APS

Palo Verde Nuclear
Generating Station

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10CFR50.54(f)

102-04603-CDM/SAB/RJR
September 4, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Docket Nos. STN 50-528, 50-529, and 50-530, Units 1, 2 and 3
Response to NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

On August 3, 2001, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles. The bulletin requests that information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles be provided within 30 days of the bulletin's issue date in accordance with 10 CFR 50.54(f). Arizona Public Service Company's (APS) response is enclosed.

The following commitments are being made in this letter:

APS will perform inspections of the VHP nozzles using surface and volumetric examination techniques in accordance with the schedule provided in the enclosure.

APS will provide the information requested by item 5 of NRC Bulletin 2001-01 within 30 days after plant restart following the refueling outage in which the VHP nozzle inspections are performed.

Please contact Thomas N. Weber at (623) 393-5764 if you have any questions or require additional information.

Sincerely,

Gregg R. Overbeck

CDM/SAB/RJR/kg

Enclosure

cc: E. W. Merschoff
L. R. Wharton
J. H. Moorman

A088

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Gregg R. Overbeck, represent that I am Senior Vice President – Nuclear, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

Gregg R. Overbeck

Gregg R. Overbeck

Sworn To Before Me This 4 Day Of September, 2001.

Nora E. Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

April 6, 2003
Notary Commission Stamp

ENCLOSURE

PALO VERDE NUCLEAR GENERATING STATION

RESPONSE TO NRC BULLETIN 2001-01

On August 3, 2001, the Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles. The bulletin requests that information related to the structural integrity of the reactor pressure vessel (RPV) head penetration (VHP) nozzles be provided within 30 days of the bulletin's issue date in accordance with 10 CFR 50.54(f).

This is Arizona Public Service Company's (APS) response to the requested information.

NRC Request

1. All addressees are requested to provide the following information:

   a. the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;

   b. a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;

   c. a description of the RPV head insulation type and configuration;

   d. a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;

   e. a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.

APS Response

1.a. Palo Verde Nuclear Generating Station (PVNGS) Specific Primary Water Stress Corrosion Cracking (PWSCC) Susceptibility Ranking

The PVNGS units have been ranked for the potential for PWSCC of the RPV top head nozzles using the time-at-temperature model and plant-specific input data reported in Electric Power Research Institute's (EPRI) MRP-48. As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take approximately 17.0 additional effective full power years (EFPY) of operation for Units 1, 2 and 3 to reach the same time at temperature that Oconee Nuclear Station Unit 3 (ONS3)

had at the time that its leaking nozzles were discovered in February 2001. By letter dated August 21, 2001 the Nuclear Energy Institute (NEI) submitted MRP-48 on behalf of the industry to the NRC staff. This report provided an industry response to the information requested by the NRC for questions 1a and 1b of NRC Bulletin 2001-01.

Using the criteria stated in NRC Bulletin 2001-01, the three PVNGS Units fall into the NRC category of those plants which can be considered as having a moderate susceptibility to PWSCC based upon more than 5 EFPY but less than 30 EFPY from the ONS3 condition.

1.b. Description of VHP Nozzles

Each RPV head at PVNGS has 97 Control Element Drive Mechanisms (CEDM) nozzles and one reactor head vent nozzle. The nozzle material is SB-166 and the minimum distance between CEDM nozzles is 7.52" (outside diameter to outside diameter) or 11.57" (centerline to centerline). The remaining requested information is provided in Table 2-3 of MRP-48.

1.c. Description of RPV Head Insulation

As reported in Table 2-1 of MRP-48, the PVNGS Units have reflective contoured vessel head insulation. Vendor drawings, DR-4338A-9 through 12, are provided in Attachment 1 to this enclosure and show that this type of insulation configuration cannot be readily removed or modified to allow inspection access.

1.d. Description of RPV Head and VHP Nozzle Inspections Within Past Four Years

APS has performed ASME Section XI visual examinations for leakage for all pressure boundary components as required by the ASME Code and the PVNGS Inservice Inspection Program. As reported in Table 2-1 of MRP-48, APS has not performed specific top of the head bare metal or under the head nozzle inspections on any PVNGS unit within the past four years.

1.e. Description of Missile Shield Configuration, CRDM Housing and Support/Restraint System, and all Components, Structures and Cabling on Top of the Reactor Vessel Head

The missile shields consist of 3 reinforced concrete slabs bolted to the primary shield wall. The missile shield dimensions and weights are as follows:

- East missile shield: 6.5' x 38' x 2.5' thick (47 tons)
- West missile shield: 6.5' x 38' x 2.5' thick (47 tons)
- Center missile shield: 14' x 38' x 2.5' thick (178 tons, including CEDM Air Handling Unit [AHU])

CEDM Lower and Upper Pressure Housing:

The lower pressure housing (i.e., motor housing assembly) is fabricated of type 403 stainless steel with a lower end fitting of inconel and upper end fitting of 347 stainless steel. A lower ledge supports and positions the coil stack assembly. The lower end fitting of the lower pressure housing is designed to mate with the upper end fitting of the CEDM nozzle. The upper end fitting of the lower pressure housing is designed to mate with the lower end fitting of the upper pressure housing. The end fittings utilize acme threads for strength and omega seal welds to provide the water pressure boundary.

The upper pressure housing consisting of the upper and lower end fittings and the guide tube are fabricated of type 316 stainless steel. The lower end fitting of the upper pressure housing is designed to mate with the upper end fitting of the lower housing assembly. The end fittings utilize acme threads for strength and omega seal welds to provide the water pressure boundary. See Attachment 1 for drawing of CEDM assembly.

CEDM housing support/restraint system: None.

