UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUN 27 1980

NOIXO

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support

FROM: James G. Keppler, Director, Region III

SUBJECT: CONSUMERS POWER COMPANY - RECOMMENDED ORDER

We recommend that an Order be issued to Consumers Power Company requiring licensee action to correct unacceptable anchor bolts at the Midland facility. This is a significant technical problem. Recognizing that a hearing is pending on an Order related to inadequate foundation materials at the site, and recognizing the technical significance of this problem, we believe that an Order requiring corrective action is warranted. A draft letter to the licensee and Order is attached for Headquarters use.

Certain items of noncompliance were identified during the course of the investigation and these are included as an attachment to the draft Order. Considering that these items occurred 4-5 years ago, we see no purpose in requiring a response to the items of noncompliance. The Order requires the necessary licensee corrective action.

We have been in contact with NRR personnel who are evaluating the licensee's proposed corrective action. While they indicate that their review is not complete, they believe the proposed actions will be acceptable upon final review.

Please let us know if you have questions on this matter.

for Glen W. Roy
James G. Keppler
Director

Attachments:
1. Draft Letter to licensee
w/attached Order
2. Draft Investigation Report

cc w/attachments:
H. D. Thornburg, RCI
J. Lieberman, ELD

cc w/attachment 1:
R. DeYoung, IE

B/11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

Docket No. 50-329

Docket No. 50-330

Consumers Power Company

ATTN: Mr. Stephen H. Howell

Vice President

1945 West Parnall Road

Jackson, MI 49201

Gentlemen:

This refers to the investigation conducted by Messrs. J. E. Foster and C. M. Erb of the Region III Office during February 27 - May 2, 1980, regarding the procurement and manufacture of reactor vessel holddown studs utilized for Midland Unit 1. Our findings were discussed during a meeting between J. G. Keppler, Director, Region III and you and members of your respective staffs on May 2, 1980.

Our investigation findings indicate serious deficiencies related to the specification, material selection and heat treatment for these important items, and we are concerned that your system was not sufficient to identify these deficiencies. Based on our concerns relative to bolting materials, we are issuing the attached Order requiring specific corrective actions.

Items of noncompliance identified during this investigation are attached to the Order. We recognize that the reactor vessel holddown studs were manufactured approximately five years ago, and conditions relative to their manufacture cannot be altered at this date. Therefore, no response to the specific items of noncompliance is required.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosure, will be placed in the NRC's Public Document Room.

Your response to the enclosed Order and future inspections will determine if further escalated enforcement action is required.

Sincerely,

Victor Stello, Jr.,
Director
Office of Inspection and
Enforcement

Enclosures:

1. Draft letter to licensee
   with enclosed Order
2. Draft IE Investigation Reports
   No. 50-329/80-13 and No.
   50-330/80-14

cc w/encls:

Ronald Callen, Michigan Public
  Service Commission

Myron M. Cherry, Chicago

THE UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:

Consumers Power Company

Midland Nuclear Power Plant — Docket No. 50-329

Units 1 and 2 — Docket No. 50-330

I

The Consumers Power Company (the "licensee") is the holder of Construction Permits No. CPPR-81 and No. CPPR-82 which authorize the construction of two pressurized water reactors in Midland, Michigan. The construction permets expire on October 1, 1981 and October 1, 1982 for Unit 2 and Unit 1 respectively.

II

In February 1980, the licensee reported that three reactor hold down bolts on Unit 1 had failed. An investigation into this problem, which was concluded on April 18, 1980, shows that the hold down bolts on the Unit 1 reactor vessel are unacceptable per ASME III and ASTM specifications. The bolts were made of improper material and not properly heat treated or tested. Improper engineering judgements including specification of material and quality assurance deficiencies led to the problem. The related violations of NRC regulations are set forth in Attachment 1. Under existing criteria, the bolts are rejectable on Unit 1 and similar bolts on Unit 2 and the steam generators are questionable.

III

Under the Atomic Energy Act of 1954, as amended, and the Commission's regulations, activities authorized by construction permits or portions thereof may be suspended should the Commission find information which would warrant the Commission to refuse to grant a construction permit on an original application. We conclude that the engineering and quality assurance deficiencies which led to the failure of the reactor hold down bolts are an adequate basis to refuse to grant a construction permit, and therefore, suspension of certain activities under Construction Permits No. CPPR-81 and No. CPPR-82 is warranted if these safety related issues cannot be resolved.

IV

In view of the foregoing and pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT THE LICENSEE SHALL:

a) obtain approval of the Office of Nuclear Reactor Regulation of the method of repair of the reactor vessel anchor bolts for Unit 1;

b) provide assurance that anchor bolts for the Unit 2 reactor vessel and the steam generators meet existing criteria, and if they do not meet existing criteria, obtain approval from the Office of Nuclear Reactor Regulation of the method of repair of these anchor bolts; and

c) assure that other safety related bolting and component support materials have been procured according to the proper quality standards and codes and provide a written report within 30 days to the Region III office as to the extent of the materials review.

Until such time as items a), b), and c) above are complete, the licensee shall cease all further safety related construction work regarding the bolts in question or other construction not approved by NRR to provide compensation for the unacceptable bolts.

V

The licen e or any person whose interest is affected by this Order may within twenty (20) days of date of this Order request a hearing with respect to all or any part of this Order. In the event a hearing is requested, issues to be considered will be:

1) whether the facts set forth in Section II of this Order are correct; and

2) whether this Order should be sustained.

Any request for a hearing shall not stay the effective date of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold Denton
Director
Office of Nuclear Reactor
Regulation

Victor Stello, Jr.
Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland
this day of , 1980

Attachment: Notice of
Violation

Attachment 1

NOTICE OF VIOLATION

Consumers Power Company

Docket No. 50-329

Docket No. 50-330

This refers to the investigation conducted by representatives of the Region III office at the Midland site on February 27-29, 1980; with subsequent visits March 5-6 at Southern Bolt Company; March 11-12 at J. W. Rex Company; March 18-19 at Bechtel; March 20 at Mississippi Valley Structural Steel, April 18 at Bechtel, discussed during the May 2, 1980 meeting at the Region III Offices.

It appears that certain of your activities were in noncompliance with NRC requirements as noted below. Each item is an infraction.

1. 10 CFR 50, Appendix B, Criterion IV, requires, in part, that . . . "Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements, which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, . . . whether purchased by the applicant or by its contractors and subcontractors."

Governing procurement specification No. 7220-C-233(Q), Revision 3, states that reactor vessel anchor bolts and nuts will be utilized as ASME (American Society of Mechanical Engineers), Section III, Division 1, Class 1 component supports. Complete requirements for Section III, Class 1 component supports were incorporated in the Winter 1973 amendment to Section III, and were identified as Component Supports, Subsection NF.

The purchase order for reactor vessel anchor bolts was dated September 16, 1974, making the applicable ASME Code Edition Winter of 1973 or Summer, 1974.

Contrary to the above requirement, Subsection NF was not made the requirement for reactor vessel anchor bolts with the following results:

a. ASTM A354 Grade BD was specified as the stud material, which did not have an ASME code allowable stress at the time of order, September 16, 1974.

b. While fracture toughness tests were made, no attention was given to the brittle fracture indicated by lateral deformation tests.

~~c. THERE WAS NO INVOLVEMENT OF THE AUTHORIZED NUCLEAR INSPECTOR IN THIS ORDER OF STUDS.~~

PAGE 3 IS MISSING FROM THIS PACKAGE.

3. 10 CFR 50, Appendix B, Criterion IX, requires, in part, that "Measures shall be established to assure that . . . heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with codes, specifications, criteria, and other special requirements."

Contrary to the above, measures did not assure that heat treating and nondestructive tests were controlled in accordance with applicable codes and specifications. Examples are:

a. The Southern Bolt Quality Assurance manual in Paragraph 2M, Section 10.0, requires that purchase orders state "where the heat treater is to Brinell (hardness test) pieces."

Contrary to this requirement, no location (e.g. surface of bolt) for this test was specified in the heat treatment purchase order.

b. ASTM Code requirements (A-354, A-370) provide for hardness testing of bolting materials. These requirements call for surface hardness tests, with subsurface tests being allowed under specific and limited conditions.

Contrary to these requirements, greater than specified hardness results on the surface of the studs led to performing hardness tests at the mid-radius, on the end of tensile test specimens. Conditions to allow such testing under ASTM Codes were not present, and such tests defeat the purpose of the hardness test as a nondestructive test.

c. The heat treat procedure utilized for treating the reactor studs, J. W. Rex #1, Section 2, states that a "furnace load shall consist of approximately 10 pieces plus test bars."

Contrary to the above, furnace temperature charts submitted for documentation (dated April, 1975) indicate that tempering furnace loads exceeded 10 pieces (38-39 studs were tempered per furnace load). (It is also noted that, in one case, two test pieces did not accompany production bars during heat treatment. Therefore, the test results for this test piece may not represent those for the production pieces).

d. Purchase Order #24844, from Mississippi Valley Structural Steel to Southern Bolt and Fastener Corp., in section 5, indicated that "total material traceability is required."

J. W. Rex Heat Treat Procedure, J. W. Rex #1, Rev. 4, in section 2, required testing and documentation to be on the basis of material heats.

Contrary to the above, material traceability was not maintained in that J. W. Rex was not notified that the studs to be heat treated consisted of two types of steel [illegible]d four material heats until initial heat treating had been acco[illegible]shed.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-329/80-13; 50-330/80-14

Docket No. 50-329; 50-330 License No. CPPR-81; CPPR-82

Licensee: Consumers Power Company
1945 West Parnall Road
Jackson, MI 49201

Facility Name: Midland, Units 1 and 2

Investigation At: Midland Site, Midland
Southern Bolt Company, Shreveport, LA

J. W. Rex Company
Lansdale, PA

Mississippi Valley Structural Steel Co.
St. Louis, MO

Investigation Conducted: February 27-29, March 5-6, 12-13, 20,
April 18, and May 2, 1980

Investigator: ______________________ ________ Date
J. E. Foster

Inspector: ______________________ ________ Date
C. M. Erb

Reviewed by: ______________________ ________ Date
C. E. Norelius
Assistant to the Director

______________________ ________ Date
R. C. Knop, Chief
Projects Section 1

Investigation Summary

Investigation on February 27-29, March 5-6, 12-13, 20, April 18, May 2, 1980
(Report Nos. 50-329/80-13; 50-330/80-14)

Areas Investigated: Special, announced investigation concerning manufacture and installation of reactor pressure vessel holddown studs utilized in Midland Unit 1. The investigation required 150 inspector hours by two NRC personnel.
Results: Of the areas investigated, 3 items of noncompliance were identified: (Infraction - Inadequate Procurement Document Control - Details section, Paragraph 6b; Infraction - Inadequate Control of Special Processes - Details Section, Paragraphs 6d, 6e, 6f; Infraction - Inadequate Control of Purchased Material, Equipment and Services - Details section, Paragraph 6f).

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REASON FOR INVESTIGATION

On September 14, 1979, Consumers Power Company (CPCo) personnel notified NRC Region III, by telephone, of the discovery of a broken reactor vessel holddown stud on the Midland Unit 1 reactor vessel. This condition was subsequently reported under the requirements of 10 CFR 50.55(e) on October 12, 1979, with interim status reports on December 14, 1979 and March 3, 1980. Two other studs were subsequently found to be broken. As this condition reflected a significant deficiency, an NRC investigation was initiated to review the materials, manufacture, and installation of the studs.

SUMMARY OF FACTS

Region III (RIII) inspectors visited the Midland site on February 4-5, 1980, and also attended a meeting at the supplier's facility on February 14, 1980. The results of this inspection and meeting are reported in IE Inspection Report No. 50-329/80-05, 50-330/80-05.

The investigation into the causes of the stud failures was initiated by a site visit during February 27-29, 1980. Subsequently, visits were made to the principal contractor (Mississippi Valley Structural Steel), the supplier (Southern Bolt and Fastener Corporation), the heat treating facility (J.W. Rex Co.), and the Architect-Engineer (Bechtel Power Corporation). During these visits, pertinent files were reviewed, and personnel were interviewed. Materials gathered during these visits were intensively reviewed.

The investigation findings indicate that the root cause of the anchor stud failures was the failure to characterize the studs as American Society of Mechanical Engineers (ASME) Section III, Class 1, Component Supports (Division NF). This failure allowed use of an American Society of Testing and Materials (ASTM) standard specification which would not be allowed under Division NF. Among contributing factors were:

1. The ASTM specification utilized (ASTM A-354) allowed use of American Iron and Steel Institute (AISI) 4140 and 4145 steel in stud manufacture. This material is very difficult to properly heat treat in the diameter required for these studs. Difficulties in through-hardening of the steel in the larger diameters may produce a hard surface and softer center.

2. The heat treater had extreme difficulty treating the material and obtaining acceptable hardness and tensile test results. Finally, hardness tests taken from halfway between the surface and center locations provided acceptable hardness results, but did not indicate the unacceptably hard surface (44-48 Rockwell C). Two reported tests were from test pieces which did not receive the same treatment as the production run of studs.

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3. Charpy impact tests were obtained for the studs, and test results provided indications of questionable properties. However, these impact tests had been performed "for information only" and the results were not reviewed. Previously reported manufacturing problems had not triggered any concern which would cause a review of the Charpy tests.

Several Quality Assurance deficiencies were noted; (1) lack of licensee involvement; (2) failure to advise the heat treater of different heats of material; (3) inadequate document review, (4) failure to respond to indications that the studs were deficient; (5) failure to review materials previously purchased, when the purchase specification was revised; and (6) miscalculation of the stud stress area resulting in a slight over-specification stressing of the studs (this item was licensee identified).

The stud failure mechanism has been identified as stress-assisted corrosion cracking, resulting from properties of the stud material. The licensee is in the process of de-tensioning the Unit 1 studs and evaluating their use.

Tests indicate that some studs utilized in Unit 2, although of different material and heat treatment, have above-specification surface hardness readings. Some steam generator bolts are also questionable and are under review.

An unresolved item was identified during file reviews. A Bechtel memoranda indicated that it had been project practice not to include reference to ASME III in design documents. It is not known if other items were procured without reference to ASME III. An unresolved item is one where more information is needed to determine if noncompliance exists.

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DETAILS

1. Personnel Contacted

Consumers Power

W. Bird, Manager, Midland QA
J. Cook, Vice President, Midland
T. Cooke, Project Superintendent
J. Corley, Section Head, IE&TV
S. Howell, Senior Vice President
H. Hudson, Procurement
D. Keating, QA Group Supervisor
B. Peck, Construction Supervisor
H. Slager, Materials Section, Design
R. Wheeler, PND-Civil Section
J. Wood, Quality Assurance Group Supervisor

Bechtel Power Corporation

J. Barbee, Supervisor, Codes and Standards
W. Barclay, PFQCE
A. Boos, Project Field Engineer
C. Boyak, Project Engineer
R. Brown, Attorney
P. Corcoran, Resident Assistant Project Engineer
L. Davis, Construction
L. Dreisbach, PQA Engineer
M. Elgaaly, Project Engineer
P. Goguen, Field Engineer
H. Hudson, Procurement
J. Russell, QC
J. Rutgers, Project Manager
R. Sevo, QA Engineer
E. Smith, QA
T. Suplee, Project Engineer
D. Yuan, Project Engineer

Mississippi Valley Structural Steel

M. Cohn, Engineer
J. Pantukhoff, Vice President

Southern Bolt and Fastener Corporation

R. Alexander, Vice President
K. Day, QC Administrator
T. Goin, Field Sales Representative
E. Nelson, President
D. Sibley, Quality Assurance
J. Williams, Shipping
J. Wood, Purchasing

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J. W. Rex Company

G. Derstine, Director, Quality Control
K. Krewson, Division Superintendent
F. Vasso, Sales Manager

2. Introduction

The Midland Nuclear Power Plant, Unit 1 and 2, licensed to Consumers Power Company, is under construction on a site approximately one mile south of Midland, Michigan. Bechtel Power Corporation is the Architect-Engineer and Constructor for the plant, designed to utilize a Babcock and Wilcox Nuclear Steam Supply System. Unit 1 is designed to supply process steam to nearby Dow Chemical Corporation in addition to producing electric power.

The reactor pressure vessels for these units are supported by a reactor vessel skirt, which rests on a sole plate in the reactor pedestal. Two rows of reactor holddown studs (48 inner, 48 outer) secure the reactor skirt to the sole plate. These studs are 2 1/2 inches in diameter, 7 feet 4 inches in length, weigh approximately 124 pounds each, and are secured to an embedded anchor plate. By design, the studs were to be pretensioned to 75 KSI (See Exhibit I). These studs are designed to accomodate postulated accident loadings (vessel tip and uplift) and perform no critical function during normal reactor operation.

While the reactor holddown studs are studs by definition (no bolt head is present) the terms stud and bolt have both been used to describe this equipment.

