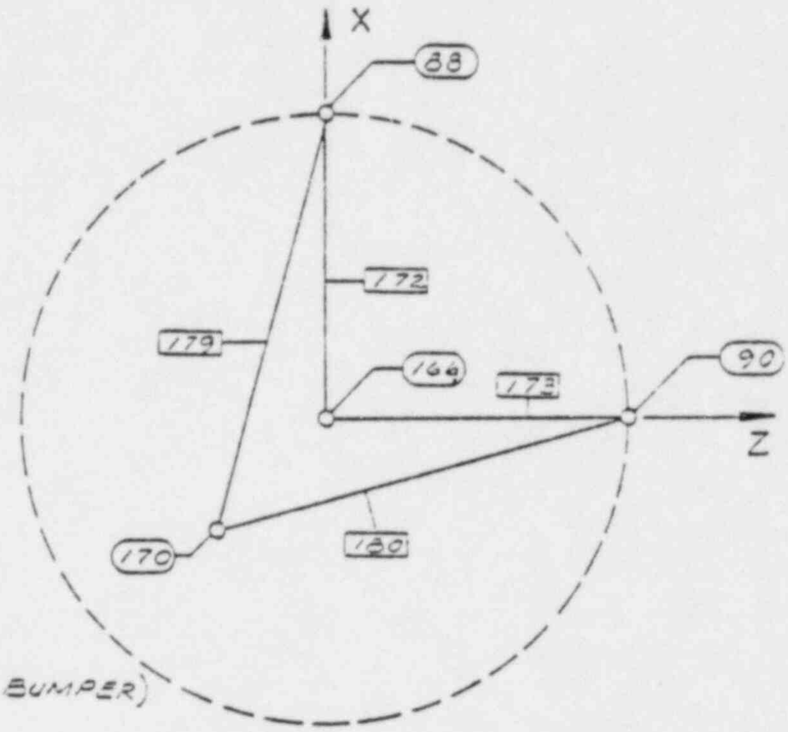


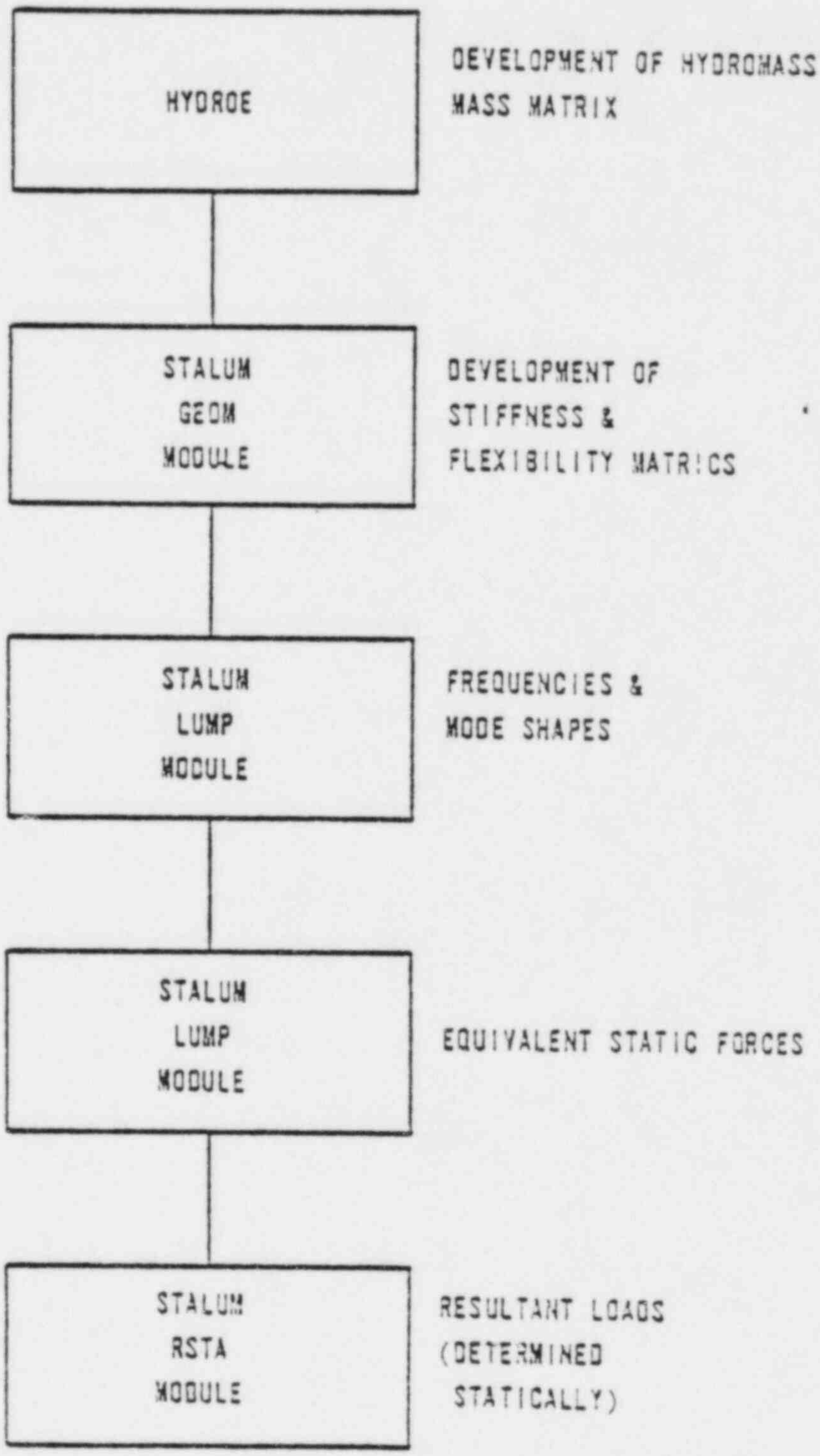
FIGURE 17 INTERNAL WALL STRUCTURES





REACTOR VESSEL BUMPER ELEVATION

FIGURE 18 Utilization of Computer Programs



Enclosure 2
Serial 9787

015263 Bechtel Power Corporation

SUBJECT: MCAR 37 (issued 12/28/79)
Broken Reactor Vessel Anchor Stud in Unit 1

INTERIM REPORT 4

DATE: November 5, 1980

PROJECT: Consumers Power Company
Midland Plant Units 1 and 2
Bechtel Job 7220

Introduction

The discrepancies discussed in this report concern the failed reactor vessel (RV) anchor studs in Unit 1.

Background

The anchor studs in question are 2-1/2 inches in diameter and 7 feet, 4 inches long, embedded in the reinforced concrete RV pedestal. The anchor studs were purchased from Mississippi Valley Structural Steel of St. Louis, Missouri; fabricated by Southern Bolt and Fastener of Shreveport, Louisiana; and heat-treated by J.W. Rex of Lansdale, Pennsylvania. These studs were received on site by Bechtel in early 1976; embedded in concrete by Bechtel in April 1977; and tensioned by Babcock & Wilcox Construction Company in late July 1979. The first stud failure was discovered on September 14, 1979. The second and third stud failures were reported on December 20, 1979, and February 5, 1980, respectively.

Investigative Action

Teledyne Engineering Services' (TES) investigation for Bechtel is complete. The resulting reports discuss the stud failure investigation and the use of the present studs for service. Consumers Power Company and TES are currently investigating the root