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DESCRIPTION:

Letter furnishing additional information to A/O #50-261/75-9, Concerning Corrective Actions and Inspections Subsequent To "C" Reactor Coolant Pump Seal Failure ....
With Attached Appendix A......

ENCLOSURES:

PLANT NAME: H.B. Robinson # 2

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CP&L
Carolina Power & Light Company

July 28, 1975

Regulatory File Cy

File: NG-3513 (R) Serial: NG-75-1139

50-261

Mr. Norman C. Moseley, Director
U. S. Nuclear Regulatory Commission
Region II, Suite 818
230 Peachtree Street, N.W.
Atlanta, Georgia 20303

Dear Mr. Moseley:

H. B. ROBINSON UNIT NO. 2
LICENSE NO. DPR-23
CORRECTIVE ACTIONS AND INSPECTIONS SUBSEQUENT
TO "C" REACTOR COOLANT PUMP SEAL FAILURE

In response to an informal request by a member of your staff, the following summary of actions taken following Abnormal Occurrence 50-261/75-9 is provided for your information. Included are summaries of repairs and inspections performed during the forced outage in order to return the plant to safe operation.

The event which caused this outage was the failure of "C" reactor coolant pump (RCP) seals and subsequent flooding of the containment lower level. This was reported via Abnormal Occurrence Report 50-261/75-9. An account of this occurrence is included below.

On Thursday, May 1, 1975, the plant was operating normally at 100% power having just completed a two-week maintenance outage. At about 1700 the seal leak-off flow on RCP "C" began to oscillate a bit. At 1811 the seal leak-off flow failed high, indicating a failed No. 1 seal. The plant was reduced to 38% power at 10% per minute by control of the turbine (can operate at 40% on two loops). At 1818 RCP "C" was stopped. At 1819 Reactor Trip No. 192 occurred due to "Turbine Trip" caused by high steam generator level. At 1832 RCP's "A" and "B" were stopped when the Component Cooling Return Valve from the RCP's (CCW-626) shut on high flow and would not stay open. Later this proved to be a result of flashing of the component cooling water in RCP "C" thermal barrier cooling coils which caused surges on the return line and on flowmeter FIC-626 (0-150 gpm) which shuts CCW-626 on high flow. There was not continuous high flow in this line, no increase in component cooling surge tank level was noted, and radiation monitor R-17 showed no increase. For these reasons the integrity of RCP "C" thermal barrier cooling coils was concluded to be intact. A high standpipe level on RCP "A" was also received at this time. There was concern that seal flow would be lost on RCP's "A" and "B" since there was flashing and high temperature (300°F) on the seal water return line for these pumps, thus the pumps were secured.

8918

Following the shutdown of RCP's "A" and "B", the plant remained in hot shutdown with temperature and pressure normal for that condition. At 1915 seal leak-off flow was lost to RCP "A" and at 1928 to RCP "B". The objective at this time was to get the plant cooled down so that replacement of the seal on RCP "C" could be made. However, no RCP's were running, and at least one was needed to circulate the reactor coolant to equalize temperatures and boron concentration in the RCS (some boric acid had been charged by this time from the RWST through the charging pumps) and to give correct temperatures indication on the loop bypass line RTD's. At approximately 2130, to try to establish seal leak-off flow on RCP's "A" and "B", four men entered Containment and made an unsuccessful attempt to rotate RCP's "A" and "B" by hand. Pressure on the RCS at this time probably prevented the pumps from rotating. Adjustments were made to seal injection flow, bearing lift oil lineup, and seal leakoff and bypass lineup, but flow could not be established through the No. 1 seals.

The decision was made to start RCP "C". It was started at 2242 (component cooling water had been restored to all three pumps by this time). When the pump had run for a minute, No. 1 seal leakoff was shut to make the seal "film riding." The pump seemed to operate well. Seal leak-off temperature pegged high at 300°F immediately, but motor stator temperature came up normally and all bearings in the pump and motor had normal temperatures.