3. Scope

This investigation was conducted to review the history of the reactor pressure vessel studs at the Midland Plant as to their specification, materials, fabrication, heat treatment, testing and installation. The investigation focused on the studs utilized for Unit 1.

The chronology of the NRC investigation is attached as Exhibit II, and a chronclogy of bolt manufacture is attached as Exhibit VI.

4. Technical Background

The hardenability of an alloy is defined as its ability to transform to a fully hardened structure (martensite) throughout a cross section from the austenitizing temperature in the quench medium used. Statements from the bolting section of the 1978 Metals Handbook indicate that (1) "As strength increases and section size increases, hardenability becomes the most important factor in choosing a bolting material," and (2) following an oil quench, the center section of a bolt should be 90% martensite.

The choice of AISI 4140/41~~6~~4 5 steel for studs 2½ inches in diameter by 7 feet 4 inches in length, weighing approximately 124 lbs. each, makes meeting this important metallurgical requirement extremely difficult. Test results indicate that the studs have varying pro-

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perties, indicating that the heat treatment did not produce uniform results. However, due to the properties of the steel itself, it is questionable whether AISI 4140/4145 steel could have been adequately heat treated in this size range without high rejection rates.

AISI 4140-4145 steel is a commonly utilized bolting material, found in many applications. It is recognized by the ASTM Code as an acceptable material in smaller diameter bolting, in a range from 1/2 - 1 3/4 inches. In this size range, the material can be heat treated with relative ease. In larger sizes the material is very difficult to through harden, with the center of the material being several points Rc (Rockwell Hardness) softer than the surface.

As a consequence of the material properties and heat treatment, the surface of the studs is extremely hard, while the mid radius properties barely meet or are below the hardness and mechanical requirements of the stud specification.

Certain anomolous indications raised questions about the stud material. In addition to hardness gradients across the studs, test records indicate some locations along the length of the studs are harder than other locations.

The bar stock utilized for reactor holddown studs did not receive any of the special treatments commonly utilized for critical nuclear grade bolts. Such bolts are typically purchased as vacuum-degassed steel, and purchased oversize. The material is then machined to the needed size, eliminating surface defects which could be a cause for rejection when magnetic testing is done.

The application of the studs is as important as the material in judging suitability. The studs are considerably stressed, and embedded in concrete, conditions conducive to stress assisted corrosion cracking. The threaded areas provide a notch area where this failure mechanism is most likely to occur.

5. Review of FSAR

The Midland Final Safety Analysis Report refers to the reactor vessel anchor bolts specifically in several sections, and by inference in other sections.

Section 3.8.1.6.4, "Containment Liner Plate," in Paragraph 3.8.1.6.4.1, "Materials," notes that the bolts are to be to ASTM 354, grade BD (modified).

Paragraph 3.8.3.1.1 describes the bolts, but does not discuss their design.

Paragraph 3.8.3.4.1 addresses Reactor Coolant Equipment Supports, and on Page 3.8-49, refers to design standards for bolts utilized in Seismic Category I structural supports. This section was added as part of

Revision 17, dated January, 1979, and was in response to NRC questions on FSAR statements. This section appears to commit the licensee to ASME Section III.

NRC question 110.51(3.9.3) resulted in the revision of Section 3.8.3.4.1, noted above. The question dealt with anchor bolts, and support designs. NRC question 110.57(3.9.3) requested further clarifying information after the initial response to question 110.51, and also applies to bolting.

The licensee stated that this response had been mislocated in the FSAR, and was not meant to pertain to reactor vessel support bolting.

Table 3.8-32 appears to apply to the bolts, again describing their material as ASTM A-354, Grade BD.

Figure 3.8-30 is the drawing in the FSAR reflecting stud location and arrangement.

None of the FSAR sections appear to specifically commit to ASME Section III for reactor support holddown bolts.

6. Manufacture of Holddown Studs

a. Material purchase. AISI 4140 and 4145 (low alloy) hot rolled steel rods, 2 1/2 inches in diameter, were utilized for stud manufacture. The steel was purchased from Shill Steel (heat "0000," and not utilized), Armco Steel (heat "00") and Bethlehem Steel (heats "0" and "000") during February 1973 to March 1974. No special requirements were imposed on the material, such as vacuum degassing or machining to reduce surface defects. Chemical analyses supplied by the suppliers showed typical values for these steels. As the rods were purchased well prior to issue of the stud specification or purchase order, Southern Bolt and Fastener (Southern Bolt) did not know how this material would be utilized, and was simply stocking steel rod. Southern Bolt personnel advised that this material was utilized due to unavailability of other grades of steel or larger diameter material.

Discussions indicated that, at this time, Southern Bolt and Fastener was a relatively small firm which manufactured bolts and studs by cutting and threading steel rods and forging heads for bolts. This was their first significant nuclear order.

b. Specification. Requirements for reactor vessel anchor studs were included in Bechtel Specification No. 7220-C-233(Q), "Technical Specifications for Purchase of Miscellaneous Metal for Consumers Power Company."

The specification, in Revision No. 3, dated December 5, 1974, and later revisions, included in Section 5.10 the notation that "These anchor bolts and nuts will be utilized as ASME Section III,

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Division 1, Class 1, Component Supports." File information, (See Exhibit III) indicates that this notation is not an error, and ASME Section III was intended to govern the procurement of reactor vessel anchor bolts.

While component Supports were described in ASME Section III, 1968, a separate Subsection, NF-Component Supports, was added to the 1973 Winter Addenda of the ASME Code, and was required for materials purchased to ASME III specifications six months later. As the purchase order for the reactor vessel holddown bolts was issued on September 16, 1974, the studs should have been characterized as ASME Section III, Class 1, Subsection-NF materials (the reactor pressure vessel code dated 1968 is not applicable to these bolts as they were not a part of the reactor vessel contract). File documents indicate that attempts were made to specify the studs to the equivalent of NF requirements. However, the specification does not meet NF requirements in several significant areas, including ASTM specifications, materials, and testing requirements.

Failure to properly characterize the studs is contrary to 10 CFR 50, Appendix B, Criterion IV, and the Procurement Specification No. 7220-C-233(Q). (50-329/80-13-01, 50-330/80-14-01).

Included in file documentation was a memo (See Exhibit IV) indicating that it was a project practice to refrain from citing ASME Section III in purchase specifications. It is not known if other items were procured without reference to ASME Section III. This is an unresolved item (50-329/80-13-01U, 50-330/80-14-01U).

As originally issued for procurement on May 3, 1974, Bechtel Specification No. 7220-C-233(Q), Revision 2, required anchor studs to ASTM A-490-1971 requirements.

ASTM A-490(1970) "Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints" included a range of 1/2 inch to 4 inch diameter bolts in its scope. This was changed in 1971 to allow a range of only 1/2 - 1 1/2 inch diameter bolts under the specification. The vendor, Mississippi Valley Structural Steel (MVSS) advised Bechtel that ASTM A-490 (1971) did not apply to bolts 2 1/2 inches in diameter, and following discussion, the specification was revised to require ASTM A-354-1966 (Quenched and Tempered Alloy Steel Bolts, Studs, and other Externally Threaded Fasteners). ASTM A-354 is not acceptable under ASME Section III.

When ASTM designations were changed from ASTM A-490 to ASTM A-354, a requirement for Charpy impact test (a measure of ductility) to show a minimum lateral expansion of 25 mils was deleted. The revised specification required Charpy impact test results "for information only." Bechtel personnel advised that this requirement was deleted on the basis of an engineering decision.

ASTM A-354-1966 Grade BD allowed the use of a number of steels, as long as they met the chemical, tensile, and hardness require-

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ments specified. When the vendor proffered AISI 4140-4145 material, Bechtel advised them that it would be acceptable if it met the specification requirements. However, Bechtel file memos indicate a recognition that AISI 4140-4145 material was "marginal" for the application, and suggestions were made by Bechtel personnel to purchase additional bolts because of expected test failures. No action was taken in response to these comments.

As originally issued, the stud specification did not contain testing requirements. Specification Change Notices (SCNs) added these requirements (SCN 4004 dated September 27, 1974, SCN 4005 dated October 11, 1974). Following these changes, the purchase order was modified to include the testing requirements. The specification provided values for minimum yield, and minimum but not maximum, tensile strength, (See Exhibit V, two pages of the Specification).

c. Fabrication. The AISI 4140-4145 rods were cut to size and threaded at each end. This was apparently completed in early December, 1974.

d. Heat treatment. The studs were shipped to the J. W. Rex Company (REX), Lansdale, PA, sometime during December 1974-January 1975. Southern Bolt personnel indicated that REX was selected due to availability and size of rod they could accommodate.

J. W. Rex personnel indicated that they were not initially notified that there were four heats contained in the stud order, and for several months treated the studs indiscriminately as though all material was one heat. This is contrary to 10 CFR 50 Appendix B, Criterion IX, and material traceability requirements contained in Mississippi Valley Structural Steel, Purchase Order 24844 and J. W. Rex Heat Treat Procedure #1. (50-329/80-13-02, 50-330/80-14-02).

REX documents indicate the first full heat treatment (austenitizing and then tempering) was performed during late January 1975. The REX Laboratory Mechanical Property Test Report for this treatment (tests performed on reduced size mechanical specimens), dated January 28, 1975, indicates tensile strength values of 144,500-158,000 PSI, yield strengths of 116,200-130,800 PSI, and Rockwell hardness of Rc 37-42. Twelve of the values reported do not meet requirements, including those pertaining to hardness. These results were reported to Southern Bolt, Mississippi Valley Structural Steel, and Bechtel.

Mississippi Valley inquired if the specification could be changed to ASTM A-354-74, Grade BC, or if hardness requirements could be relaxed. When questioned by Bechtel as to the amount of relaxation on hardness specifications necessary, they requested an allowable Rockwell hardness of Rc 45. Bechtel advised that the test results were unacceptable and hardness requirements could not be relaxed. Southern Bolt was advised of this via telecon on March 21, 1975. This information was passed on to REX.

Several tests were made at the REX facility in attempts to ascertain a heat treatment procedure which would yield acceptable studs. A request to lower the tempering temperature was made, partially as a result of these tests. A letter from Southern Bolt to MVSS, dated April 13, 1975, was used as partial basis for the request to lower tempering temperature. It reflects a resultant hardness of Rc 37 from a tempering run at 850° F. However, the REX file test for this run indicates a hardness value of Rc 41 (all other reported values were correct). As the tempering temperature requested was within the allowable range per ASTM A-354, the change from a tempering temperature of 900°-1000° F to a tempering temperature of 850° F was approved by Bechtel.

Further heat treating was performed at REX, and 21 mechanical property tests were run between April 22-May 16, 1975. These results were given to Southern Bolt and transmitted to MVSS by letter of May 28, 1975. Five of the reported values did not meet minimum yield value requirements. These results apparently were not reported to Bechtel.

Further heat treatments were run at REX, utilizing 850° F as a tempering temperature. Results from tests run on June 27, 1975 and July 2, 1975 (heat "00" at 925° F) were reported for formal documentation. Test reports were to be on pieces accompanying production runs. However, records indicate that two tests run on July 2, 1975 were for test pieces which did not accompany the production pieces, and one hardness value appears to have been reduced from Rc 39 to Rc 38 on the REX file report. This is contrary to 10 CFR 50, Appendix B, Criterion IX and the J. W. Rex Heat Treat Procedure #1. (50-329/80-13-02, 50-330/80-14-02).

Steel from Heat "0000" could not meet specification requirements, and it was apparently scrapped. No information concerning the disposition of this material could be developed.

There are some indications that the heat treatment was improper as to temperature actually induced in the studs during tempering. Furnace heat charts for most furnace runs were from wall thermocouple readings, and for heat "00" the thermocouple placed on the studs was utilized. A comparison of the furnace charts indicates that the studs did not heat as rapidly as the furnace wall, and may not have reached tempering temperatures for the desired length of time.

There are also indications that the presence of a suspending nut as part of the heat treatment fixture may have caused that portion of the stud covered by the nut to heat more slowly than other sections, and hence be tempered to a lesser degree.

Consumers Power personnel have obtained flow rates for the oil bath quench, and have indicated their belief that flow rates are low for a sufficiently rapid quench following stud austenitizing. This would affect the hardening of the studs.

From a review of test records, the dates on the furnace heat charts (date of heat treatment) supplied for the formal documentation package are in error (only month and year were noted on these records). Actual dates were determined from dates on test records and penciled dates on furnace chart margins. In some cases the date is nearly one month in error.

Furnace charts submitted for documentation indicated that 38-39 studs were tempered per furnace load. This is contrary to 10 CFR 50, Appendix B, Criterion IX and J. W. Rex Heat Treat Procedure #1, which required a maximum furnace load of 10 pieces plus test bars. (50-329/80-13-02, 50-330/80-14-02).

REX personnel stated that the heat treatment of the Midland studs was possibly the most frustrating order that they had taken. They noted that the studs were in their facility over six months, when a routine order is processed in approximately two weeks.

e. Testing. Tensile, yield, and hardness testing was performed at J. W. Rex Company following heat treatments. As allowed, tensile and yield tests were performed on reduced specimens. No test pieces were preserved.

REX personnel stated that hardness tests were performed on the stud surface for the initial hardness tests. The tests performed subsequent to June 1975, were subsurface tests done on the tensile specimens themselves at the mid radius of the bolt.

Correspondence indicated that there was discussion of ASTM A-354, Paragraph 4.3, which states "Acceptance on the basis of the tensile requirements shall take precedence where minimum requirements are subject to controversy." It was indicated that a part of ASTM A-370, which gives hardness testing guidance, was also discussed. This part provides for an "arbitration point" in the threaded area of a bolt, and mid radius hardness testing in the thread areas. This portion of the specification is intended for use when the readings are in dispute.

Hardness tests are non-destructive examinations, often done on each piece of critical equipment. Many standards (such as ASTM A-490, ASTM A-540) specify such surface hardness tests be performed. The sections of ASTM discussing subsurface tests, mentioned above, were apparently intended to be utilized in case of controversy over requirements, not in case of unacceptable results from surface hardness tests. Therefore, the subsurface tests do not meet the requirements of the stud specification. This is in noncompliance 10 CFR 50, Appendix B, Criterion IX, and ASTM Code requirements (ASTM A-354, A-370). (50-329/80-13-02, 50-330/80-14-02).

Charpy impact testing was performed on the studs and nuts following heat treatment, by a laboratory at Standard Pressed Steel (SPS). Charpy acceptance criteria of 25 mils lateral expansion had not been removed from the purchase order to Southern Bolt, and the SPS lab noted this requirement on their nut Charpy Impact Test

Report. When reviewed by Bechtel, they were advised to delete this statement from the test form. SPS did not place it on the stud Charpy Impact Test Report. Values reported for lateral expansion on studs range from 1.5-9 mils and would not have met the lateral expansion requirement for the studs had it been imposed.

It was noted that the notarized Charpy Impact Test Report in the site documentation file contained the statement "Charpy test specimens on studs were taken longitudinally, more than one inch below the surface and from the mid ten inches of the seven foot four inch stud. Tests were run after heat treatment." This statement is not contained on the SPS file copy of the report and was apparently added following notarization of the document.

Magnetic particle inspection was performed on the studs by Peabody Testing. On August 8, 1975, the Bechtel shop inspector witnessed this testing, and observed unacceptable linear indications (exceeding one inch in length). It was found that Peabody was using a less strict standard than specified, and all of the tested studs were rejected by the Bechtel shop inspector.

The studs were then returned to Southern Bolt, and actions were taken to remove the indications. The studs were variously hand ground and some 20 were machined to 2.257 inches in diameter. During the period September 30, through October 3, 1975, actions were taken by Southern Bolt to procure alternate bar material (AISI 4340) and to begin stud manufacture again. File memos indicated that this action was apparently begun on the belief that the studs could not be acceptable due to difficulty in meeting magnetic particle test criteria. Due to withdrawal of material suppliers, this course of action was abandoned.

Records indicate that on January 6, 1976, the Bechtel shop inspector witnessed magnetic particle testing at Southern Bolt and approved 97 studs for shipment to Midland. These studs were utilized in Unit 1. Unit 2 bolts were subsequently manufactured of AISI 4340 steel and heat treated at a different facility.

f. Quality assurance review. During this investigation, aspects of quality assurance related to studs were reviewed. File reviews indicated that Consumers Power personnel had no active involvement, beyond approval for financial expenditures, in stud procurement or document review.

No Bechtel shop inspection was performed until after the material had been procured, the studs manufactured, heat treated, and magnetic particle examined. Shop inspection points are at the discretion of the purchaser and inspection prior to final shipment was chosen.

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By memo dated July 17, 1975, (BCBE 604) Bechtel personnel accepted testing on the basis of heat numbers, but required the number of Charpy impact tests to be as specified in Section 5.10.4(c) of Bechtel Specification No. 7220-C-233(Q). This required at least two Charpy tests for heat "0" (approximately 6,325 pounds), and one test for heats "00" and "000." However, only one test was supplied for each heat, and this was not identified during document reviews. This is contrary to 10 CFR 50, Appendix B, Criterion VII and Procurement Specification No. 7220-C-233(Q). (50-329/80-13-03, 50-330/80-14-03).