cause for the excessive hardness of the studs. Bechtel and Consumers Power Company are in the process of retaining a consultant in bolt tensioning to evaluate the tensioning procedure and explain the scatter of lift-off values that occurred during detensioning of the Unit 1 studs.

Bechtel has calculated stresses in the studs and upper lateral support brackets based on conservative preliminary loads provided by B&W for the accident condition of a combined seismic and loss-of-coolant accident event. Bechtel has found those stresses to be within the allowable range.

0152-63

Corrective Action

The prestress levels of the Unit 1 studs have been lowered to 6 ksi. The lift-off values, recorded for these studs during detensioning, are shown in Figure 1. The studs that lifted off at a stress of less than 75 ksi were proof-test tensioned to 75 ksi so a minimum value of 37.5 ksi could be used as an allowable short-term stress.

Reactor Pressure Vessel Support Modification for Midland Nuclear Power Plant, Midland, Michigan, Preliminary Report No. 1, July 1980, was transmitted to Region III by Serial 9330 on July 24, 1980. Report No. 2, which provides the analytical techniques for design, is currently being prepared and will be transmitted by the end of October 1980.

Safety Implications

If uncorrected, this deficiency could adversely affect the safety of operation of the Midland plant at any time throughout the plant's expected life.

Reportability

This condition was reported to the NRC by Consumers Power Company under 10 CFR 50.55(e) on September 14, 1979.