Main steam isolation bypasses were opened and drawing of a vacuum was begun to cool down the plant. At about 2300, with the core at about 480°F and RCS pressure about 1700 psig, pressurizer level began a slow downward trend. The one-half foot level alarm in the reactor sump was received about this time. At 0015, May 2, after receiving a high standpipe level and with pressurizer level at 12% on the strip chart, the decision was made to stop RCP "C". When RCP "C" was stopped, pressurizer level turned downward at a rapid rate. At 0016 SI Pump "A" was started and the hot leg injection valves opened. Pressurizer level on the strip chart had reached zero by the time, but level indicator LI-462 (cold calibration) read 6% and never bottomed out during the transient. Two minutes after the first SI pump was started, the other two pumps were started. Pressurizer level began to increase and reached zero on the strip chart by 0028. Thereafter, pressurizer level was maintained between about 20% and 80% by cycling of SI pumps. By 0036 RCS pressure had reached 1150 psig and core temperature was below 400°F.

The objective at this point was to reduce temperature and pressure sufficiently to go on RHR and to depressurize as quickly as possible to slow the leak. Valve 311 (auxiliary spray) was used to reduce pressure, and at 0048 pressure was 500 psig. Use of the RHR System was delayed due to letdown valve 460B being shut with the air line to the operator broken. This was repaired allowing pressurization of the RHR System. At 0145 vacuum was broken on the condenser and use of steam was secured. At 0341 the RHR System was warmed up and placed in service with the RCS at approximately 285°F and 400 psig. Cold shutdown conditions were reached at 0448. To reduce the leak rate, it was desirable to reduce system pressure. This was attempted by opening valve 311 (auxiliary spray) to collapse the pressurizer bubble. When valve 311 was opened the RCS pressure did not drop noticeably, but the pressurizer level increased rapidly (more rapidly than charging and SI would raise it). When valve 311 was shut, pressurizer level decreased rapidly as the bubble reformed

in the pressurizer. Core thermocouple temperatures showed the reactor temperature to be stable and later when the reactor head vent was opened, little gas or steam escaped. At 0514 RCS pressure reduction below 100 psig was initiated. At 0715 charging was secured and system draindown commenced with pressurizer level at 55%. By midmorning, the leak was terminated by draining to below +50 inches relative to the reactor vessel flange. At 1530, May 2, RCS level was 39 inches below the reactor vessel flange.

When the leak was terminated there was about 12 inches of water on the Containment ground floor and temperature was at approximately 100°F. RCS boron concentration was 1694 ppm at 1000 hours, May 2.

Following, the incident plans were developed to clean up and decontaminate the affected areas, to perform numerous inspections to assure component integrity, to commence reactor coolant pump repairs, and to perform evaluations of the effects of the event regarding cooldown and associated stresses. A listing of individual areas of concern and the resultant actions taken are provided below:

Reactor Coolant Pump Repairs and Inspections
Westinghouse Assistance and Evaluations
Evaluation of Wetted Insulation and Piping
Evaluation of Containment Flooding
System Flushing
Filter Inspections
Electrical Inspections
Hanger Inspections

Reactor Coolant Pump Repairs and Inspections

Repair work was initiated with the removal of "C" Reactor Coolant Pump from its casing. Upon removal it was determined by survey that radiation levels were excessive, (30 R/hr) for prolonged work on the impeller end. A special procedure was written for decontamination of the impeller end of the coolant pump. Two solutions were used to remove the contamination, an alkaline permanganate and an oxalic acid. Three cycles were used in the effort to reduce the radiation levels. Nonetheless radiation levels at the pump remained high.

It was decided to continue with the disassembly with the pump in a highly contaminated condition. After removal of the impeller and thermal barrier, the shaft was removed. There was obvious shaft damage on the bearing and seal surface area. The damage occurred after the seal failure and during the period in which there was upward flow of primary coolant across the bearing. This overheated the bearing, resulting in its failure and eventual shaft damage. The labyrinth seal was inspected and also found to be damaged due to the failed bearing and seals. Since the labyrinth seals are an integral part of the thermal barrier, this necessitated complete replacement of the thermal barrier.

With the exception of the thermal barrier, all replacement items were in stock on site. A new thermal barrier was obtained from another utility. During reassembly of the pump it was necessary to machine the seating surface

of flange to provide a proper fit of the thermal barrier. After completion of the machining, the affected surface was liquid penetrant tested. The surface was found to be in satisfactory condition. The pump was reassembled and the impeller lock nut was seal welded in place using approved procedures and a qualified welder.