As the Charpy impact test had been required "for information only" no technically knowledgeable personnel reviewed the test results. Bechtel personnel indicated that tests "for information" are not reviewed unless manufacturing problems are identified.

The following indications of manufacturing problems, did not result in further review:

(1) Questionability of material.
(2) Early failing tests.
(3) Request for relaxation of hardness requirements.
(4) Magnetic particle examination failures.
(5) Length of time to successfully heat treat the material.
(6) Total length of time for stud manufacture.

Review of the Southern Bolt Quality Assurance Manual indicated that it contained requirements for the content of the heat treatment purchase order (Document sent to Heat Treating Company describing treatment). Section 10.0 of Revision 4 (February 27, 1972) in Paragraph 2.M., requires that the purchase orders state "where the heat treater is to Brinell (hardness test) pieces."

Southern Bolt personnel indicated that they could not locate a copy of the heat treatment purchase order for the Unit 1 studs, but provided a copy of the heat treatment purchase order for the Unit 2 studs. The required information on hardness tests location was not provided on this purchase order, and there is no blank provided for recording this information on the standard heat treat form. This is contrary to 10 CFR 50, Appendix B, Criterion IV and the Southern Bolt Quality Assurance Manual. (50-330/80-14-02).

Bechtel Specification No. 7220-233(Q) was revised by Specification Change Notice 6007 on November 8, 1976. This change added Charpy impact acceptance criteria to the section of the specification pertaining to reactor vessel anchor bolts. However, no review of materials procured prior to this change was made to ascertain whether the change affected their status.

Bechtel personnel stated that their review of the specification, done when bolt failures were identified, determined that this revision had been intended for another part of the specification.

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A review of pertinent codes indicates that, in the diameters and strength ranges specified, Charpy impact tests have no acceptance criteria.

7. Installation

There were no indications of shipping deficiencies or receipt inspection problems other than the failure of the document review to note that a sufficient number of Charpy impact tests were not provided, and identify that furnace loads had exceeded those set by the Heat Treatment Procedure.

The studs for Unit 1 were embedded in concrete during April 1977, and tensioned during the period July 23-30, 1979.

The licensee advised RIII that the studs were over-tensioned due to mis- calculation of the effective stress area. The studs were preloaded to an initial stress of 75 KSI in the shank area, but should have been preloaded to this figure in the thread area. The effect of this mis- calculation was to prestress the bolts to approximately 92 KSI versus the specified 75 KSI in the thread area.

A review of Region III records of inspections pertaining to reactor vessel anchor bolts revealed that during an inspection on November 16-19, 1976, a citation was issued to the licensee for failure to protect some of the threads in embedded bolts for Unit 2. There were no other inspec- tion reports relevant to reactor vessel holddown bolts.

8. Identification of Problems

On September 14, 1979, workmen placing jam nuts on the tensioned studs found that a stud (with a nut attached) had failed, and could not be located. This stud was subsequently retrieved from a scrap pile.

Consumers Power advised RIII by telephone of this discovery on September 14, 1979, and followed with a formal letter under the requirements of 10 CFR 50.55(e). Status reports dated October 12, 1979, December 14, 1979, and March 3, 1980 advised of the status of their review. Two additional studs were subsequently found to have failed.

9. Consultant Review

Consumers Power contracted with Teledyne Engineering Services to per- form a failure analysis of the Unit 1 studs, and a review of Unit 2 studs.

Their initial report "Investigation of Preservice Failure of Midland RPV Anchor Studs," (TR-3887-1), dated January 25, 1980, indicates that the studs have a severe hardness gradient, and indicates the failure mechanism as stress corrosion cracking.

10. Management Meeting

A management meeting with representatives of Consumers Power Co., and Bechtel Power Corporation was held at the RIII office on May 2, 1980.

During this meeting, the findings of the investigation were discussed, including matters which were being considered as items of noncompliance (no delineation of noncompliance items was made at that time).

Consumers personnel indicated that they disagreed with the RIII position regarding ASME Section III applicability.

Consumers and Bechtel personnel discussed possible modifications being considered to compensate for the identified stud deficiencies. Any engineering changes formally proposed will be referred to the Office of Nuclear Reactor Regulation for review and acceptance.

The licensee advised that Unit 1 studs were in the process of being detensioned, and detensioning of Unit 2 studs was planned for the near future.

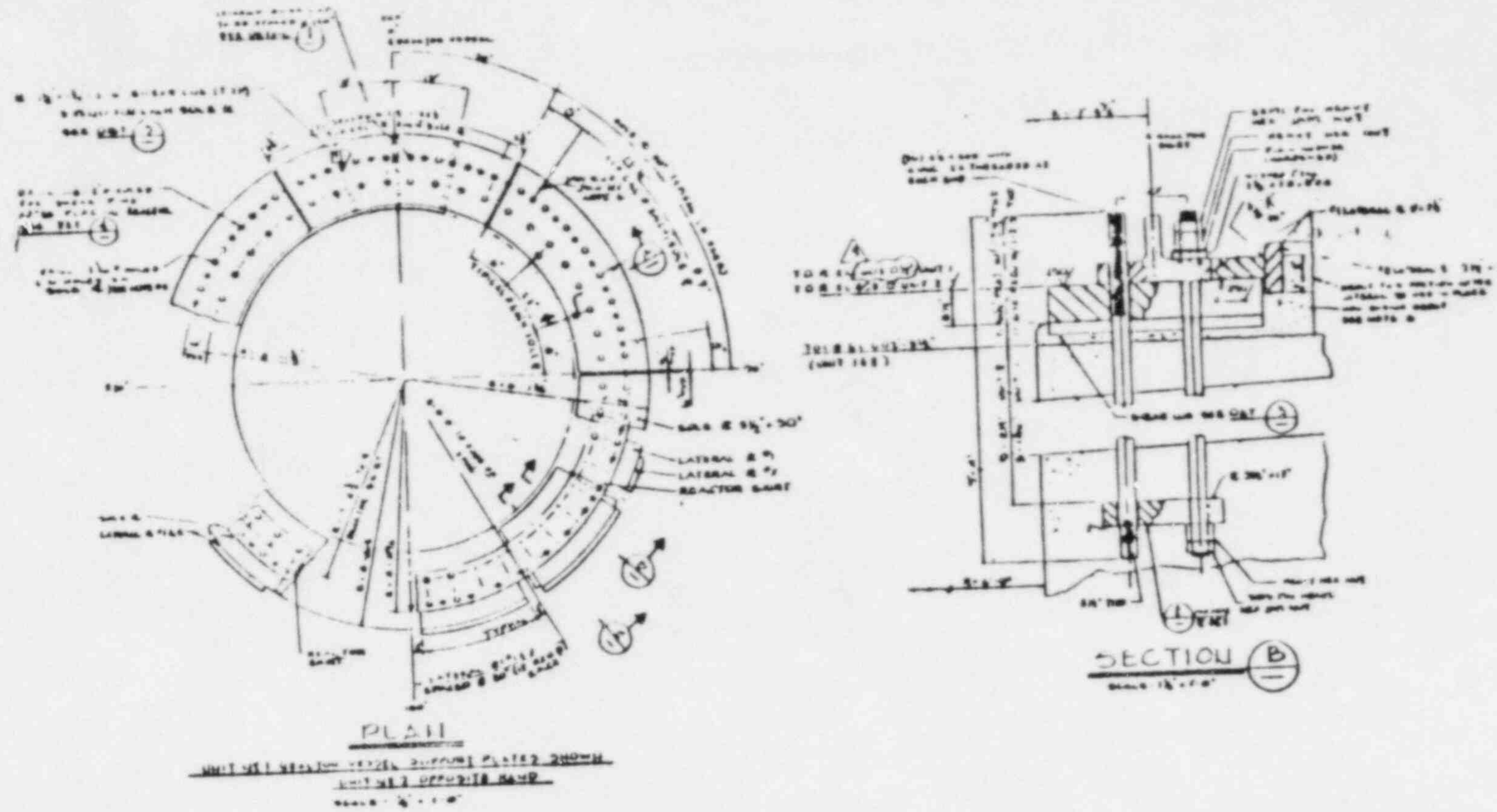
11. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. Unresolved items disclosed during the inspection are discussed in Paragraph 6.B.

Attachments:
Exhibit I. Reactor Vessel Support Diagram (Bolts)
Exhibit II. Investigation Chronology
Exhibit III. File Information Related to ASME III
Exhibit IV. Memorandum on ASME III Usage
Exhibit V. Stud Specification (2 pages)
Exhibit VI. Stud Manufacture Chronology

EXHIBIT I

DRAFT



CONSUMERS POWER COMPANY
MIDLAND PLANT UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Reactor Vessel Support

(C-376, Rev 9)

FSAR Figure 3.8-30

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NRC INVESTIGATION CHRONOLOGY

9/14/79 Licensee reports stud failure.

10/12/79 50.55(e) report from licensee.

12/14/79 interim report on 50.55(e).

2/4-5/80 Inspection of studs on site.

2/5/80 Third stud found broken.

2/14/80 Meeting at Southern Bolt, NRC, CP, SB personnel.

2/27-29/80 NRC Investigation initiated, Midland site.

3/3/80 second interim report from licensee.

3/5-6/80 NRC investigation at Southern Bolt and Fastener.

3/20/80 Inspection Report 80-05 transmitted (2/4-5/80 inspection report).

3/12-13/80 NRC Investigation at J. W. Rex Company.

3/18-19/80 NRC Investigation at Bechtel office, Ann Arbor, Michigan.

3/20/80 NRC Investigation at Mississippi Valley Structural Steel.

4/2/80 Call to Consumers passes on issues for resolution.

4/15/80 Phone call to clarify issues for resolution.

4/18/80 NRC Investigation at Bechtel, Ann Arbor (answers to questions).

5/2/80 Meeting with Consumers Power.

Exhibit II

FILE INFORMATION RELATED TO ASME III APPLICATION TO HOLDDOWN STUDS

(excerpts)

9/23/74, telephone call memo by R. Grote to R. Ryden/L. Koski: "project engineering added to the magnetic particle inspection of the nuts so to be in accordance with ASME Section III - NF," "the subject bolts are classified as ASME Section III Class 1 component supports."

9/27/74, Specification Change Notice (SCN) C-223-4004: "(Note: these anchor bolts will be utilized as ASME Section III Division 1 Class 1 component supports)."

10/1/74, memo BCBI 436, by R. E. Felton to R. L. Castleberry (pg. 2): "Project engineering has affirmed the magnetic particle examination requirement on nuts, the reason being that ASME Section III governs the procurement of reactor anchor bolts."

4/11/75, unsigned notes identified as having been made by Mr. John Hink: "the RVAB (reactor vessel anchor bolts) are classified as component supports in Section NF, Section NF is not mandatory," "design appears to be fairly close to the design requirements of NF."

Exhibit III

DRAFT

Bechtel Memorandum

To: R. L. Castleberry

From: G. Tuveson

Subject: Midland Units 1 & 2 application of ASME B&PV Code Section III Division I Subsection NF Requirements to Component Support Structure

Location: A2-6A

Date: 8/30/76

Job No. 7220
File: C-2135

The above mentioned subject was discussed between M. Rothwell and M. Elgaaly, A. Desai and B. Dhar of civil group on August 19, 1976.

It was agreed that to be consistent with Midland project position, the ASME code would not be directly referred to in the design documents. But the design, fabrication and construction would meet, to the extent possible, the ASME code requirements within the applicable boundaries.

Accordingly, to meet the intent of the code, civil group will add a section to the specifications C-38 and C-233. When required, the design drawings will call out the applicability of this section for a particular structure.

typed copy of handwritten memorandum

Exhibit IV

5.9 Shear Stu shall be in accordance ... The material shall conform to either ASTM A 307 or ASTM the following. The material shall conform to either ASTM A 307 or ASTM A 108 as applicable, and shall meet the tensile requirements contained in AWS D1.1.

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5.10 Reactor Vessel Anchor Bolts and Nuts

(NOTE: These anchor bolts and nuts will be utilized as ASME Section III Division 1 Class 1 component supports.)

5.10.1 Bolts shall be ASTM A 354 Grade BD, with the following additional requirements:

a. ASTM A 614 as specified in Section 5.10.3 below.

b. ASTM A 354 Section 4.4 and Table 3 - Mechanical tests on machined specimens from the Grade BD 2-1/2 inch diameter bolts shall have a minimum yield strength of 130,000 psi, minimum elongation of 14 percent and minimum reduction of area of 35%.

c. ASTM A 354 Table 2 - The Grade BD 2 1/2 inch diameter bolts shall have a minimum tensile strength of 150,000 psi, a minimum proof load of 120,000 psi and a minimum yield strength of 130,000 psi.

d. The bolt material shall be subjected to impact testing as specified in Section 5.10.4 below.

5.10.2 Nuts shall be in accordance with ASTM A 194 Grade 2 or 2H, with the following additional requirements:

a. ASTM A 614 as specified in Section 5.10.3 below.

b. ASTM A 194 Section 5.1 - Certified Material Test Reports shall be in accordance with ASTM A 614 Section 8. If ladle analysis is not available a check analysis may be substituted.

c. ASTM A-194 Section 9 - The Cone stripping test is not required.

d. ASTM A 194 Section 14.1 applies.

e. ASTM A 194 Section 14.3 - Certification shall be in accordance with ASTM A 614 Section 8.

f. The nut material shall be subjected to impact testing as specified by Section 5.10.4 below.

5.10.3 The materials, testing and documentation of the subject nuts and bolts shall be in accordance with ASTM A 614 with the following additional requirements:

a. ASTM A 614 Section 9.1.2 - The written procedure shall be submitted to the Buyer.

b. ASTM A 614 Section 9.1.4 - The written procedure shall be submitted to the Buyer.

c. ASTM A 614 Section 10 is required.

d. ASTM A 614 Section 11 is required.

e. ASTM A 614 Section 12 is required.

f. ASTM A 614 Section 13 is a Seller's option to ASTM A 614 Section 12.

5.10.4 The Charpy V-notch test (C_V) shall be required for the bolts and nuts in accordance with the following:

a. Testing Procedure - Test procedures shall be in accordance with ASTM A 370-72a.

b. Location and Orientation of Test Specimens - The C_V impact test specimens shall be prepared with the longitudinal axis of the specimen located at least 1/2 radius or 1 inch below the surface plus the machining allowance per side, whichever is the lesser. The fracture plane of the specimen shall be at least 1 diameter or thickness from the heat treated end.

c. Sampling Frequency - One test shall be made for each lot of material where a lot is defined as one heat of material heat treated in one charge or as one continuous operation, not to exceed 3,000 lbs by weight.

d. Condition of Material - The test specimens shall be taken after heat treatment.

e. Test Temperature - The impact specimens shall be tested at 40°F.

f. Certified Material Test Report - The test temperature, lateral expansion, absorbed energy and percent shear fracture as well as the orientation and location of all tests shall be reported for information in accordance with ASTM A 614 Section 8.

5.10.5 Handling, shipping and storage shall be in a manner that shall avoid damage to the material. The Seller shall submit written procedures for handling and shipping for approval by the Buyer prior to shipment.