Components, Structures, and Cabling from the top of the RPV head up to the missile shield:

Closure head lift rig - A lifting device integrally attached to the RPV head that remains installed during power operation consisting of a support skirt and lifting links. See Attachment 1 for drawing of closure head arrangement and interfaces.

Support skirt - a flanged cylinder (154" diameter x 103" high) that is bolted to the reactor head. This support skirt has thirty-two rectangular slots (9" x 30") to allow airflow to the CEDM AHU ductwork assemblies. The skirt has an internal 2" thick plenum plate that prevents air bypass around the CEDM cooling shrouds and is the support for the CEDM coil stacks and CEDM cooling shroud assemblies.

Lifting links (3) - 10" diameter piping with clevised ends located above the support skirt. A welded horizontal strap assembly holds the links in place. One clevis is always pinned to the support skirt. The others are pinned to the removable delta beam / lifting frame. The support skirt and lifting links remain in place at all times. Since construction they have not been removed from the head. The delta beam is stored elsewhere in containment during power operations.

Cable support structure. - The structure resides directly above the lift rig and is supported off the primary shield wall. It is an integrally welded I-beam rectangular array that supports the cable trays for the cabling mentioned below. The area directly above the CEDM housing does not have any significant intervening steel

structures. The cable support structure weighs approximately 25 tons.

Cabling - Power cables for the CEDM magnetic coil stack assemblies and cabling for reed switch position indication transmitters hang from the cable support structure. They are connected using quick disconnects at the top of the CEDM housings. The other ends of this cabling plug into connectors on the primary shield wall.

Proximity of equipment relative to the missile shield (given in elevation):

- Bottom of missile shield: 157'-6"
- Top of cable support structure: 149'-9"
- Bottom of cable support structure: 141'-11"
- Top of lift rig: 141'-4"
- Top of upper pressure housing: 141'-3"
- Upper pressure housing to lower pressure housing omega seal weld: 126'-3"
- Lower pressure housing to VHP nozzle omega seal weld: 122'-5"
- Top of RPV head: 121'-10"
- Bottom of RPV head: 114'-0"

NRC Request

2. If your plant has previously experienced either leakage from or cracking in VHP nozzles, addressees are requested to provide information :

APS Response

Not applicable to PVNGS

NRC Request

3. If the susceptibility ranking for your plant is within 5 EFPY of ONS3, addressees are requested to provide information:

APS Response

Not applicable to PVNGS

NRC Request

4. If the susceptibility ranking for your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:

a. your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;

b. your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:

   (1) If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

   (2) The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.

APS Response

4.a PVNGS consists of three Combustion Engineering "System 80" Nuclear Steam Supply Systems. Each of the units is on an 18-month refueling cycle. As such, there are spring and fall outages each year. The next scheduled refueling outage begins the end of September 2001 for Unit 3. A Unit 2 outage is scheduled in March 2002, and a Unit 1 outage is scheduled in September 2002.

APS is planning on performing an under-the-head surface and volumetric examination of the VHP nozzles. These examinations will require the use of specialized equipment and qualified examination techniques. These have not been fully qualified for all areas of the VHP nozzle that should be inspected.

The inspection schedule proposed by APS is based on performing a 100 percent under the head inspection using surface and volumetric examination of the VHP nozzles. This inspection will be performed using robotic equipment. APS expects a significant reduction in personnel exposure when performing this type of inspection versus a bare metal visual inspection of the outside of the head. Industry dose estimates for performing under the head inspections using robotics are less than 10 man-rem versus a 70 man-rem estimate at PVNGS for bare metal visual inspections.

The schedule for these inspections is provided in the table below. Specialized equipment will need to be developed to not only perform the examinations, but also to perform any needed repairs and/or perform mitigation techniques as well. APS will be working with other industry groups and suppliers to develop inspection equipment and methodology, including the qualification requirements and acceptance criteria that will provide meaningful inspection and allow proper management of this issue. The inspection plan includes one PVNGS unit each

year starting in the spring of 2003 with the tenth refueling outage of Unit 3. Since the three PVNGS units are identical in design and the time-at-temperature susceptibility rankings are within 1 EFPY of each other, the inspection results in the first unit will provide a good indication of the expected results in the other two units.

INSPECTION SCHEDULE

UNIT	OUTAGE	YEAR
U3	R10	Spring 2003
U1	R11	Spring 2004
U2	R12	Spring 2005

APS will monitor the results of bare metal and under the head inspections performed at other utilities. This information along with the information gained from inspections of the PVNGS Lead Unit will be evaluated to determine if the APS schedule should be modified. Any proposed changes to the current inspection plan for the PVNGS Units will be communicated to the NRC.

APS has evaluated performing a bare metal inspection of the VHP nozzles during the upcoming Unit 3 outage. APS' preliminary evaluation of a bare metal inspection of the RPV head identified that performance of this examination presents significant impact and burden to the plant without the commensurate quality of data as gained through under the head inspections. Additional information is provided below.

The PVNGS reactor heads are insulated with contoured insulation that is difficult to remove. Drawings of this configuration are provided in Attachment 1. This restrictive insulation was installed while the head assembly was still being fabricated. Each insulation section completely encircles several penetrations. Around each penetration is a tight fitting fiberglass collar. Consequently, insulation removal would require the CEDM coil stack/cooling shroud assemblies and lift rig to be removed and each insulation piece to be lifted to the top of the CEDM rod travel housing, a distance greater than 15 feet. Since PVNGS' insulation is in a contour matching that of the head it provides no access without removal. This insulation has not been removed previously, so a more thorough evaluation would need to be performed for this complex task. Special tooling and procedures would need to be developed to perform inspections that would yield meaningful information without insulation removal. It is estimated that removal of the insulation, performance of the inspection and reinstallation of the insulation would be a significant impact to a normal outage schedule.