During the installation of new seals in "C" RCP, fragments of the old No. 1 seal were observed to be imbedded in the seal housing wall. These were in a nonseal surface area and were ground out to prevent the possibility of future seal damage. The surface was then liquid penetrant inspected and released for installation. All seals were replaced and the pump was reinstalled in its casing.

While work on "C" pump was progressing, valve CVCS-303C, seal water return isolation valve for "C" pump, was removed and inspected. During the seal failure incident, a flow transmitter downstream of this valve exhibited indication that the valve did not fully close. Though the indication was erratic and could not be identified as true indication, it was decided that an inspection would be in the best interest. The inspection revealed no visible wear or damage to the valve. The valve was reassembled using new gaskets and returned to service.

The motor on "B" reactor coolant pump was removed to permit access to the pump seals for inspection. After seal removal, the seal assemblies and runners were inspected for wear and damage. The No. 1 seal ring and runner were inspected and found suitable for continued use. The No. 2 and No. 3 seal rings showed normal wear, but were replaced based on the amount of time remaining in this cycle. Since seal rings and runners come as matched pairs, the runners were also replaced.

Prior to reinstallation of the motor on "B" pump, the oil lift system orifices were removed and cleaned as part of preventative maintenance. No other problems were identified on "B" Reactor Coolant Pump.

The motor was removed on "A" Reactor Coolant Pump to allow access to the shaft seals. These seals were disassembled and inspected for wear. No. 3 seal ring exhibited evidence of wear and was replaced for the same reason as those were on "B" pump. All other seal rings and runners were suitable for continued operation. The orifices of the oil lift system on "A" were removed and cleaned.

All work was performed under the supervision of a Westinghouse representative and a CP&L job coordinator. The job coordinator also provided QA surveillance coverage.

Westinghouse Assistance and Evaluations

Vendor assistance, employed as a result of the event, was provided by Westinghouse Electric Corporation. Westinghouse reactor coolant pump and motor service representatives were utilized in performing pump repairs and inspection. Westinghouse also provided evaluations of pipe and reactor

vessel stresses that resulted from the excessive cooldown following the RCP seal failure. This study indicated that no significant stresses resulted that would jeopardize plant safety.

Westinghouse also provided recommendations regarding action to be taken in respect to wetted insulation and piping resulting from the seal failure and subsequent flooding. These recommendations and final action are addressed in the following section.

A meeting was held with NRC ONRR on May 12, 1975 in Bethesda, Maryland to provide them with first-hand information of the details of the event. Westinghouse assisted Carolina Power & Light in relating the incident, and as a result of questions arising during the meeting, provided an evaluation of the leak rate which occurred as a result of the failure. Original estimates were that a loss of pump seals would only result in a maximum leakage of about 100 gpm. A much higher leakage was experienced in this case. In order to explain the higher leak rate, Westinghouse prepared a report defining the leakage and estimating its magnitude. This study was submitted to ONRR and posed no further concern. A report was also prepared and submitted regarding the loss of No. 1 seal flow on the "A" and "B" RCP's. No firm conclusion as to the cause was formed, but the study established that the most probable cause was small particles which interrupted the flow across the seal face.

Evaluation of Wetted Insulation and Piping

As a result of the containment flooding, piping near the containment base mat was submerged. The stainless steel piping, which was submerged, consisted of the seal water injection line to RCP "A", excess letdown lines, normal letdown lines, residual heat removal line to "A" loop, and safety injection line to "A" loop cold leg. This constitutes approximately 650 feet of piping. Portions of this piping were flooded from early Friday morning, May 2, 1975 to about 0400 Sunday morning, May 4, 1975. The chloride concentration of the water was 0.25 ppm as measured on May 2, 1975. Therefore, there was a concern for the saturation of the insulation with chlorides, possible leaching out on piping, and the potential for chloride stress corrosion. Westinghouse was contacted to provide recommendations for corrective action.

The initial Westinghouse reply was received on May 6, 1975. Recommendations were to remove insulation from all submerged stainless steel piping, clean the pipe, and reinsulate. It was also suggested to remove insulation from the bottom of the reactor vessel and to rinse the vessel and clean the incore thimble penetrations. The specification, which was referenced for use in the evaluation, was Westinghouse PS 84351 NL, Revision 2, "Determination of Surface Chlorides and Fluoride Contamination on Stainless Steel Material." This specification requires a maximum chloride concentration of 0.0015 mg/dm^2 for insulated surfaces.