STUD MANUFACTURE CHRONOLOGY DRAFT

11/6/73	Specification 7220-C-233(Q) issued for client review.
3/25/74	steel heats "0" and "000" received, "00" and "0000" received previously.
6/28/74	bid requests sent.
8/5/74	Decisions made as to applicable nondestructive testing requirements.
8/8/74	TWX adds nondestructive testing requirements, original supplier withdraws.
8/21/74	TWX, MVSS-Bechtel, ASTM A-490 is not right specification.
8/23/74	TWX, MVSS-Bechtel, need specification, trying to find material.
8/23/74	Bechtel response TWX, A-490 is correct, 4140/5 steel not approved.
8/27/74	Memo, test requirements, tensile values, 25 mils expansion for Charpy test.
9/3/74	TWX, MVSS-Bechtel, proposal, ASTM-354, 4140 steel, 25 mils expansion.
9/10/74	Memo BEBC 527, approves use of ASTM A-354 as specification.
9/16/74	Contract date.
10/1/74	Memo, history of studs to date.
12/20/74	Rex heat treatment procedure #1, revision 0.
12/74-1/75	Studs shipped from SB to Rex.
1/28/75	Rex material test report, specimens #1, 2, 6, 8, 9, 11, 12.
2/4/75	letter, SB-MVSS, material cannot meet requirements.
2/6/75	TWX, MVSS-Bechtel, provides Rex test results.
2/12/75	Phone call memo, SB requests relaxation of hardness to Rc 45.
2/18/75	Memo, discusses six tests, hardness relaxation request.
3/21/75	TWX, test results unacceptable, not relax hardness requirements.
4/3/75	letter, SB-MVSS, justifying 850 degree temper (reported hardness is wrong).
4/11/75	Notes, 4140 marginal, excessive hardness, where was hardness tested?
4/18/75	Memo, hardness, tempering, material is marginal.
4/22/75	Rex material test, test 1-4 of 19 finally made.
4/25/75	Rex material test, tests 5-11.
5/1/75	Bechtel approval of Rex heat treatment procedure, revision #3.
5/5/75	SB Quality Control manager visits Rex.
5/16/75	Rex material test, tests 11-19.
5/28/75	19 test reports sent with "dummy" documentation package for review.
6/3/75	Rex test, "machined from 2' of end of bar."
6/9/75	Rex test, "machined from 7' from end of bar."
6/16/75	Rex test, stud #1 from heat "000".
6/18/75	Rex test, stud #8 from heat "000", 850 degree temper.
7/2/75	Rex test, "machined from center of bar," heat "00".
7/15/75	Midland meeting, Bechtel and MVSS determine allowable number of tests.
7/17/75	Memo, BCBE 604, physical and mechanical tests to be by heat number.
7/21/75	date on thermocouple furnace chart for heat "00".
7/24/75	Revision #4 of Rex heat treatment procedure approved.
7/29/75	date of material properties report supplied for documentation.
8/18/75	Rex surveillance report, all studs rejected for linear indications.
8/20/75	SB Quality Control manager visits Rex.
8/22/75	TWX on reducing diameter of shank of stud by .060 inches.
9/30/75	TWX, history of studs, start again, new material suppliers withdraw.
11/11/75	TWX, SB proposes turning some studs to 2.257 inches in diameter.
11/20/75	TWX approves turning to 2.257 inches in diameter.
1/4/76	97 studs pass examination, are released for shipment, 96 shipped.
1/22/76	studs received at Midland site.
11/8/76	SCN 6007 adds 25 mil expansion criteria to stud section, possibly in error.
4/77	Unit I studs embedded in concrete at Midland.
7/23-30/79	Unit I studs tensioned.
9/14/79	first stud found to have failed.
1/25/79	Teledyne Engineering report on stud failure mechanisms.

Exhibit VI

Consumers Power Company

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James W Cook
Vice President - Projects, Engineering and Construction

General Offices: 1945 West Parnall Road, Jackson, MI 49201 • (517) 788 0453

December 10, 1980

Mr J G Keppler, Regional Director
Office of Inspection and Enforcement
US Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

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 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

oc1280-0081a112

GALinenberger, ASLB Panel
FPCowan
AS&L Appeal Panel
MMCherry, Esq
MSinclair
CRStephens, USNRC
WDPaton, Esq, USNRC
FJKelly, Esq, Attorney General
SHFreeman, Esq, Asst Attorney General
GTTaylor, Esq, Asst Attorney General
WHMarshall
GJMerritt, Esq, TNK&J

oc1280-0081a112

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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REACTOR PRESSURE VESSEL

SUPPORT MODIFICATION

FOR

MIDLAND NUCLEAR POWER PLANT

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rp1280-0036a112

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 NSSS SUPPORTS

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 LOCA FORCING FUNCTIONS

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.

Core Bounce

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 tne 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

ž	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

Sequence	Stud Number (2) B&W	Stud Number (2) Teledyne	Date	Hydraulic Pressure (psig) 1	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)

Sequence	Stud Number (2) B&W	Stud Number (2) Teledyne	Date	Hydraulic (psig) 1	Pressure 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)

	Stud Number (2)			Hydraulic	
Sequence	B&W	Teledyne	Date	(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

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TABLE 1 (Continued)

Sequence	Stud Number (2) B&W	Stud Number (2) Teledyne	Date	Hydraulic (psig) 1	Pressure 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)

	Stud Number (2)			Hydraulic	Pressure Bolt Stress
Sequence	B&W	Teledyne	Date	(psig) 1	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		BROKEN	
93	45 in	48 in	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.

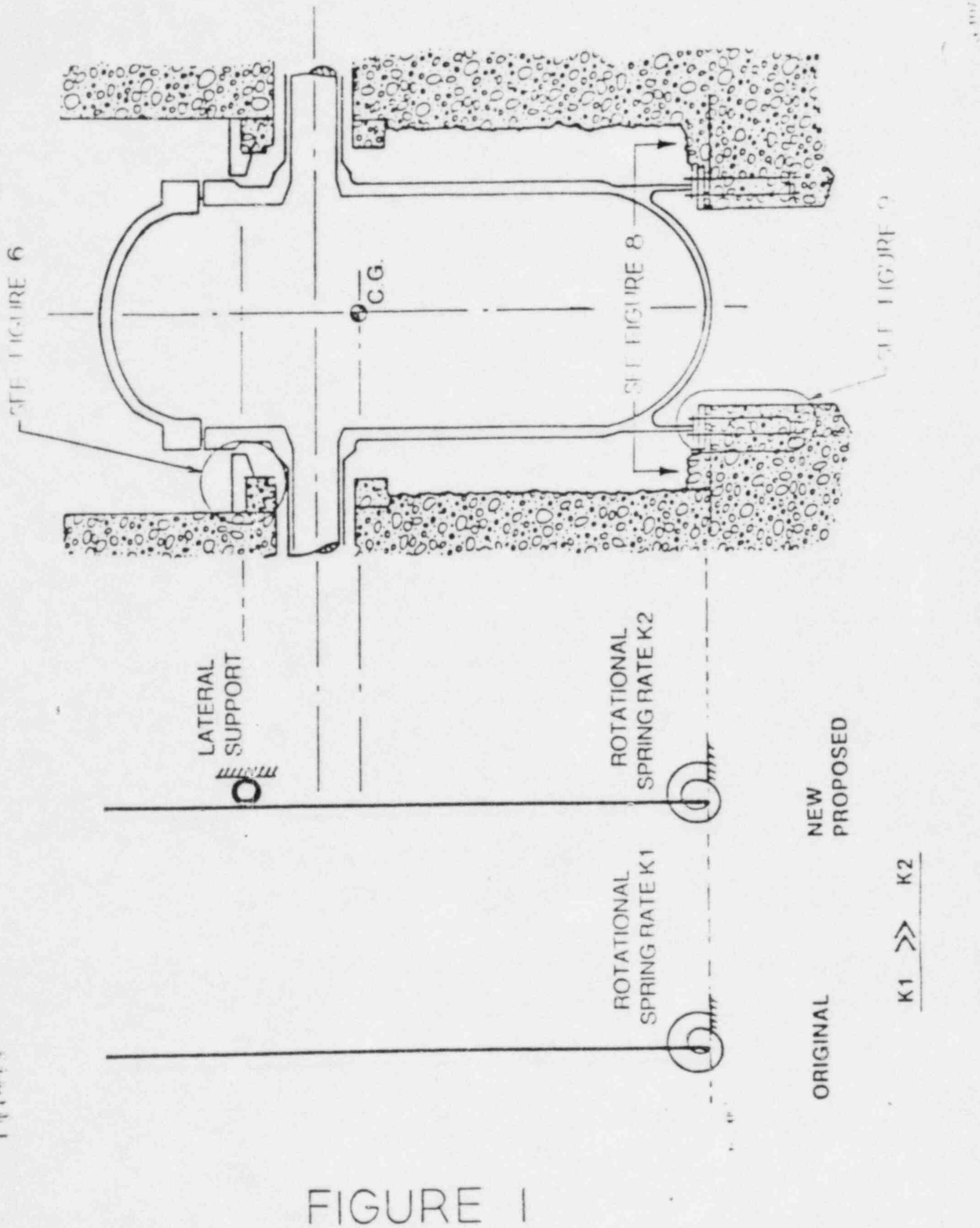
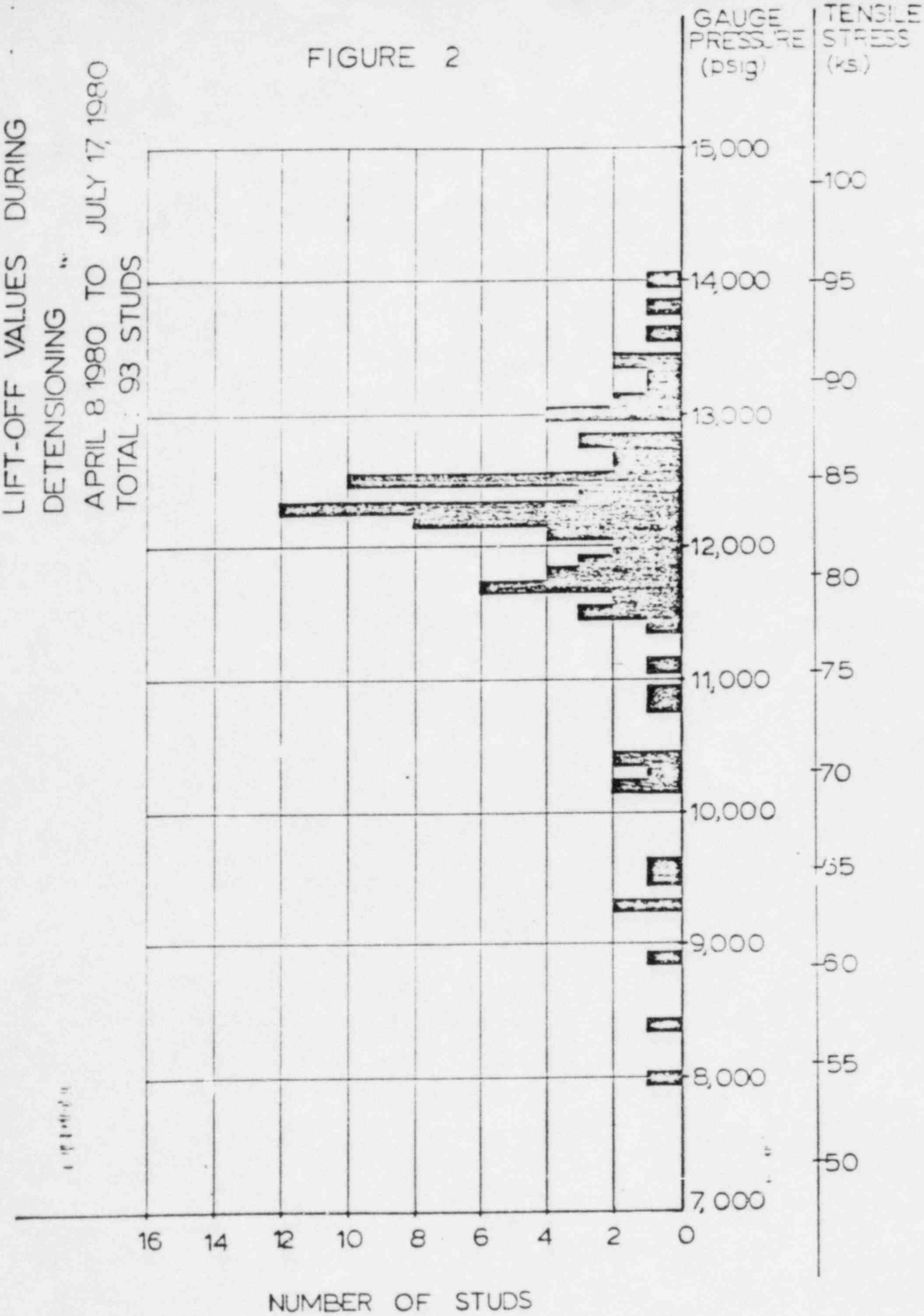


FIGURE I

UPPER LATERAL SUPPORT CONCEPT

FIGURE 2

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



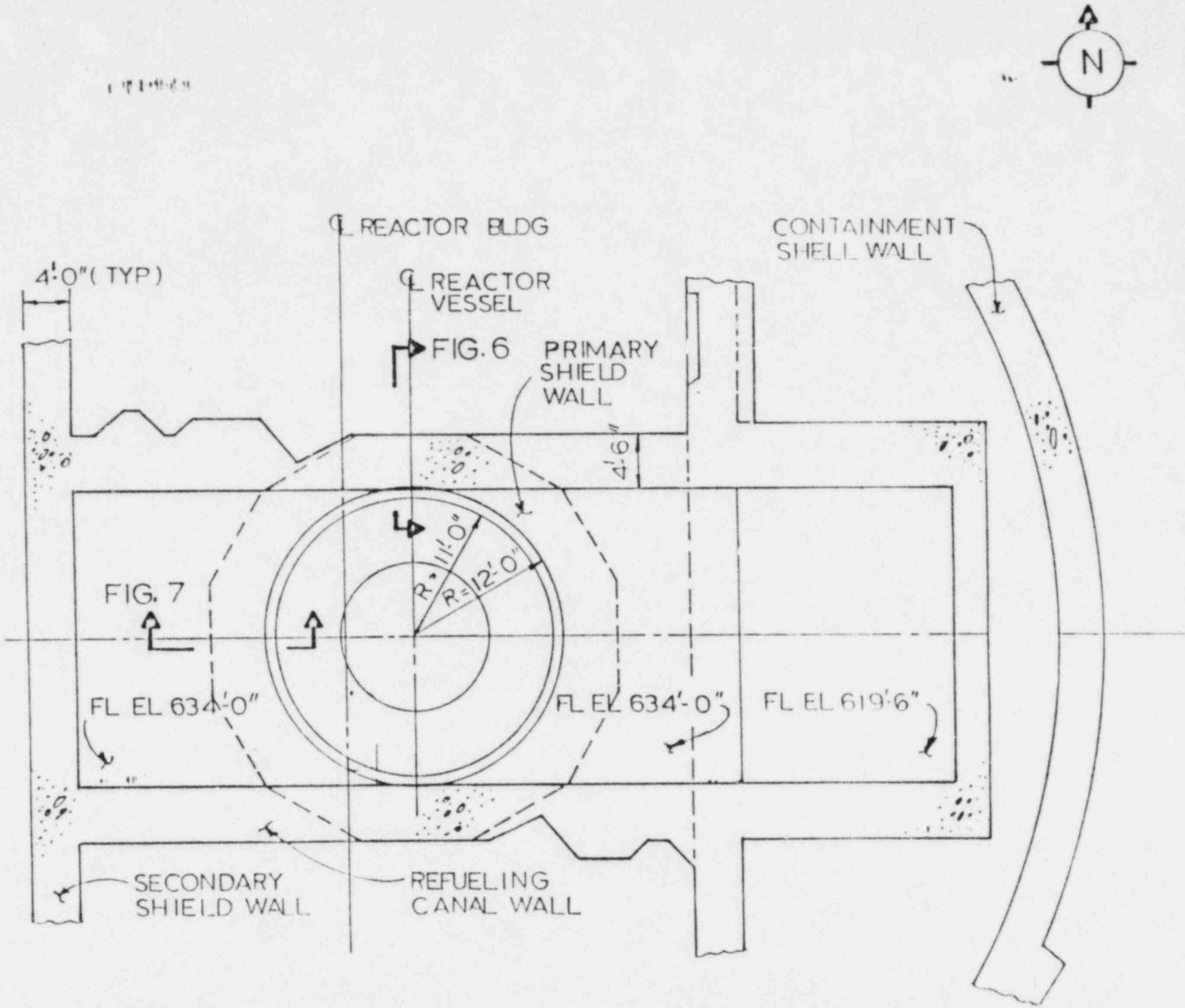
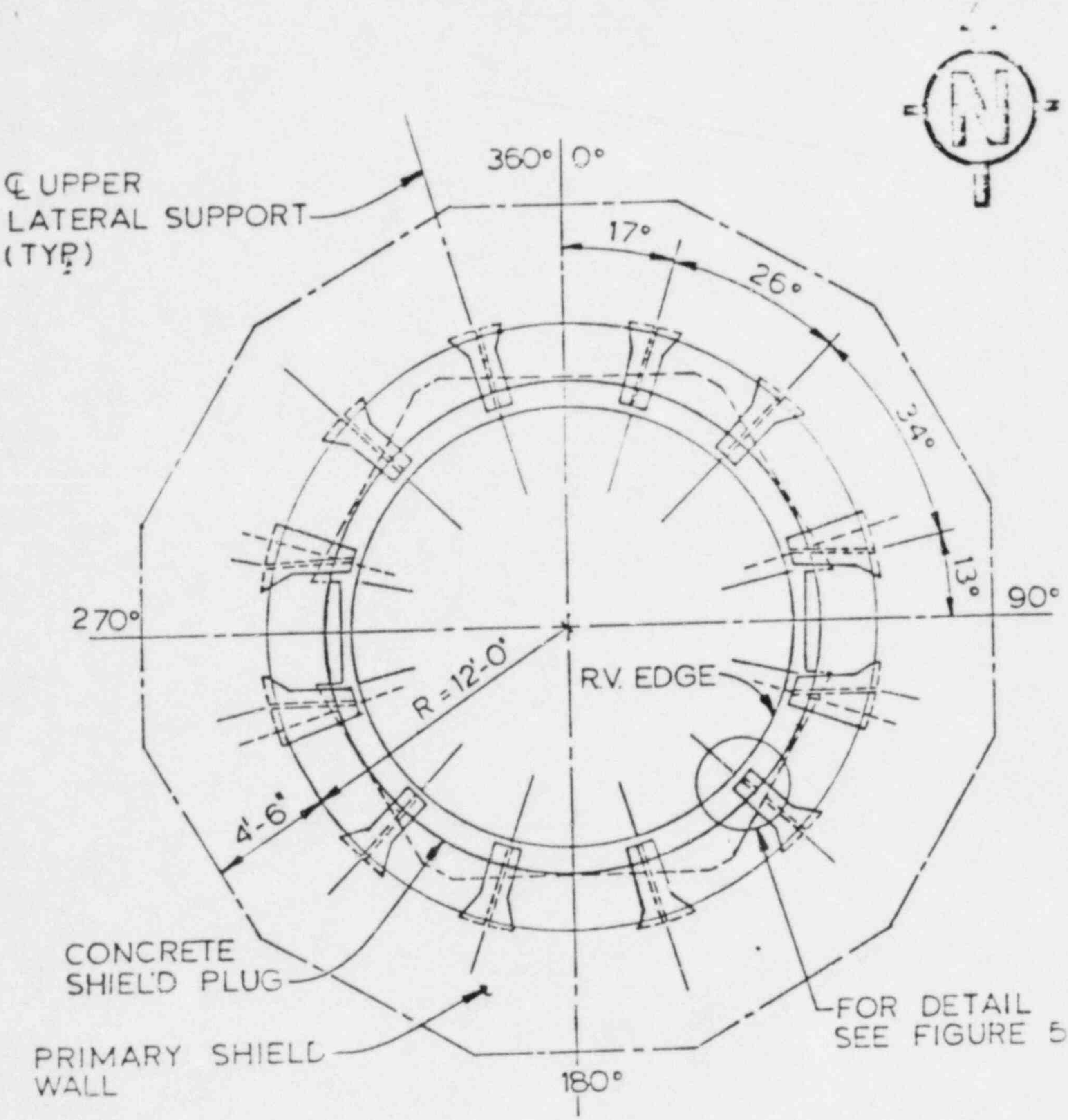
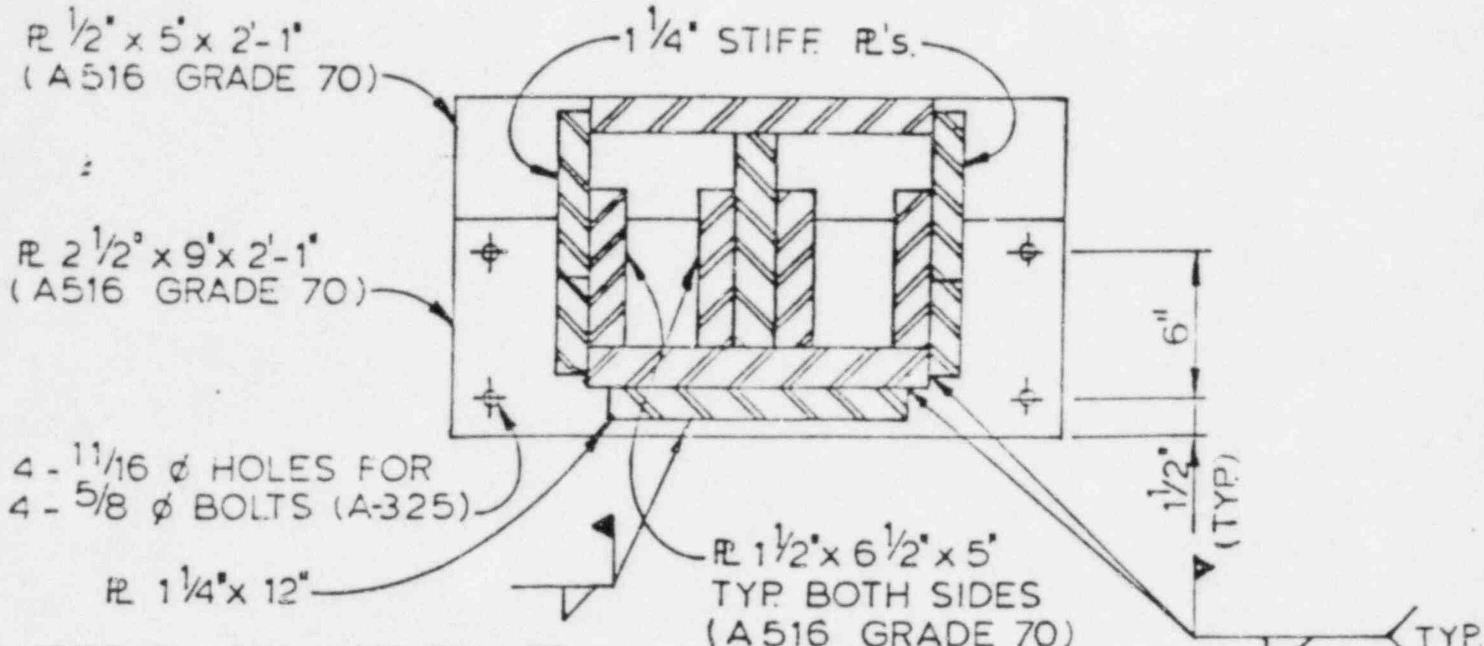