The level of effort to gain access to the bare metal of the RPV head would be highly dose intensive and would result in considerably more dose than has been incurred by utilities with stepped air reflective insulation that stands off of the RPV

head. The preliminary dose estimate to perform the inspection would be on the order of 70 man-rem. The estimated dose for this inspection alone is in excess of the expected dose for an entire routine outage that averages 65 man-rem. This dose estimate is considerably higher than the estimate provided in EPRI report 2001-50, which estimated 6 man-rem for an expedited bare metal visual inspection.

4.b The alternate inspection method proposed by APS provides for more reliable detection of potential defects than top of the head bare metal visual inspection. A visual inspection will only detect a through-wall crack with sufficient leakage that results in a detectable deposit on the reactor vessel head. Using visual examination methods, it is also very difficult to accurately characterize the source of the leakage, considering the potential for preexisting deposits on the reactor vessel head and other factors that could interfere with the detection of leakage. An under the head volumetric examination would be expected to have the capability to reliably detect cracking before it would become a through-wall crack with visible leakage external to the reactor coolant pressure boundary. APS' inspection plans are considered to be conservative and prudent, yet also provide a more definitive and proactive approach to managing a potential VHP nozzle cracking issue.

4.b(1) Several provisions of the NRC regulations pertain to the issue of VHP nozzle cracking. These include the general design criteria, Title 10 of the Code of Federal Regulations, and the plant Technical Specifications. PVNGS performs all inspections required by these regulations as well as those required by the ASME Code and commitments made in response to NRC issued Generic Letters, Bulletins and Notices. A detailed discussion of regulatory requirements and how the industry is meeting these requirements is provided in MRP-48, Section 3. APS' approach is consistent with the MRP justification. Normal plant inspections will continue during the applicable plant conditions (scheduled refuelings, unanticipated short notice outages, etc), prior to implementation of the proposed "under the head" inspection plan. As discussed in MRP report 2001-50 response to Section 3.0, comment 1, the expectation is that leakage should be detectable given the significant cracking found at Oconee and ANO-1. Note that the axial cracks found at ANO-1 posed no safety concern other than allowing leakage to occur. In addition, leakage was detected at Oconee while sufficient structural margin existed.

Per MRP-44 report, Section 5.2, the maximum circumferential flaw above the j-groove weld for ONS3 was less than the calculated limiting flaw for a pressure of three times design pressure. Per MRP-44 and MRP-48, the remaining ligament in the two ONS3 nozzles with large cracks following the j-groove was 2.2 times the ligament required by Code requirements. This shows additional margin against a conservative postulated CEDM nozzle ejection event. In addition to the above, there is margin provided by the simplistic time-temperature

model correlation for Oconee and PVNGS by a difference of approximately 17 EFPY's. Note that the simplistic model does not include differences associated with the Oconee nozzles. Per MRP-48, it is possible that the more aggressive cracking on the surface of the nozzle is related to certain heats of material produced by B&W Tubular Products. Laboratory tests of specimens removed from ONS3 showed that they had a significant through-thickness hardness gradient with the outside surface being harder than the inside surface. The yield strength measured on a tensile specimen taken from the outer third of the wall thickness of one ONS3 nozzle was 67 ksi. This is higher than the reported yield strength of 49.5 ksi on the nozzle material certification and higher than the maximum reported yield strength of 64 ksi for all other CRDM nozzles in PWR plants in the United States. In summary, the cracks at Oconee and ANO-1 appear different from previous experience, and cracking may currently be limited to certain heats of material from one supplier although there is no proof that this phenomenon is limited to these materials.

A detailed risk evaluation that considers the increase in Core Damage Frequency (CDF) due to the potentially increased probability of a LOCA event due to RPV head penetration nozzle cracks has not been performed. An order of magnitude estimate may be determined by considering the increase in risk to be a product of two terms, the Conditional Core Damage Probability (CCDP), given that a LOCA has occurred and the increase in the frequency of the LOCA Initiating Event (IE) given this potential failure mode; or,

$$\Delta CDF = \Delta IE_{LOCA} * CCDP_{LOCA}.$$

If the break potentially created from a completely failed RPV head penetration nozzle is assumed to be the size of the penetration, approximately 4.25 inches in diameter, the break would be considered a medium LOCA in the PVNGS specific probabilistic risk assessment (PRA). The CCDP for a medium LOCA is 1.50e-02. This probability is conservative for a break location in the top of the Reactor Vessel, because the system success criteria are based on breaks in more demanding locations that have to be considered, such as a reactor vessel cold leg.

Industry data generally ascribe a frequency of 10^{-05}/yr to 10^{-04}/yr for a break of this size. Assuming that this frequency is representative of the frequency of complete ruptures due to the recently observed RPV head penetration nozzle cracks, the resulting increase in CDF would be on of the order of 10^{-06}/yr.

LOCA(s) resulting from a head penetration failure would be bounded by the existing design basis analyses, which demonstrate that the core would be provided with adequate cooling and that core internals would remain in a

coolable geometry. The condition of containment following these scenarios would not require implementation of the severe accident management guidelines. Existing Emergency Operating Procedures provide guidance for the full range of LOCAs and include coverage for multiple events, including reactivity excursions that might occur during the course of an accident. Existing guidelines provide adequate direction to mitigate the transient induced by one or more penetration failures. More than one penetration failure, however, would be beyond the design basis accident for the limiting CEA ejection event. Existing Emergency Operating Procedures cover the reactivity insertion events for periods of highest rod worth, resulting in the same conclusion as stated above.