Initial plans were then made to remove all the insulation and proceed as suggested. However, a sampling program was begun on loop piping that was not wetted by the incident and the chlorides on this piping were found to be higher than the referenced acceptance limits. Based on this as-found condition and a concern for minimizing radiation exposures of workers required

to remove insulation and reinsulate the submerged piping, Westinghouse was again contacted to provide clarification and/or justification for the proposed acceptance criteria. It was at that time that Westinghouse recommended sampling the piping and piping insulation with respect to the acceptance criteria in Westinghouse PS 83336 KA, "Requirements for Thermal Insulation Used on Austenitic Stainless Steel Reactor Plant Piping and Equipment." This was done and the results indicated that the insulation and piping were acceptable. Westinghouse was then requested to provide a justification for the applicability of the process specification for insulation to the piping. A report was then received on May 21, 1975 justifying the acceptance standards. This report concluded that the data falling within the acceptance curve of the subject specification indicated that sufficient amounts of sodium silicate were present in the insulation to inhibit chloride stress corrosion. Further, there is no concern for the contamination if the surface is covered with sodium silicate inhibited insulating material when it is ascertained that there is sufficient silicate inhibitor to prevent halide stress cracking. This mechanism for inhibition of halide stress corrosion was established by H. F. Karnes in his report, "The Corrosive Potential of Wetted Thermal Insulation," presented at the AICHE 57th National Meeting, September 26-29, 1965. With this final justification, it was decided that the wetted piping and insulation were acceptable without replacement.

Other work related to the flooding consisted of removing insulation from the reactor vessel bottom, sampling for chlorides on the incore penetrations, and rinsing the vessel. This was completed and no chlorides were detectable following the sampling. The entire vessel, loop piping, or the vessel safe ends were not submerged. The piping in the area of the spill was sprayed with primary coolant but was not submerged. The chloride content of the primary water was not such as to present a stress corrosion problem and none of the insulation was replaced. A detailed report has been submitted regarding the insulation wetting and concern regarding chloride contamination.

Evaluation of Containment Flooding

In order to determine the effects of flooding on the floor of the containment vessel, it was first necessary to approximate the quantity of accumulated water. Three methods of approximation were used, i.e., calculating the wetted volume of the containment vessel; calculating the quantity of water removed from the containment vessel; and calculating the quantity of water emptied into the containment vessel. The quantities of water determined by these methods (See Appendix A, calculations 1, 2&3) were 135,105 gallons; 133,000 gallons; and 129,000 gallons, respectively.

For conservatism, Ebasco was requested to investigate the effect of flooding the containment floor by 200,000 gallons of water. Their review concluded that there had been no significant effect on the containment floor which would warrant further review prior to continued operation.

Subsequent to the failure of "C" Reactor Coolant Pump seal, the water level on the floor of the containment vessel reached a height no greater than 12 3/4". This was determined by measuring "high water marks" on the containment walls at various locations. The average height of water was approximately 12 1/2".

The elevation of the containment vessel floor is 228'0", and the elevation of the reactor vessel nozzle centerline is 242' - 2 11/16". Therefore, no more than 14' - 1 25/32" of the reactor vessel was submerged with the maximum water level at 13' - 1 15/16" below the nozzle centerline and 10' - 7 3/32" below the nozzle supports.

These dimensions are summarized in Figures 1 and 2.

System Flushing

A portion of the CVCS system was flushed to remove any foreign particles which might cause damage upon a return to operation. Filter cloth was placed over the openings of drain and vent connections. The volume control tank was filled to approximately 50% and pressurized to provide a driving head for flushing water. Lines to the charging pumps and charging line flow control valve were flushed. The flush was performed in accordance with a special procedure, and the flush continued until the flushing water condition was within the acceptance limits. These acceptance limits were as follows:

1. In each flush path, after approximately 50 gallons of water has been flushed through, a filter cloth was placed over the end of the exit valve or pipe and approximately 5 gallons flushed through the cloth. Two successive flushes must meet the acceptance criteria.

2. The general appearance of the filter cloth shall be that of a clean white wet cloth showing no more than slight speckling and no more than slight soiling or staining from rust or dirt.

3. There shall be no particles on the cloth larger than 1/32" in any dimension except that fine hair-like slivers or thin flakes (much less than 1/32" thick) may have a major dimension up to 1/16".