FIG. 3. PLAN AT ELEV. 636'-0"
WALLS AND SLAB



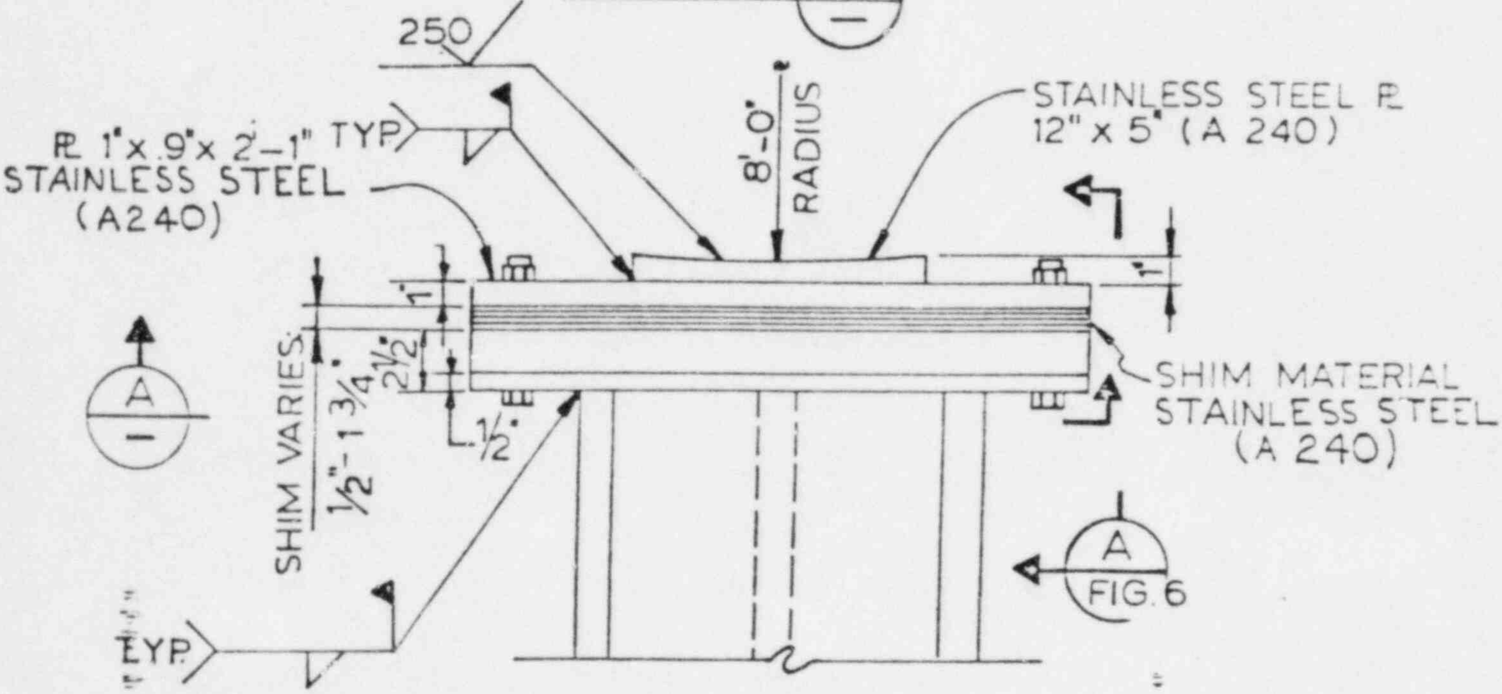
UPPER LATERAL SUPPORT PLAN
FIGURE 4

REACTOR PRESSURE VESSEL
UPPER LATERAL SUPPORT
FIGURE 5

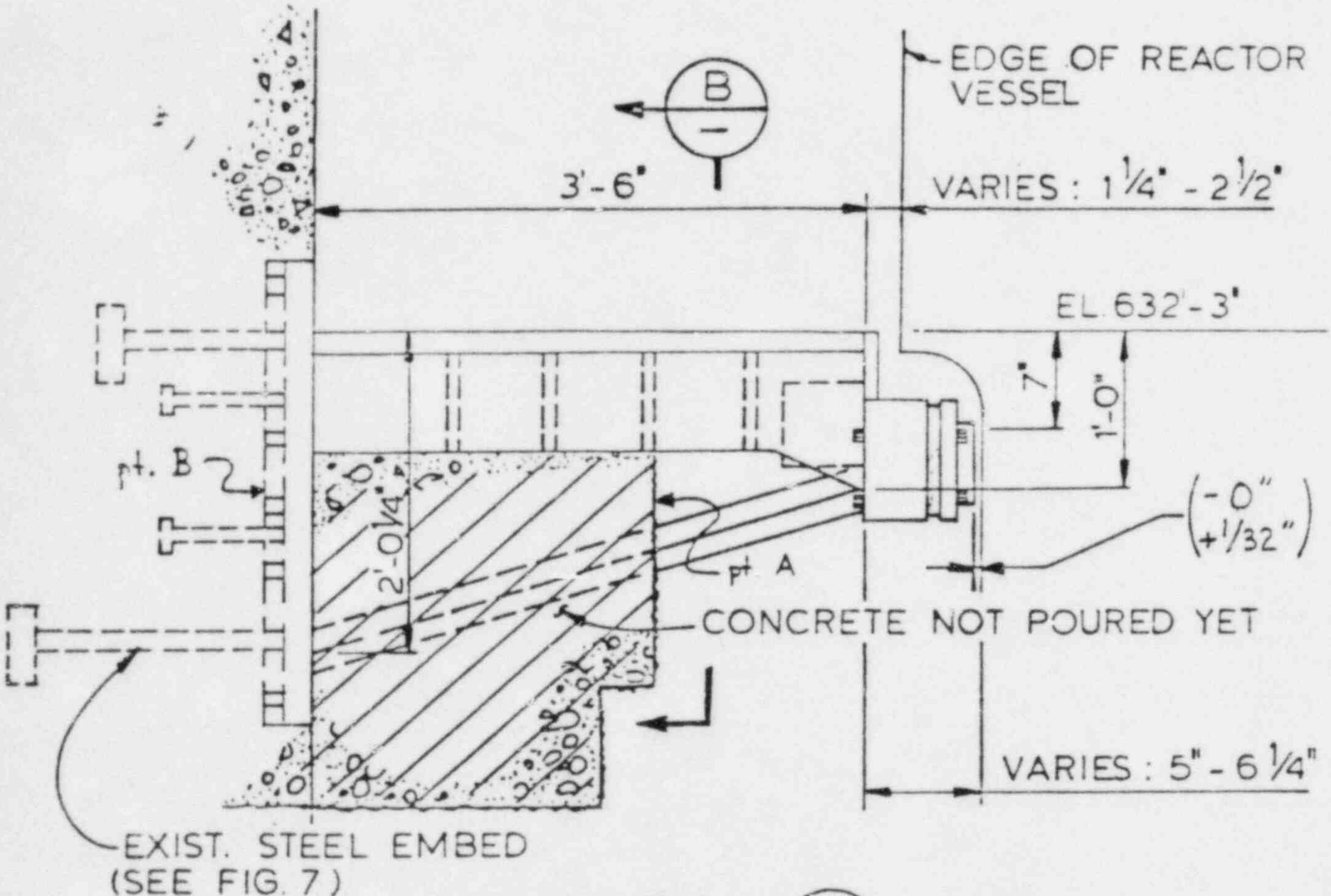


* NOTE: PLATES NOT CALLED OUT ARE EXIST. STEEL

SECTION A

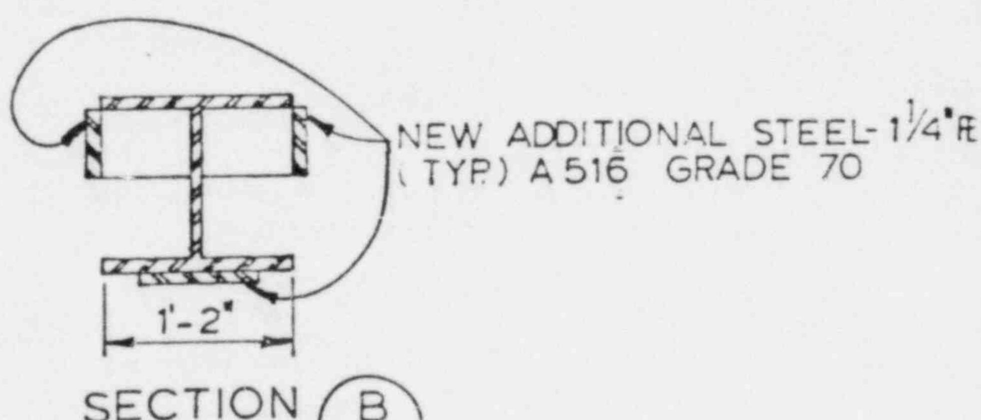


PLAN AT EL. 632'-3"