Based upon the safety basis provided, we believe that there is no significant near-term impact on plant safety in the presence of potential VHP nozzle PWSSC.

APS is confident that the PVNGS Units currently meet and will continue to meet all applicable regulatory requirements pending the completion of planning, scheduling, and performance of VHP nozzle inspections.

4.b(2) APS is planning to perform inspections of the critical areas of the VHP nozzles using surface and volumetric techniques. Corrective actions for indications will be based on the applicable regulatory and Code requirements that APS has committed to at the time of the inspection and repair.

NRC Request

5. Addressees are requested to provide the following information within 30 days after a plant restart following the next refueling outage:
   a. A description of the extent of the VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;
   b. If cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.

APS Response

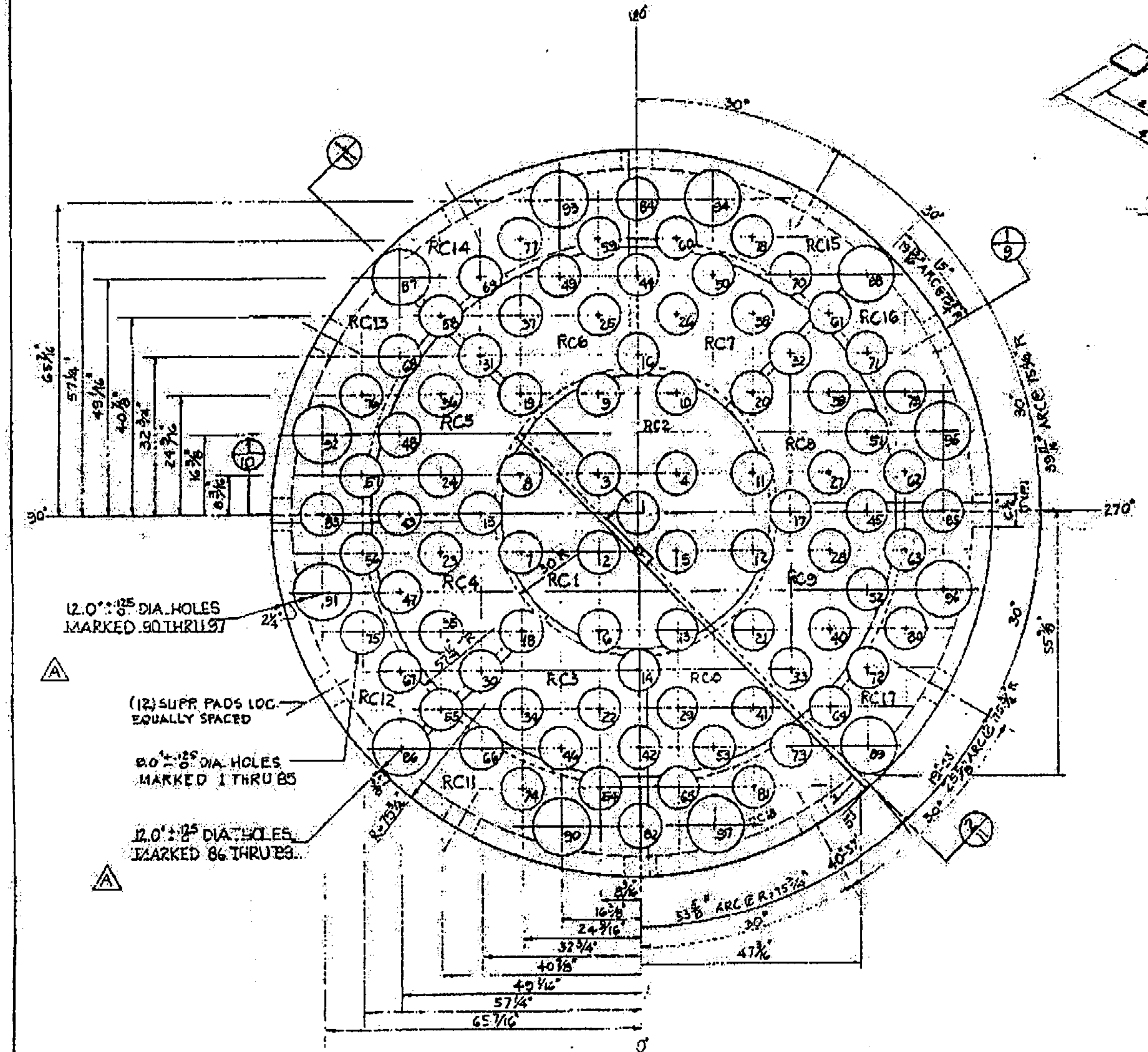
5. APS will provide the information requested within 30 days after plant restart following the refueling outage in which the VHP nozzle inspections are performed.

References

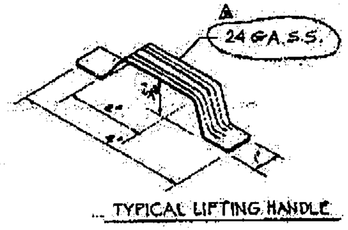
1. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), EPRI, Palo Alto, CA: 2001. 1006284
2. PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations, EPRI, Palo Alto, CA: 2001. TP-1001491, Part 2, Interim Report
3. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-2001-50), EPRI, and Palo Alto. Response to NRC Review Comments transmitted by letter dated June 22, 2001 to NEI