4. Readily apparent quantities of unusual impurities in the exit flush water or on the cloth such as resin particles, abrasive grit, oil, or other foreign matter shall be reason for nonacceptance of the flush.

The flush was successfully completed prior to filling of the Reactor Coolant System after repairs to "C" pump.

A flush of the Residual Heat Removal systems was performed to remove water from the lines which came from the containment vessel floor when water was pumped to the refueling water storage tank. The water in these lines was removed by either draining to floor drains or pumping to floor drains. These drains terminate at the auxiliary building sump. After draining, the lines were rinsed with primary water to remove any remaining material. This was satisfactorily completed during the repair outage using a special procedure approved by the PNSC.

Filter Inspections

In addition to the flushing previously discussed, it was decided to inspect the seal water injection supply filters and the reactor coolant filter to determine if any fragments potentially generated by the pump seal failure

were entrapped. There are two seal water injection supply filter housings and one reactor coolant filter. The seal water filters are in parallel and are used alternately rather than concurrently.

The seal water filters were replaced during the April steam generator outage prior to the pump failure. The seal water return filter had been bypassed earlier due to a small leak. This allowed the return water to enter the volume control tank unfiltered. Therefore, any particles returned would be entrained on the injection supply filters.

A radiation survey of the filters indicated that the filter that was in service during the event read 150R/Hr on contact of the housing. A survey of the spent cartridge indicated approximately 500 R/Hr on contact. The filters were black with no removable surface particles. Smear samples were taken for analysis. The results of the analysis revealed the materials were corrosion products.

The second housing which had not been in service was opened and all cartridges were clean. In both housings the filters were intact and properly installed. New filters were inserted in both housings at the completion of the inspection.

The reactor coolant filter was changed during the week of May 19. It was not expected that material other than normal system corrosion products would be entrained on the filter, and no fragments that would possibly have been generated by the seal failure were detected when the filters were replaced.

Electrical Inspections

Subsequent to the flooding which occurred in the containment vessel, an investigation was performed to determine the extent of moisture damage to instrumentation/electrical components.

Random checks were performed on approximately 60% of the valve limit switches located inside of containment. This consisted of all limit switches accessible without constructing scaffolding. No water or moisture was noted in any of the limit switches checked.

Transmitters were checked by removing covers and performing a visual inspection for water damage. The following instruments were included:

Instrument	Function
PI-405	Reactor Coolant System Pressure
LT-484	S/G 2 Ch. 1 Narrow Range Level
LT-485	S/G 2 Ch. 2 Narrow Range Level
LT-486	S/G 2 Ch. 3 Narrow Range Level
LT-487	S/G 2 Wide Range Level
FT-484	S/G 2 Ch. 1 Steam Flow
FT-485	S/G 2 Ch. 2 Steam Flow

Instrument	Function
PT-155	No. 1 Seal Δ P RCP No. 2
PT-128	RCP Loop 2 Thermal Barrier Δ P
FI-491	RTD Bypass Line Flow Indicator
LT-462	Pressurizer Level Cold Calibration
PT-444	Pressurizer Pressure Control
PT-445	Pressurizer Pressure Control
PT-458B	Pressurizer Pressure Calib DP Cell
LT-459	Pressurizer Level Ch. 1
PT-455	Pressurizer Pressure Protection Ch. 1
FIC-678	Rod Control Drive Cooler Cooling Water Flow Indicator
LT-460	Pressurizer Level Ch. 2
PT-456	Pressurizer Pressure Ch. 2
FT-932	SI Flow to RCS Hot Leg
FT-933	SI Flow to RCS Hot Leg
LT-461	Pressurizer Level Ch. 3
PT-457	Pressurizer Pressure Ch. 3
LT-494	S/G 3 Ch. 1 Narrow Range Level
LT-495	S/G 3 Ch. 2 Narrow Range Level
LT-496	S/G 3 Ch. 3 Narrow Range Level
LT-497	S/G 3 Wide Range Level
FT-494	S/G 3 Ch. 1 Steam Flow
PT-154	#1 Seal Δ P RCP #3
PT-125	RCP Loop 3 Thermal Barrier Δ P
FT-495	S/G 3 Ch. 2 Steam Flow
PT-403	RCS Pressure Narrow Range
PI-404	Reactor Coolant System Pressure
PI-154B	No. 1 Seal Δ P RCP 3
FI-492	RTD Bypass Line Flow Indicator
PT-138	Excess Letdown HX Outlet Pressure Indicator
TT-1058	Reactor Coolant Drain Tank Temperature
PT-1004	Reactor Coolant Drain Tank Pressure
LT-1003	Reactor Coolant Drain Tank DP
FI-490	RTD Bypass Line Flow Indicator
FT-475	S/G 1 Ch. 2 Steam Flow
LT-477	S/G 1 Wide Range Level
LT-476	S/G 1 Ch. 3 Narrow Range Level
LT-475	S/G 1 Ch. 2 Narrow Range Level
LT-474	S/G 1 Ch. 1 Narrow Range Level
PT-131	RCP Loop 1 Thermal Barrier Δ P
PT-156	#1 Seal Δ P RCP #1
FT-474	S/G 1 Ch. 1 Steam Flow
LT-922	Accumulator A Level
LT-920	Accumulator A Level
LT-CMS-4	Condensate Measuring System Level
FT-414	RCS Flow Loop 1
FT-415	RCS Flow Loop 1
FT-416	RCS Flow Loop 1
PT-921	Accumulator A Pressure
PT-923	Accumulator A Pressure