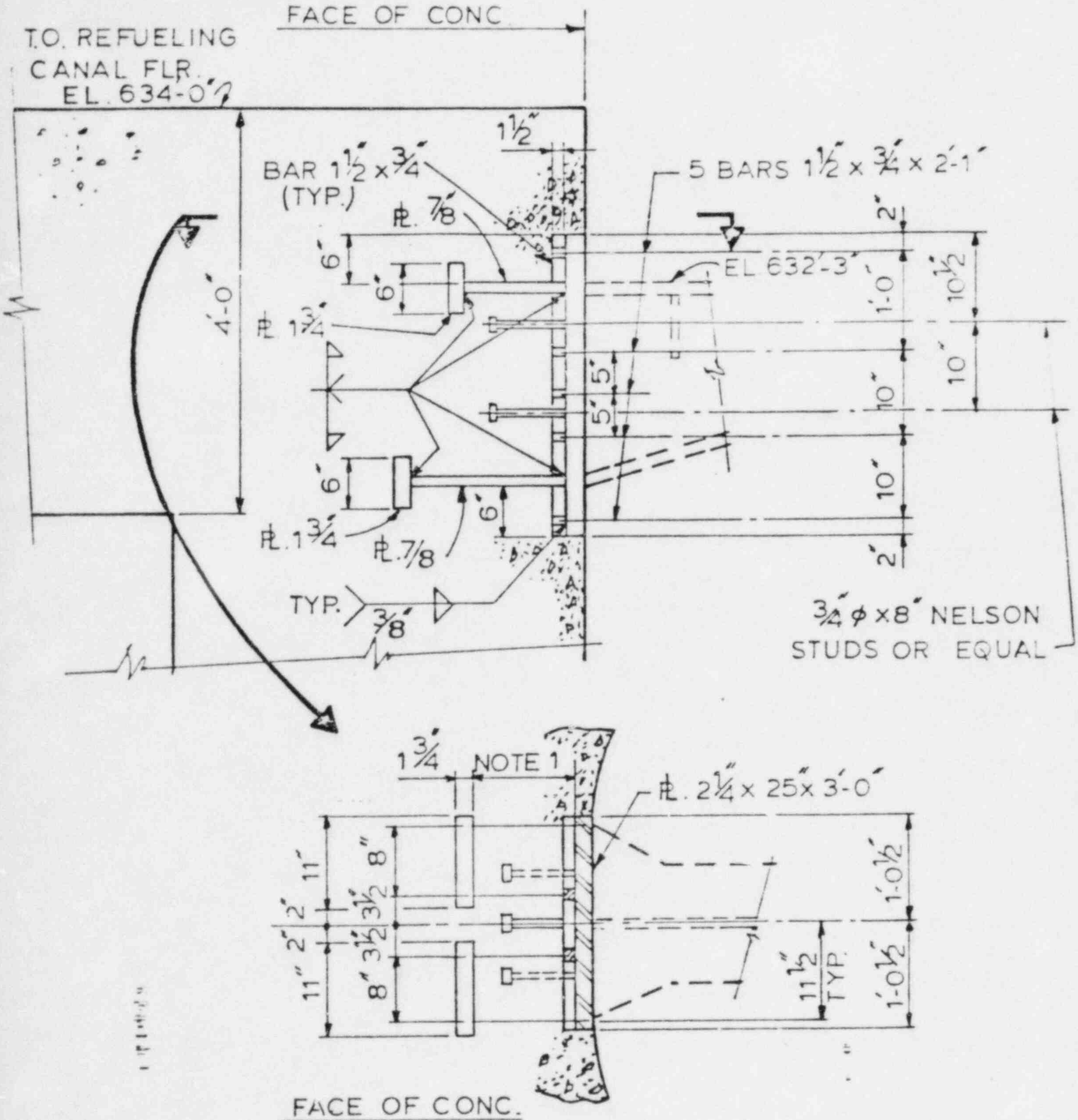


SECTION A FIG. 6

UPPER LATERAL SUPPORT DETAIL
FIGURE 6

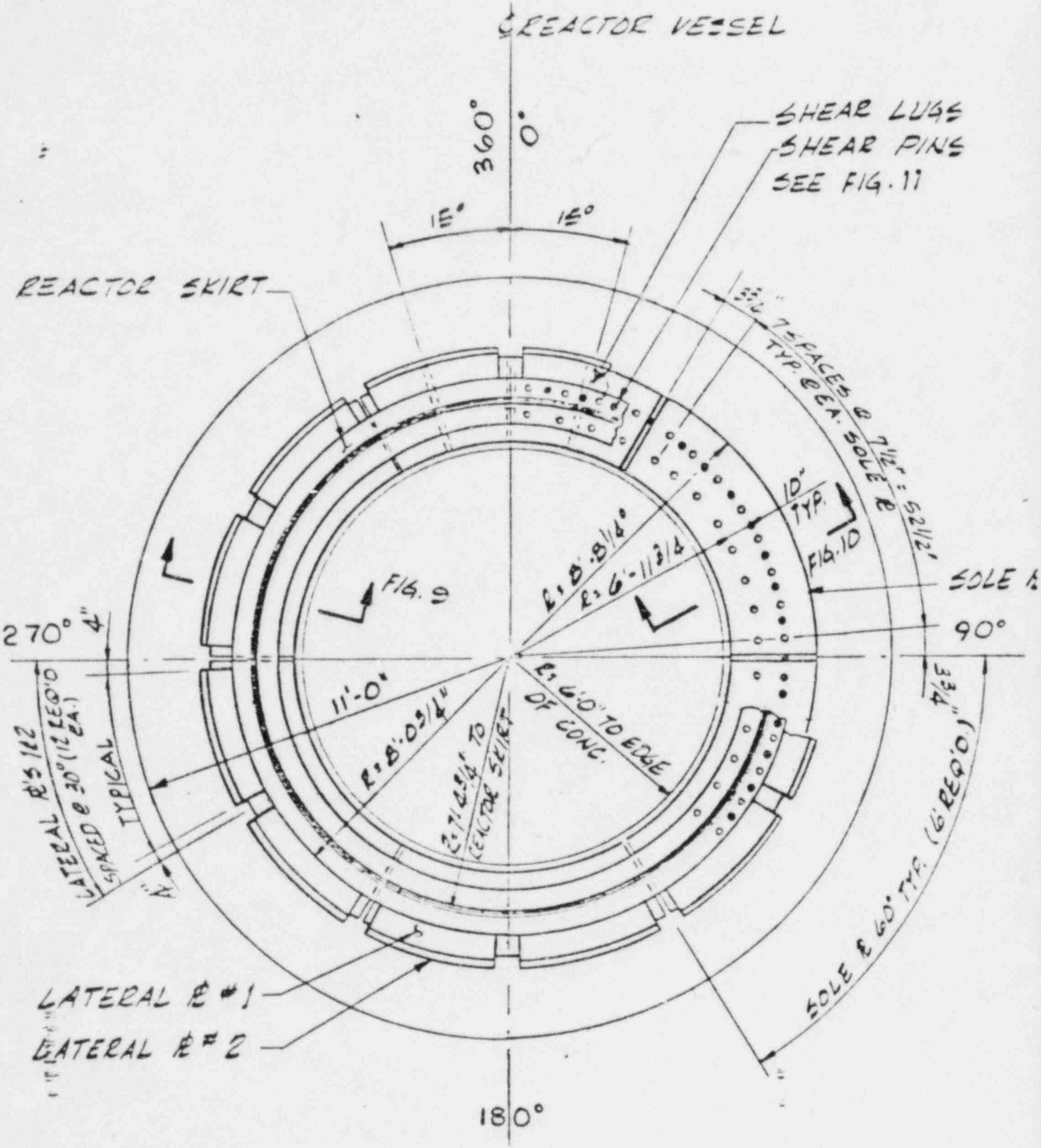


SECTION B



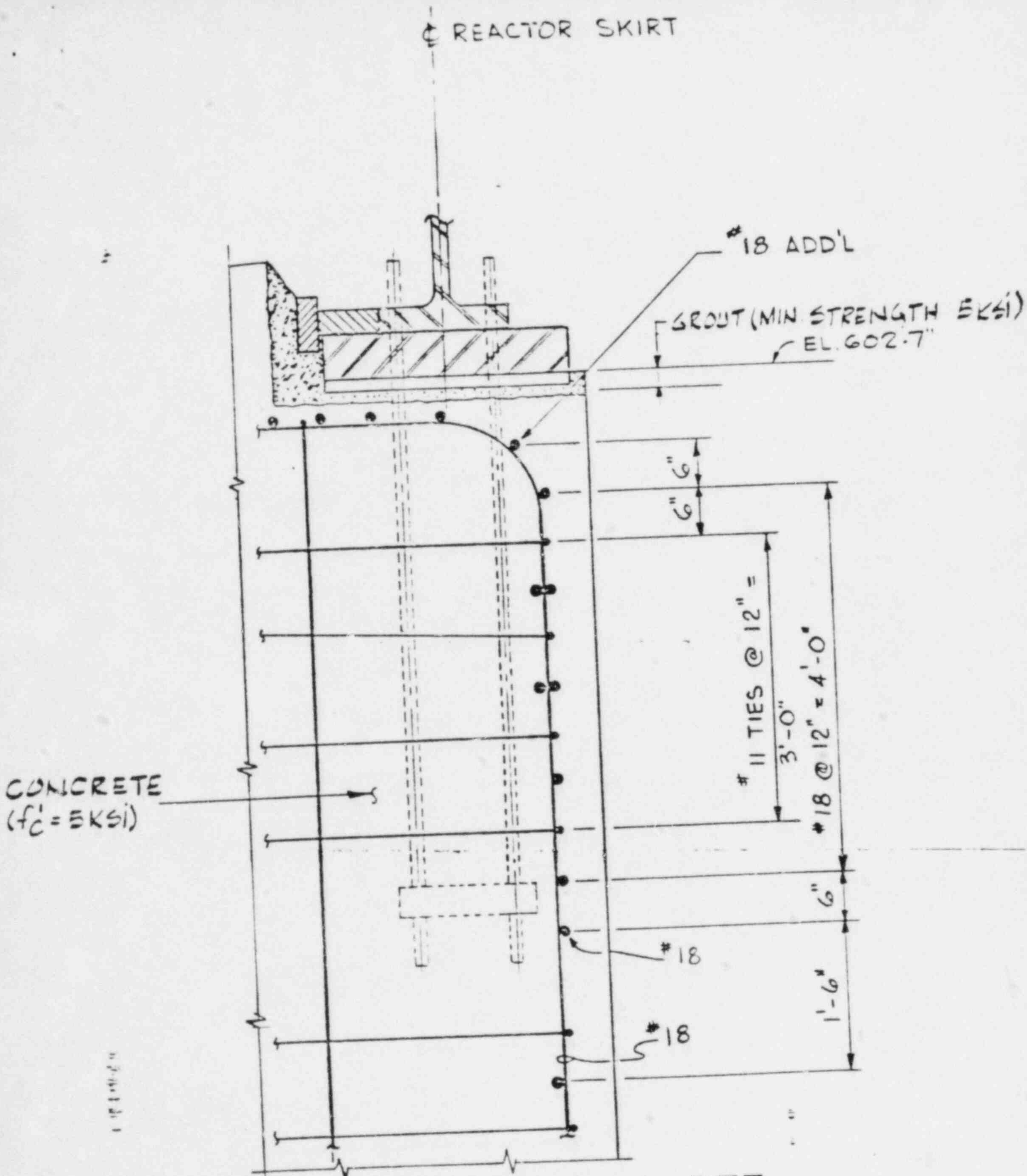
NOTE 1
THIS DIMENSION
1'-0" FOR TOP ANCHOF
1'-6" FOR BOT. ANCHOF

FIGURE 7



PLAN AT EL 603'-1"

FIGURE 8



CONCRETE PEDESTAL DET.

FIGURE 9

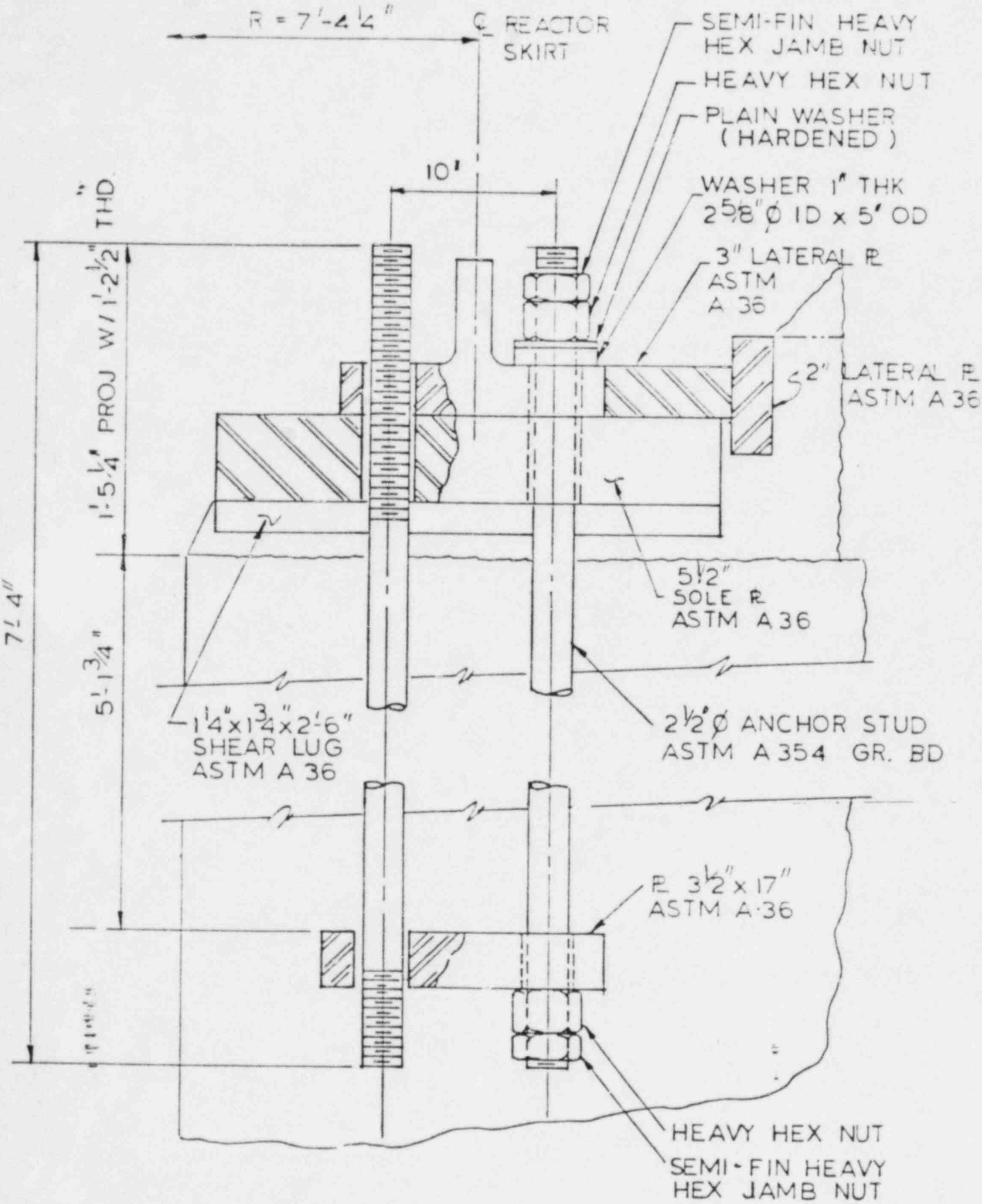
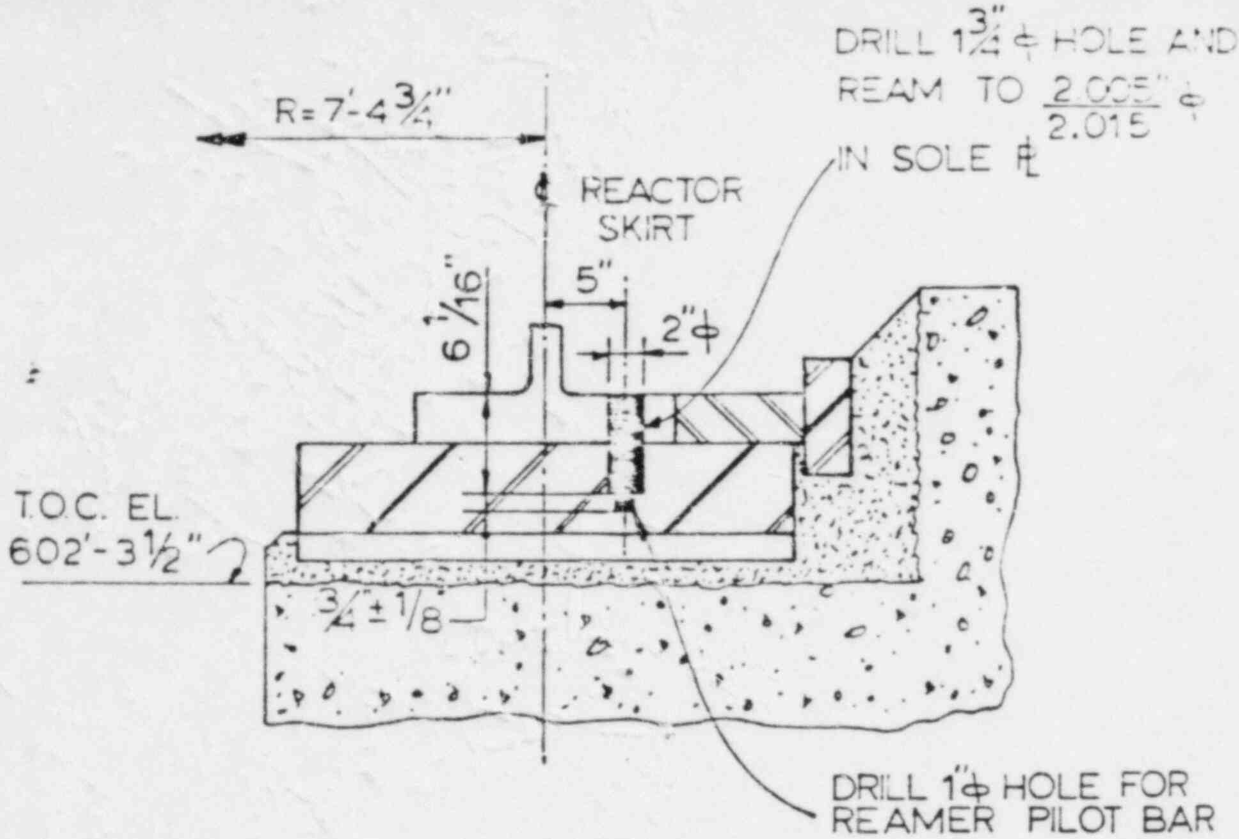
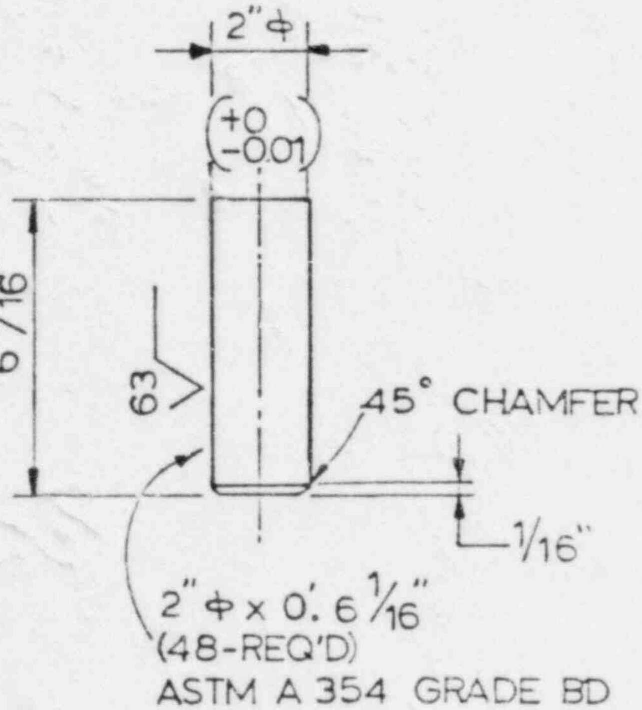