ATTACHMENT 1 DRAWINGS

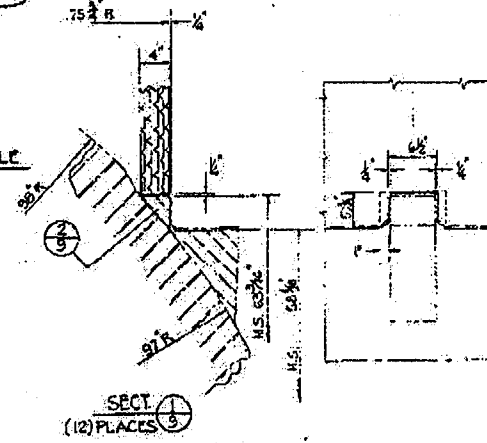
Vendor Drawings of PVNGS' Insulation Configuration
Control Element Drive Mechanism
Closure Head Arrangement & Interfaces



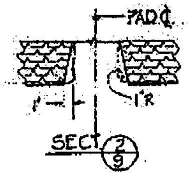
LAYOUT C1
PLAN



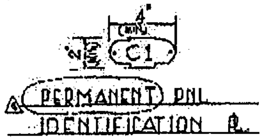
TYPICAL LIFTING HANDLE



SECT 1/9
(12) PLACES



SECT 2/9



PERMANENT PNL IDENTIFICATION ℄

NOTES:

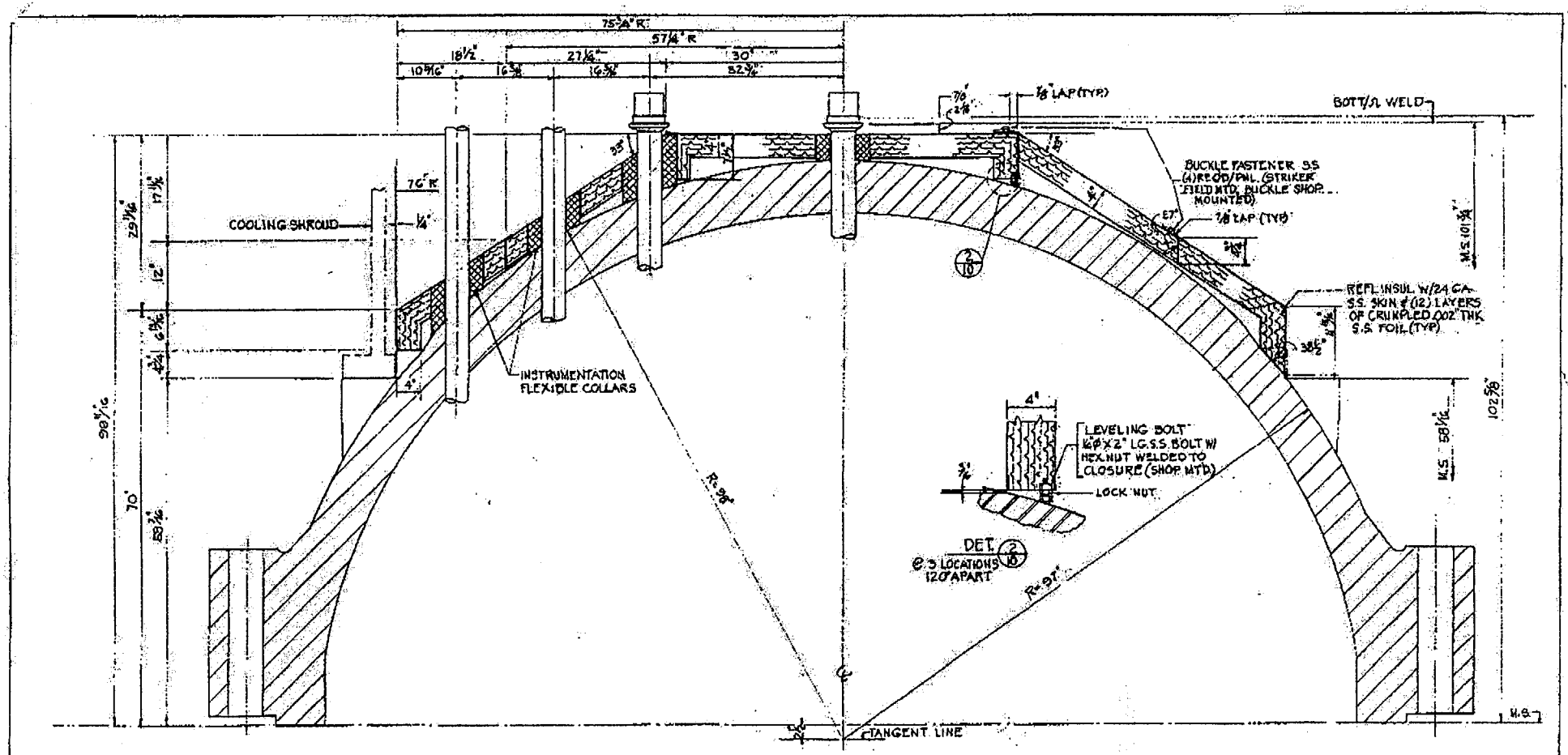
1. ALL INSULATION PANELS ARE DESIGNED FOR REMOVABILITY
2. LOCATIONS OF BUCKLE FASTENERS AND HANDLES SHALL BE DETERMINED AFTER FABRICATION.
3. SEE DWG DR-4338A-11 FOR INSULATION AT VENT PIPE

CH'D	DATE	NO.	REVISION	BY	DATE
JW	10/6/80	B	CHANGED LIFT HANDLE, [illegible]	[illegible]	11-2-80
		A	REV. DIMS. & EXTEND TO 12.0	TNL	9-3-80
		0	ISSUED FOR APPROVAL	PSV	1-15-79

CUSTOMER COMBUSTION ENGINEERING
PROJECT NAME PALO VERDE NUCLEAR STA. UNIT 1
LOCATION ARIZONA NUCLEAR POWER PROJECT
DESIGNER COMBUSTION ENGINEERING