Instrument	Function
FIC-156	No. 1 Seal Bypass Flow RCP #1
FT-156B	No. 1 Seal Leakoff RCP #1 Lo Range
FT-156A	No. 1 Seal Leakoff RCP #1 Hi Range
PT-925	Accumulator B Pressure
PT-927	Accumulator B Pressure
LT-926	Accumulator B Level
LT-924	Accumulator B Level
LT-CMS-3	Condensate Measuring System Level
FT-424	Reactor Coolant System Flow Loop 2
FT-425	Reactor Coolant System DP
FT-426	Reactor Coolant System Flow Loop 2
FT-155B	No. 1 Seal Leakoff RCP #2 Lo Range
FIC-155	No. 1 Seal Bypass Flow RCP #2
LT-CMS-2	Condensate Measuring System Level
FIC-154	No. 1 Seal Bypass Flow RCP #3
FT-154B	No. 1 Seal Leakoff RCP #3 Lo Range
FT-154A	No. 1 Seal Leakoff RCP #3 Hi Range
FT-434	RCS Flow Loop 3
FT-435	RCS Flow Loop 3
FT-436	RCS Flow Loop 3
LT-CMS-1	Condensate Measuring System Level
LT-928	Accumulator C Level
LT-930	Accumulator C Level
PT-929	Accumulator C Pressure
PT-931	Accumulator C Pressure

Evidence of moisture was found only in FT-154A and FT-154B. Additionally, FT-154A was found to be stuck in the extreme high position apparently as a result of high seal water flow. Both rotometers were cleaned, recalibrated and satisfactorily tested.

Rod drive cables were meggered and found in good working order. The rod position indication cables were resistance checked. Moisture was detected in two connectors between containment penetrations and cable inside of containment. These connectors were disassembled, dried, and reassembled.

Water level indicators in the sump were checked and evidence of moisture was observed. These indicators were disassembled, dried and reinstalled. The water level detector in the reactor coolant pump bays operated properly when tested.

All instrumentation and electrical components checked were left in satisfactory operating condition.

Hanger Inspections

All pipe hangers and supports in "C" Pump Bay and on the Safety Injection lines used during the event were inspected. The following discrepancies were observed:

Line Number	Location	Discrepancy
1. 2-SI-56	Downstream of valve 866B on SI line to loop 2 hot leg	Missing "U" Bolt
2. 8-SI-37	Near "C" Accumulator upstream of check valve 876A	Loose Pipe Hanger
3. 8-SI-38	Downstream of valve 744B on line from RHR Pumps to loop 2 cold leg	Two spring hangers show no weight
4. 14-AC-9	Downstream of valve 751 on RHR loop supply	Spring hanger shows no weight
5. 1-RC-151R	No. 2 Seal Leakoff in "C" Pump Bay	2 Missing "U" Bolts

The "U" bolts were replaced on items 1 and 5. The pipe hanger on item 2 was tightened. The spring hangers in items 3 and 4 were adjusted to the proper tension.