FIG. 10 ANCHOR STUD DETAIL

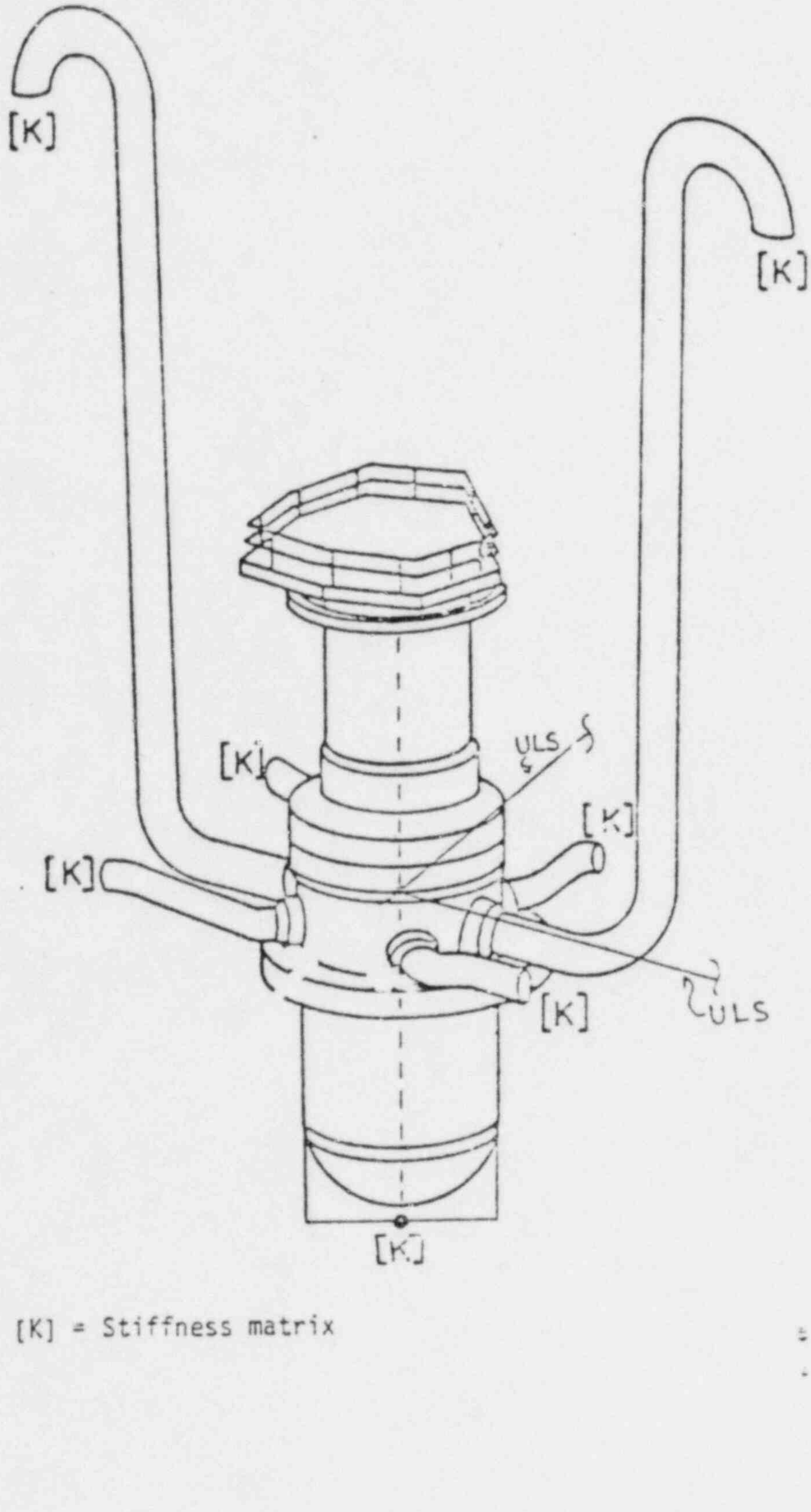




SHEAR PIN DETAIL

FIGURE 11

FIGURE 12 Reactor Coolant System Boundaries



[K] = Stiffness matrix

FIGURE 13 RV Isolated Model, Reactor Internals and SSS

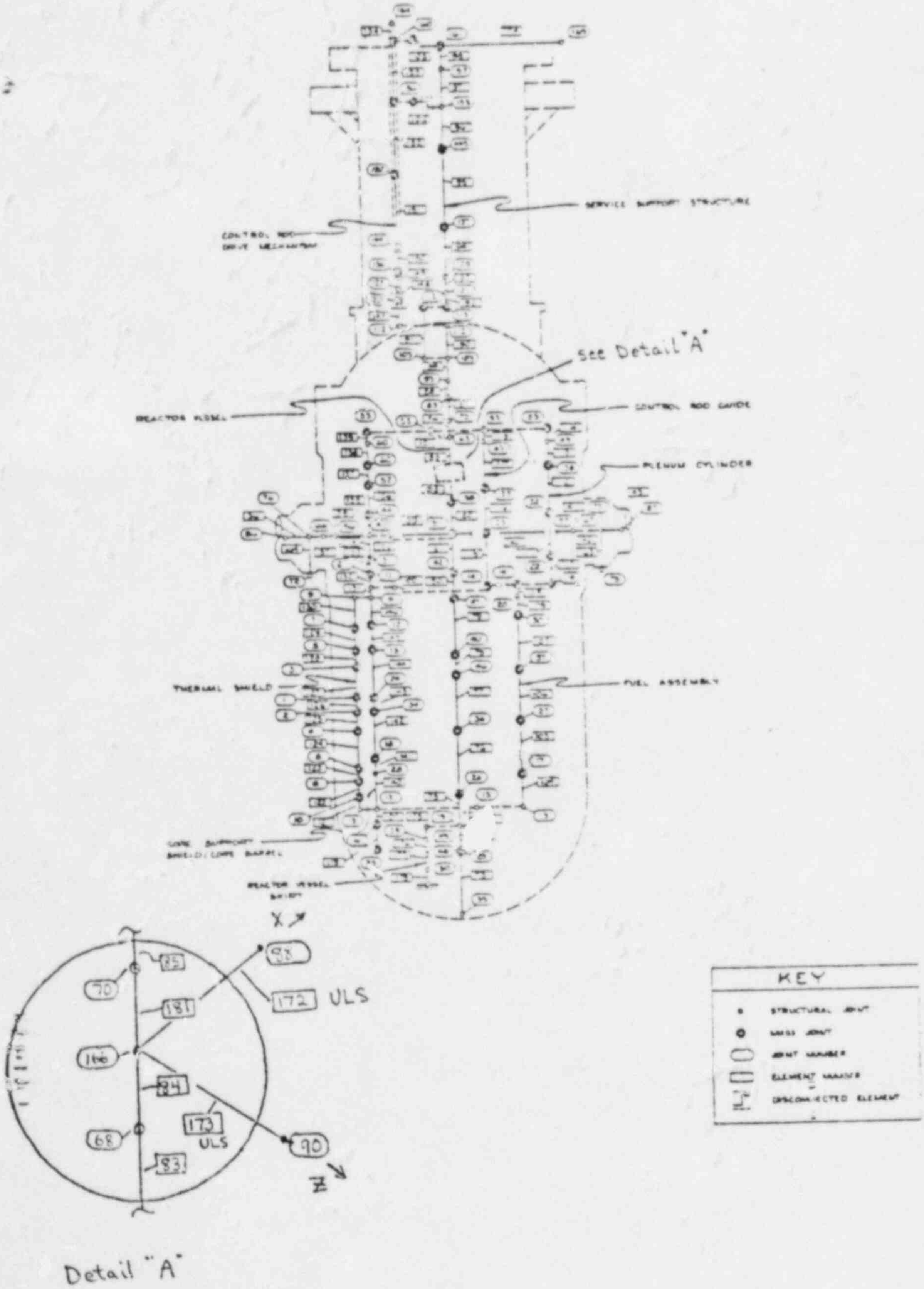


FIGURE 14 RV Isolated Model, Skirt-Supported Plant, Plan View

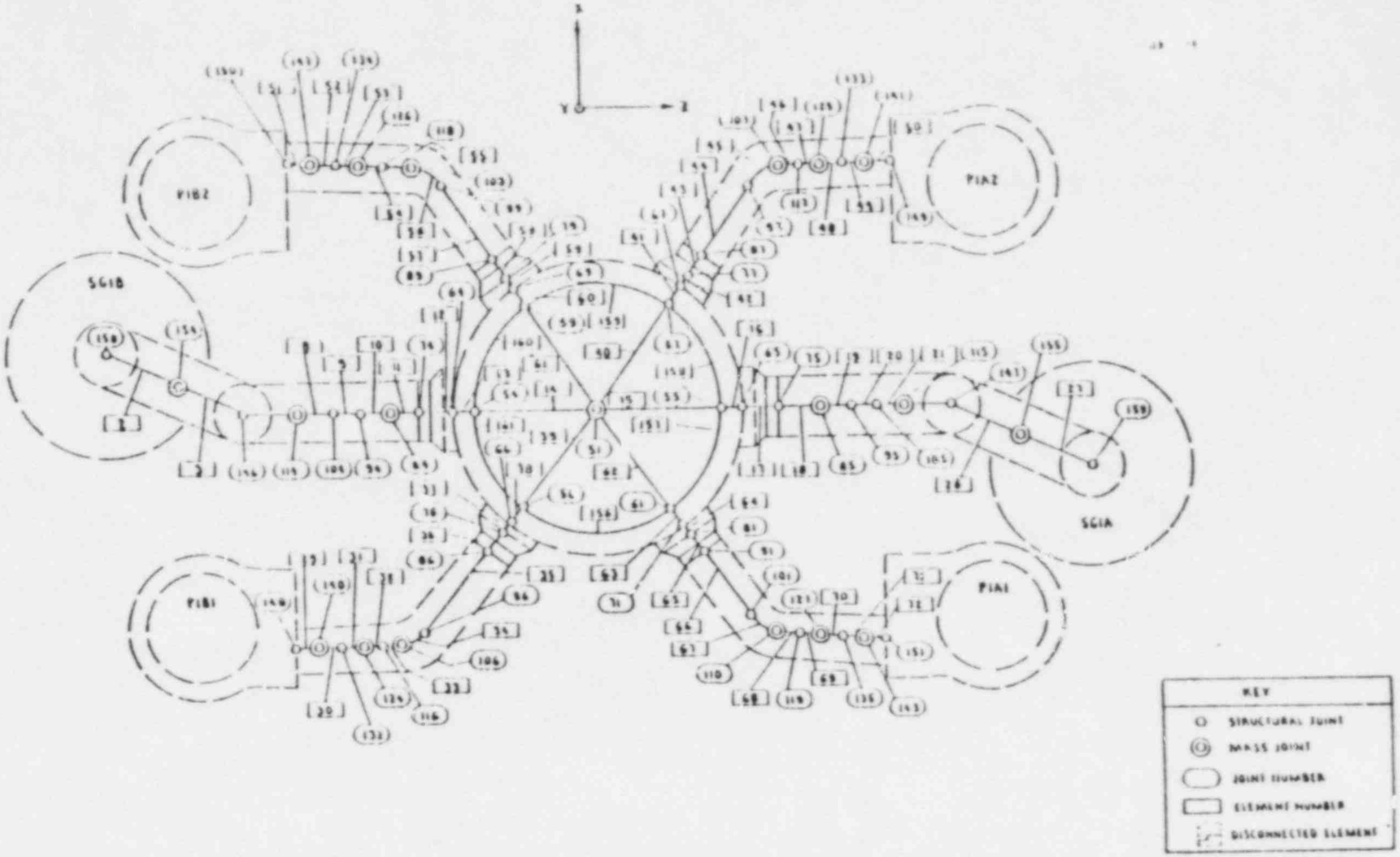


FIGURE 15 RV Isolated Model, Elevation View A-A, Hot Leg

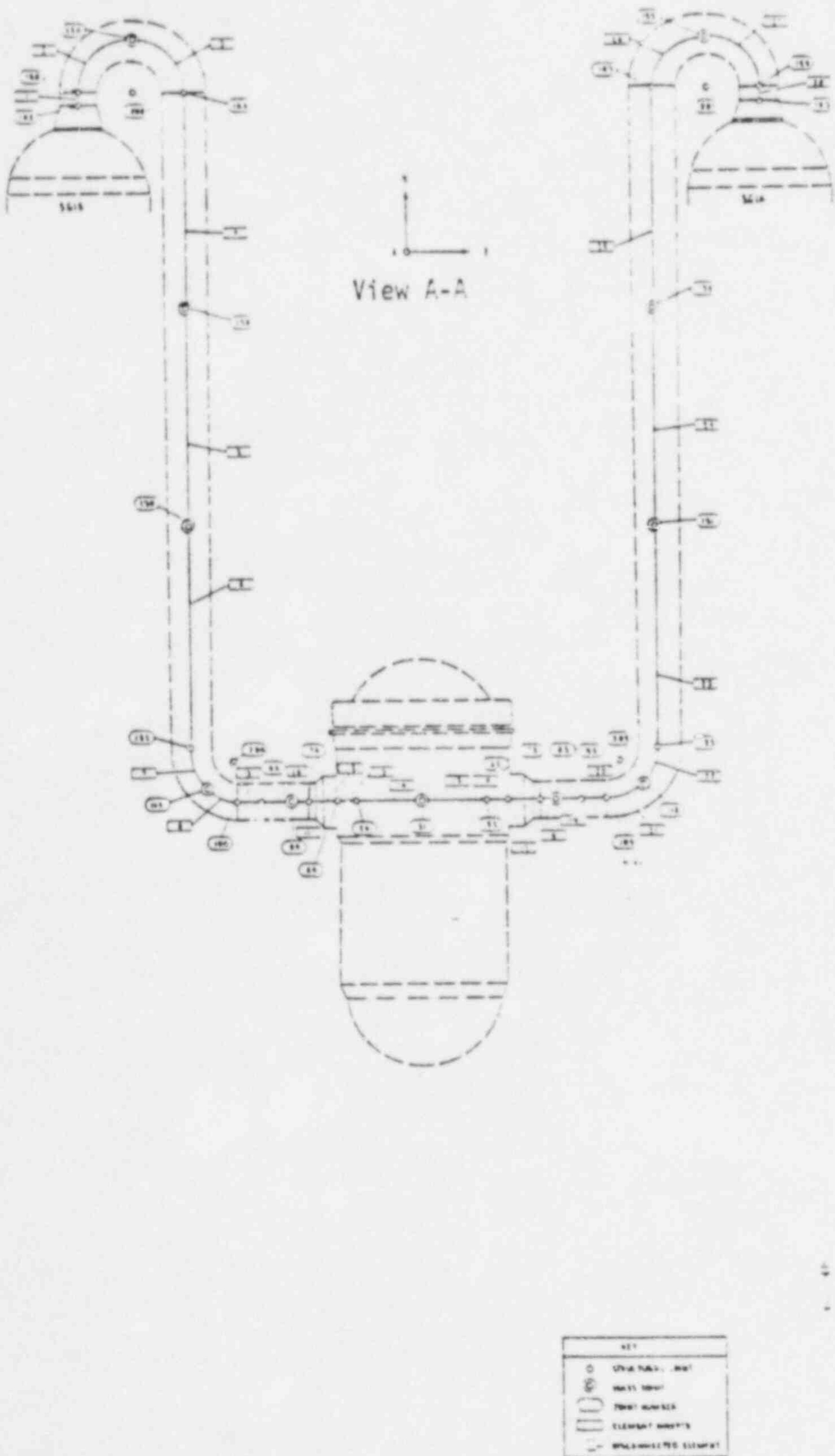


FIGURE 16 Reactor Internals and Service Support Structure

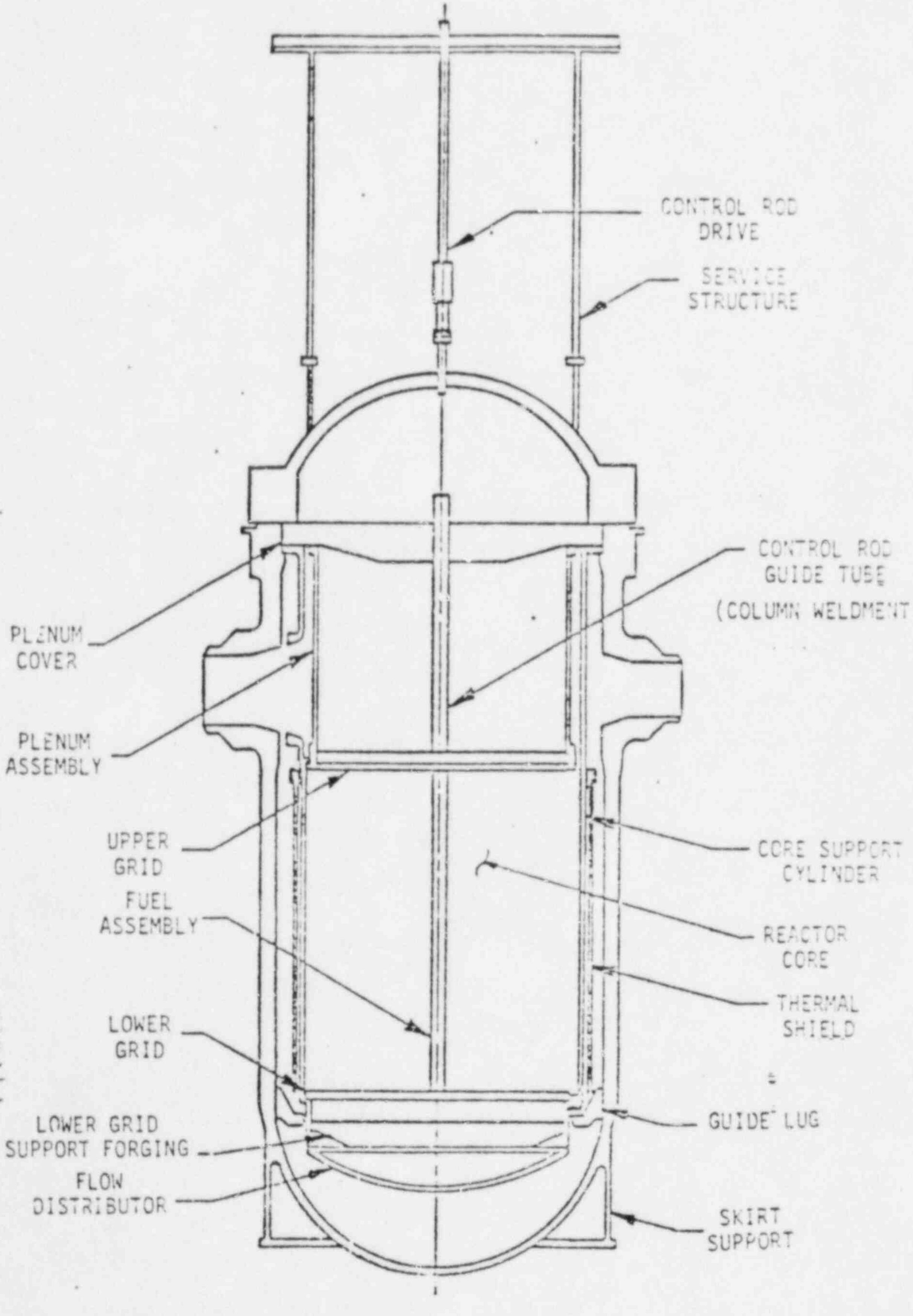


FIGURE 17 INTERNAL WALL STRUCTURE

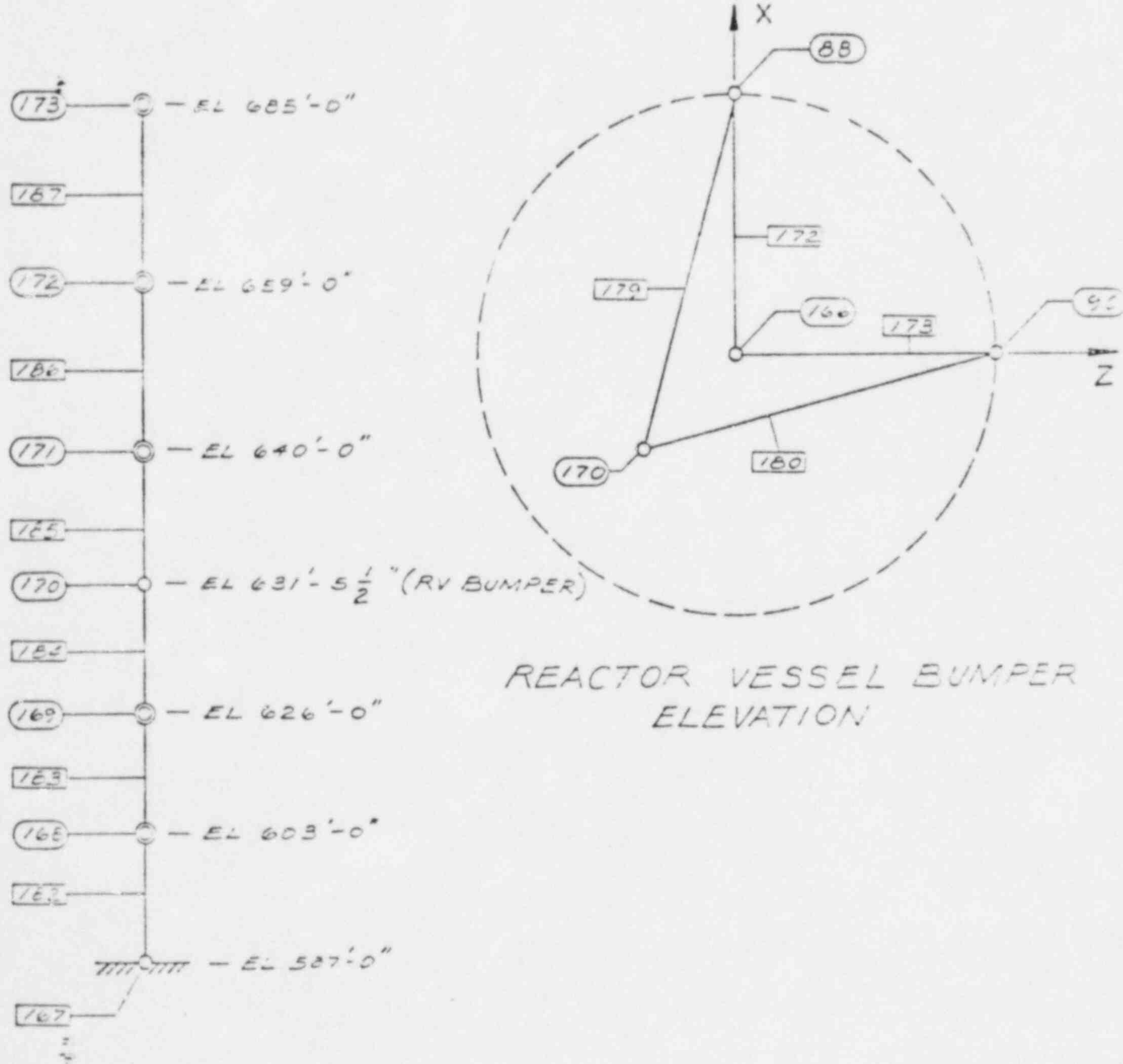
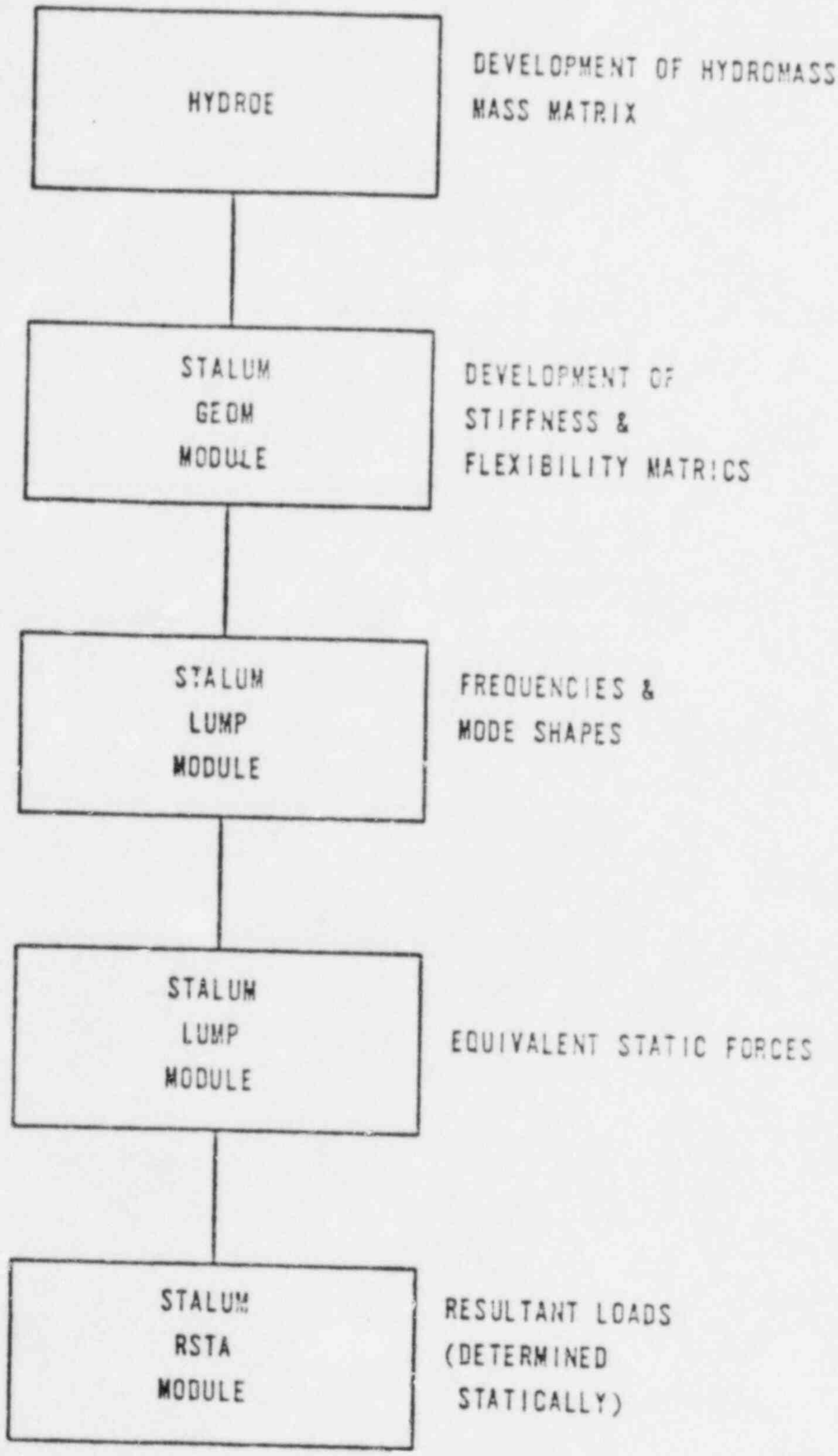


FIGURE 18 Utilization of Computer Programs



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.

015263

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 Boyah DY

Approved by: BGMcElroy FOR L.H.CURTIS

Concurrence by: K.D. Bailey

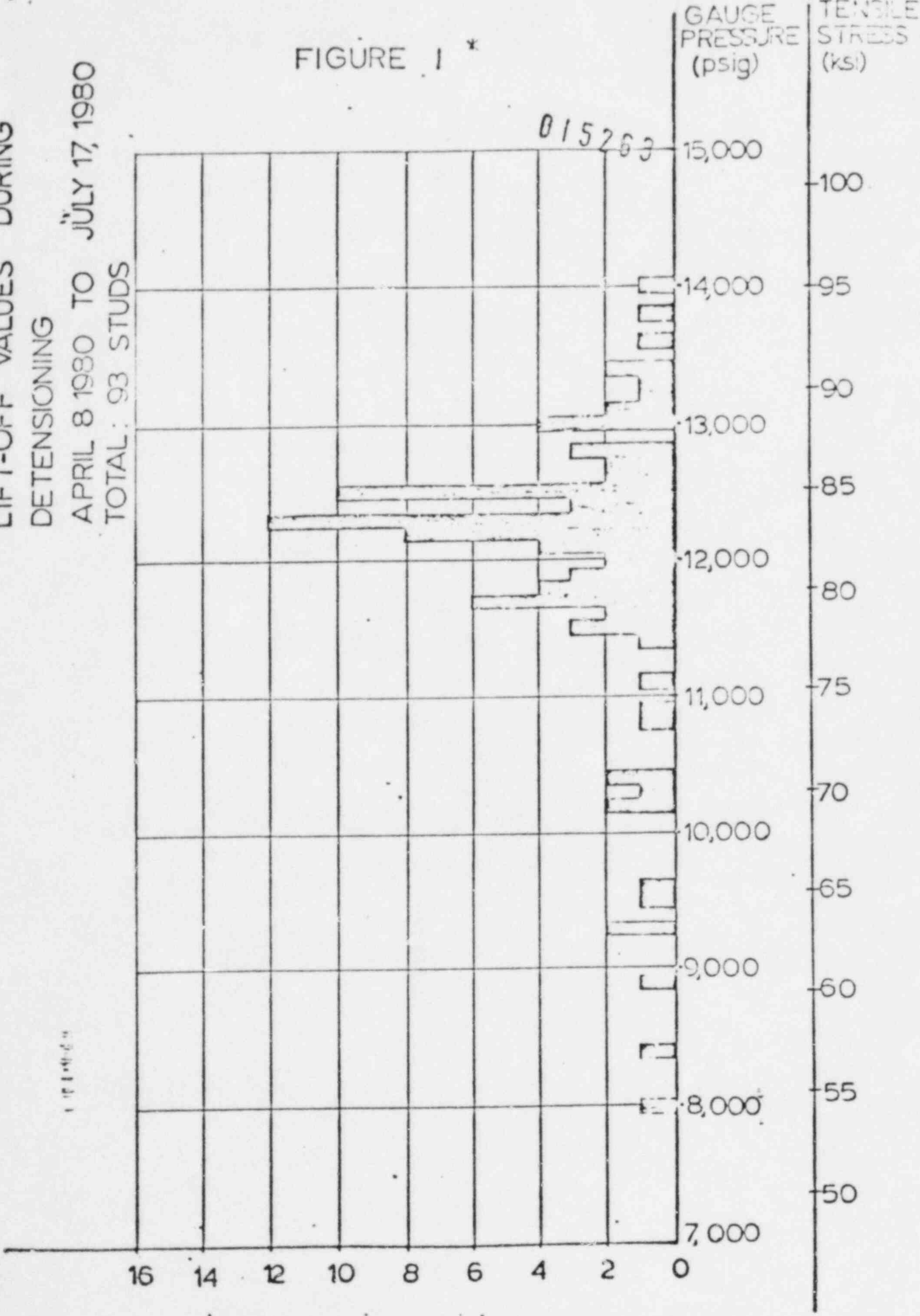
BD/CB/sg

Attachment: Figure 1

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

FIGURE 1 *

015263



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

Jim: What investigation/inspection does this refer to? Is it completed? Please call. Steve Lewis

LAW OFFICES OF
GARAN, LUCOW, MILLER, SEWARD, COOPER & BECKER
PROFESSIONAL CORPORATION
901 EAST JEFFERSON AVENUE
DETROIT, MICHIGAN 48226

THOMAS F. MYERS

(313) 962-5180

September 1st, 1983

PRINCIPAL STAFF			
RA		ENF	
D/RA		SCS	
A/RA		PAO	
DPRP		SLO	
DEPA		RC	✓
DRMSP			
DE			
ML			
OL		FILE	

Mr. James G. Keppler, Director