REACTOR VESSEL INSULATION
CLOSURE HEAD INSUL. LAYOUT C1

DRAWN [illegible] DATE 12-5-77
APPROVED RW DATE 8/26/80
SCALE 1" = 12" 1 1/2" - 12"
DWG. NO. DR-4338A-9

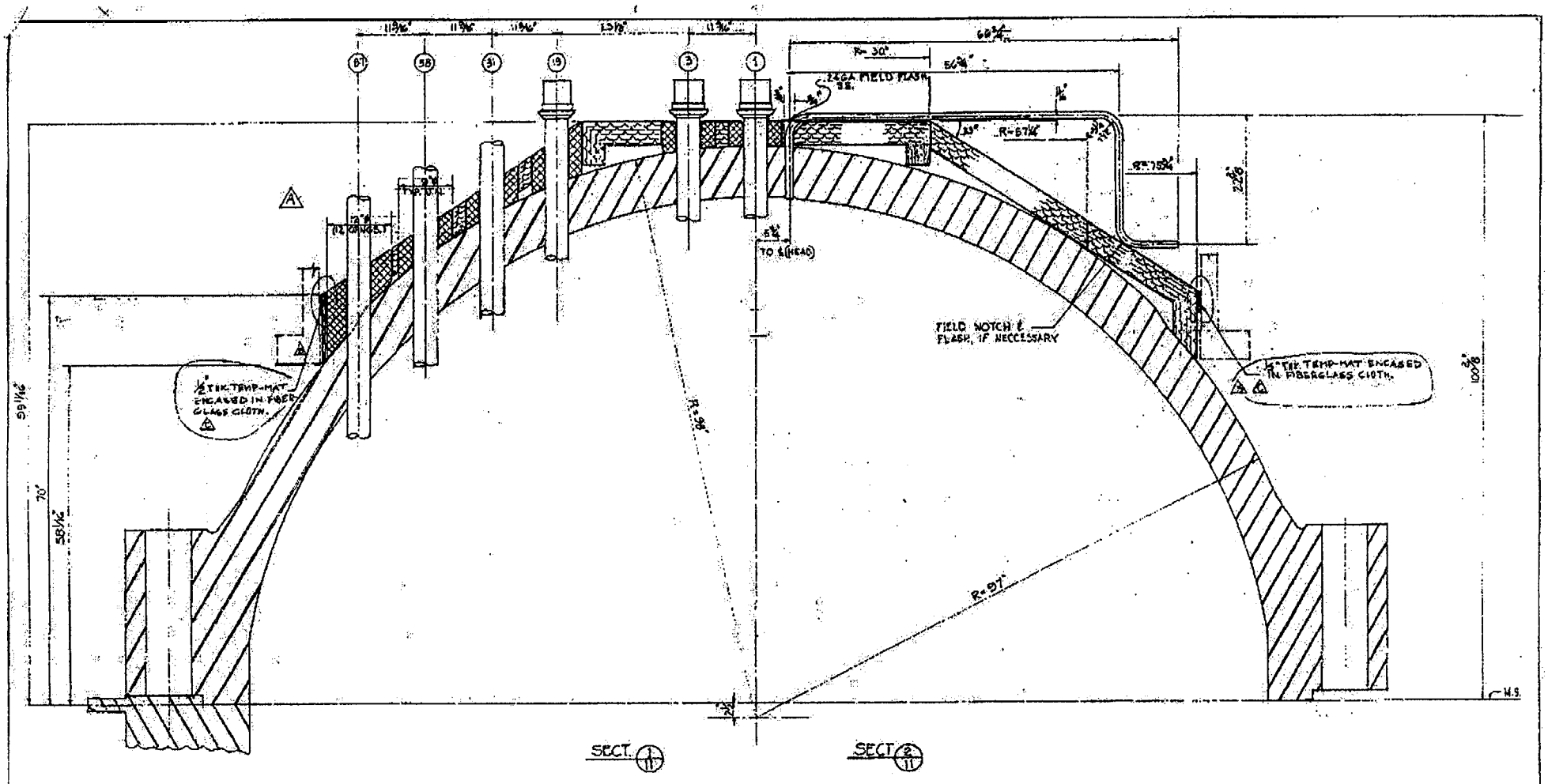


0	ISSUED FOR APPROVAL	PSV	1-18-79
NO	REVISION	BY	DATE

CUSTOMER: COMBUSTION ENGINEERING
PROJECT NAME: PALO VERDE NUCLEAR STA UNIT 1
LOCATION: ARIZONA NUCLEAR POWER PROJECT
DESIGNER: COMBUSTION ENGINEERING

REACTOR VESSEL INSULATION
TOP DOME INSULATION SECTION

DR-4398A-10



NOTE:
SEE DWG. DR-4338A-12 FOR FLEXIBLE COLLARS.

C	REMOVED RIVET ATTACH.	RW	3/26/80
B	ADDED TEKMAT SEAL	AT	1-16-80
A	CHANGE OPTIONAL DWG OF SECT. II	TMA	6-3-80
0	ISSUED FOR APPROVAL	PSV	6-8-79
NO.	REVISION	BY	DATE