Submitted by: Craig Boyak OT

Approved by: [illegible] FOR L.H.CURTIS

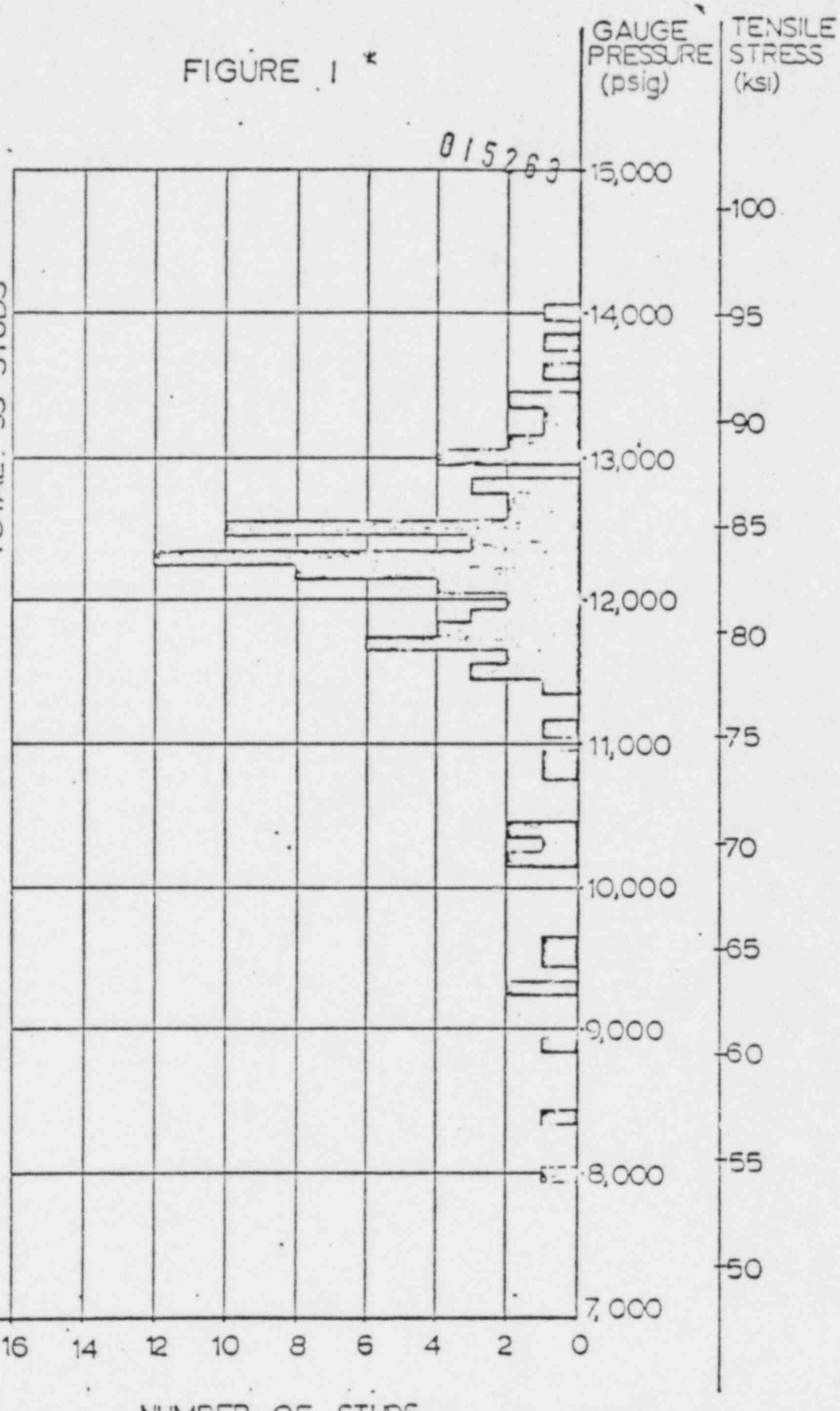
Concurrence by: K.D. Bailey

BD/CB/sg

Attachment: Figure 1

FIGURE 1 *

..DLAND JOB 7220
UNIT-1
LIFT-OFF VALUES DURING
DETENSIONING
APRIL 8 1980 TO JULY 17, 1980
TOTAL: 93 STUDS



* VALUES HAVE NOT BEEN VERIFIED WITH B&W CONSTRUCTION CO.