U.S. Nuclear Regulatory Commission
Region III
799 Roosevelt Rd.
Glen Ellyn, Illinois 60137

Re: Docket No. 50-329, 50-330
Midland Plant, Consumer Power Co.
Our File No. 82 1260 3

Dear Mr. Keppler:

The undersigned represents Mississippi Valley Structural Steel Company (Bristol Steel) a co-defendant in a lawsuit brought by Consumer Power Co. and the Bechtel Power Company against my client and others (Southern Bolt Co. and Rex Heat Treating), as a result of claimed failures of structural support bolts for the nuclear reactors at the Midland Nuclear Plant of Consumer Power Company.

It is my understanding that the U.S. Nuclear Regulatory Commission prepared a report, subsequent to the investigation of your investigator and inspector, Mr. J.E. Foster and Mr. C. M. Erb respectively. I also understand there was some support from your Engineering Section II by Mr. D. H. Danielson.

I would like to take the deposition of Mr. Erb and Mr. Foster at a time and place convenient to them and to your office as well as to all counsel in this matter. I would like to review the Nuclear Regulatory Commission's report with them at the time of deposition. Especially their findings.

We would appreciate hearing from you or someone from your office regarding the possibility of taking these depositions in the near future and on a mutually agreeable basis.

Thank you.

Very truly yours,

THOMAS F. MYERS

~~8309130209~~

TFM/cb
cc: Ronald F. DeNardis, Esq.
Roger F. Wardle, Esq.
Mr. George Schraut
Mr. Norman Cohn
Mr. Ralph Corrin

B/13

SEP 6 1983