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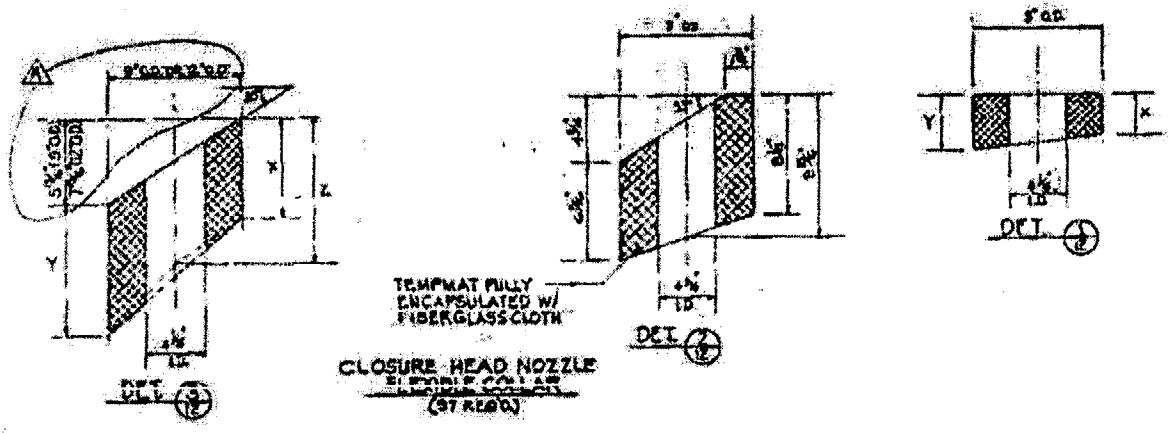
CUSTOMER: COMBUSTION ENGINEERING
PROJECT NAME: PALO VERDE NUCLEAR STA. UNIT 1
LOCATION: ARIZONA NUCLEAR POWER PROJECT
DESIGNER: COMBUSTION ENGINEERING

REACTOR VESSEL INSULATION
TOP DOME INSULATION SECTIONS

DRAWN: [illegible] DATE 12-12-77
APPROVED: RV DATE 8/28/79
SCALE: 1/4 = 1'

DR-4338A-11



CLOSURE HEAD NOZZLE
FLEXIBLE COLLAR
(37 REQ'D)

CLOSURE HEAD NOZZLES
FLEXIBLE COLLARS

HEAD I.D.	NOZZLES	QTY. REQ.	DIMENSIONS X	Y	Z	MARK	DETAIL	O.D.
0	1	1	[illegible]	[illegible]	—	A	1	[illegible]
[illegible]	[illegible]	4	[illegible]	[illegible]	—	B	1	[illegible]
[illegible]	6 - 10	6	[illegible]	[illegible]	—	C	1	[illegible]
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[illegible]	74 - 81	8	[illegible]	[illegible]	[illegible]	N	[illegible]	[illegible]
[illegible]	82 - 85	4	[illegible]	[illegible]	[illegible]	P	[illegible]	[illegible]
[illegible]	86 - 91	6	[illegible]	[illegible]	[illegible]	Q	[illegible]	[illegible]
[illegible]	[illegible]	4	[illegible]	[illegible]	[illegible]	R	[illegible]	[illegible]

NOTES:

1. FIBERGLASS CLOTH MATERIAL SHALL BE ALPHA CO CATALOGUE ITEM # 1925-9385

A	[illegible]	[illegible]	[illegible]
C	ISSUED FOR APPROVAL	[illegible]	[illegible]

COMBUSTION ENGINEERING
PALO VERDE NUCLEAR STA. UNIT 1
ARIZONA NUCLEAR POWER PROJECT
COMBUSTION ENGINEERING

REACTOR VESSEL INSULATION
CLOSURE HEAD NOZZ. FLEXIBLE COLLAR DET.

DR-4338A-12

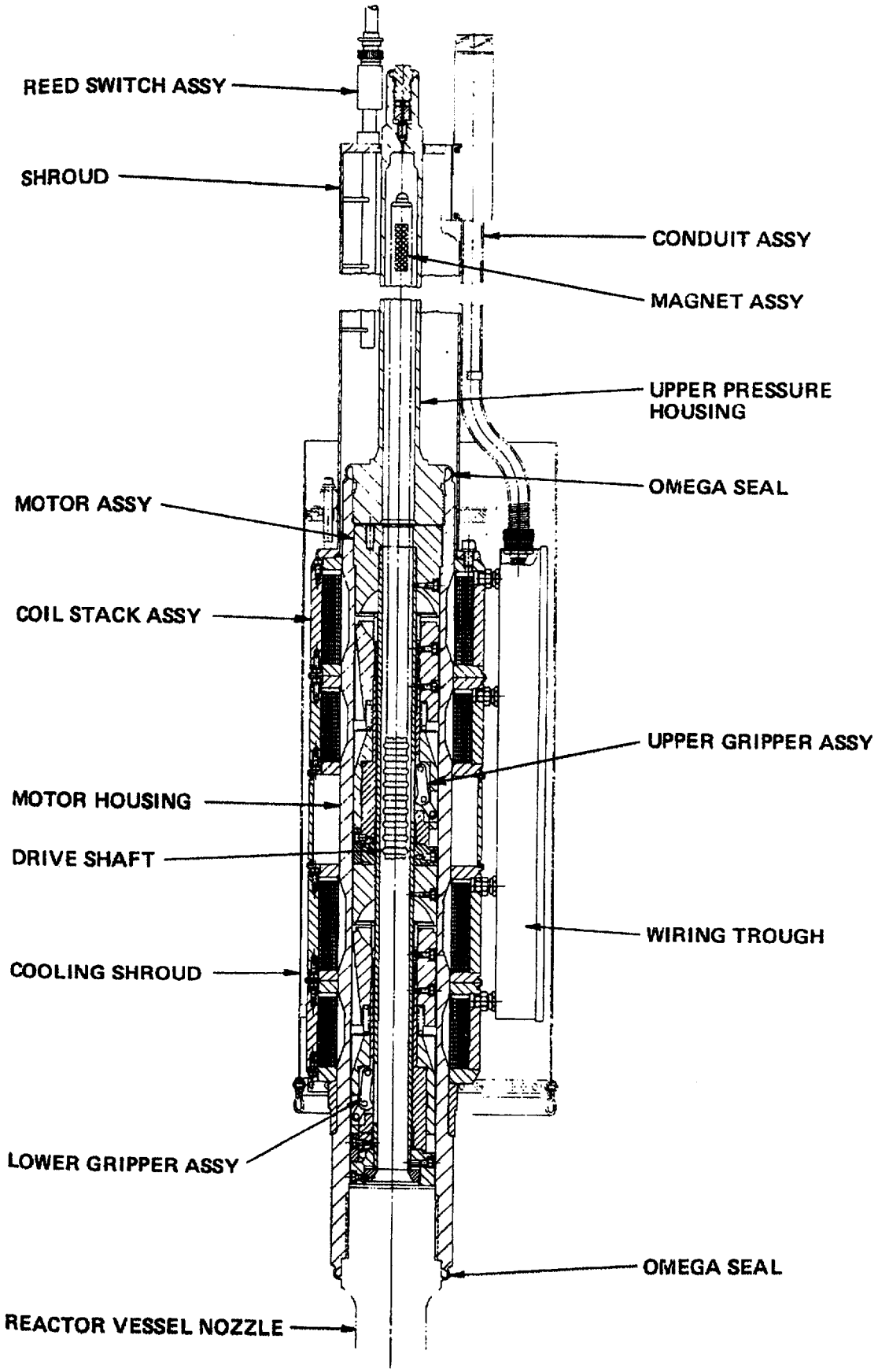
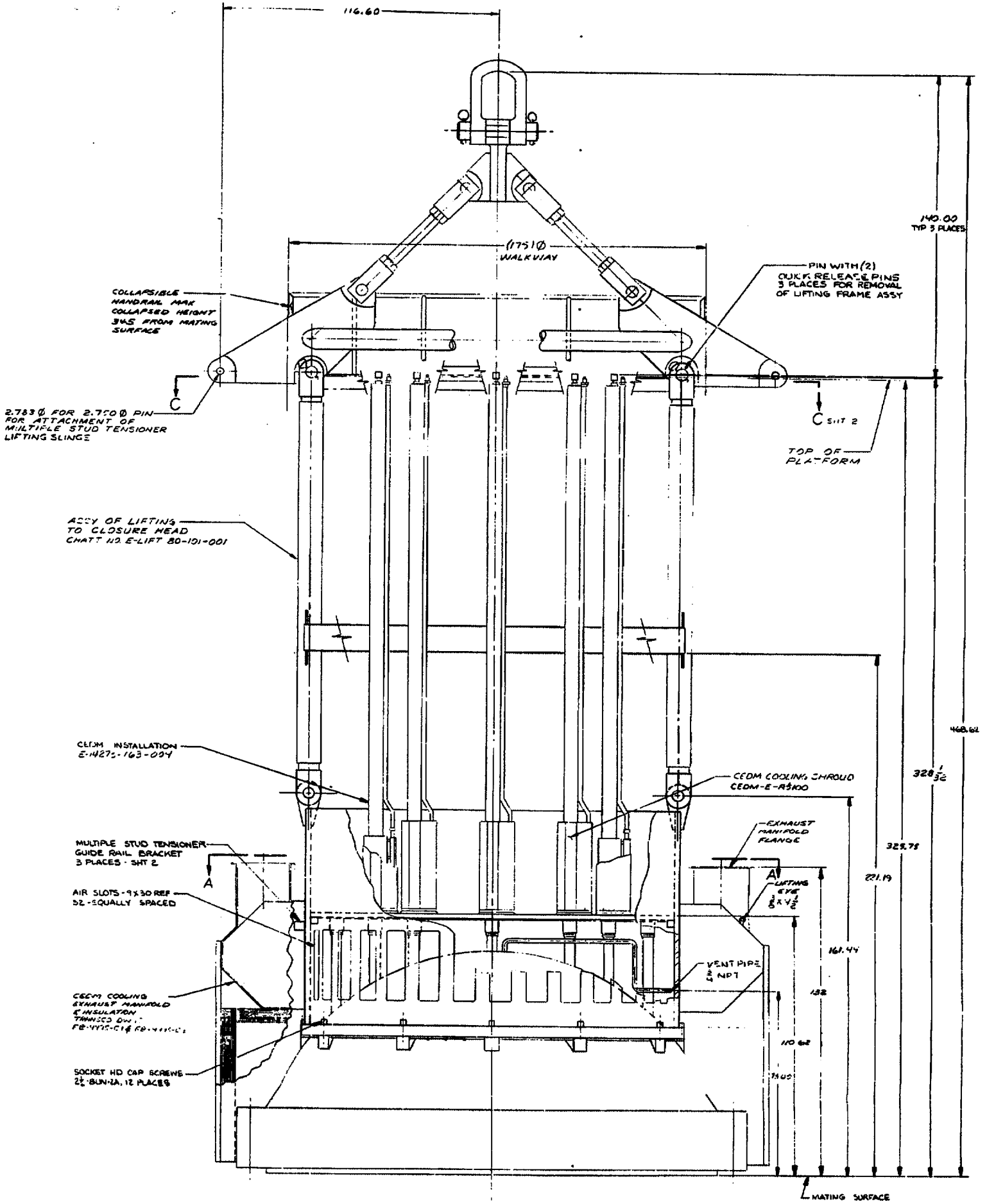


Figure 1-1

CONTROL ELEMENT DRIVE MECHANISM



CLOSURE HEAD ARRANGEMENT & INTERFACES
ELEVATION
SCALE: 3/4" = 1'-0"