No other discrepancies were noted. There is no indication that the hanger deficiencies were a result of the incident. The hangers are on portions of the system not included in the inservice inspection scope, and the problems apparently existed prior to the pump failure.

The above action was completed with all plant concerns as to safety resolved and the plant returned to operation on May 27, 1975.

Very truly yours,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:jwk

Attachments

cc: Messrs. N. B. Bessac
T. E. Bowman
P. W. Howe
J. A. Jones
R. E. Jones
W. B. Kincaid
D. Knuth
J. B. McGirt
D. B. Waters

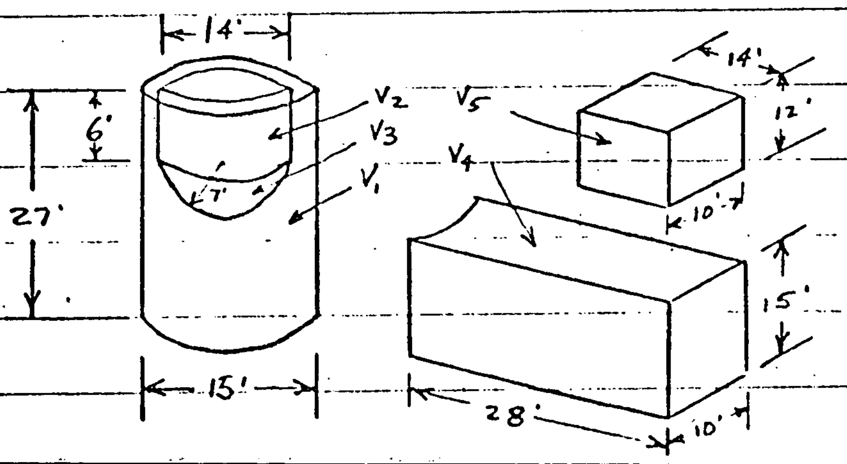
APPENDIX A

Calculating the Wetted Volume of the Containment Vessel

V_{cv} - Wetted volume of the containment vessel
V_s - Volume of containment sump
A_1 - Containment floor area at 228' level outside the polar crane wall
A_2 - Containment floor area at 228' level inside the polar crane wall
H_w - Height of water above the 228' level ($12\frac{1}{2}'' = 1.04$ ft)

$$V_{cv} = V_s + (A_1 + A_2) H_w$$

V_s can be approximated as the volume of a cylinder, V_1 minus the volumes of a cylinder, V_2 and a hemisphere (the reactor vessel), V_3 plus the volumes of two rectangular prisms, V_4 & V_5.



$V_s = V_1 - (V_2 + V_3) + V_4 + V_5$

$V_1 = \pi r^2 h$
$= 3.14 (7.5)^2 27$
$= 4771 \text{ ft}^3$

$V_4 = l w h$
$= 28(10)(15)$
$= 4200 \text{ ft}^3$

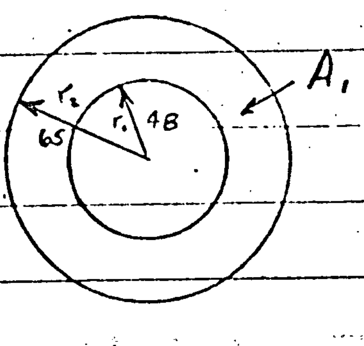
$V_2 = \pi r^2 h$
$= 3.14 (7)^2 6$
$= 924 \text{ ft}^3$

$V_5 = l w h$
$= 14(10)(12)$
$= 1680 \text{ ft}^3$

$V_3 = \frac{1}{2}\left(\frac{4}{3} \pi r^3\right)$
$= \frac{4}{6} (3.14)(7)^3$
$= 718 \text{ ft}^3$

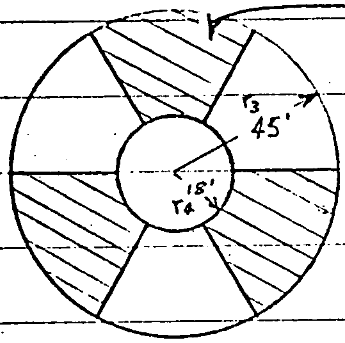
$V_s = 4771 - (924 + 718) + 4200 + 1680$
$= 9009 \text{ ft}^3$

A_1 can be approximated by subtracting the area included in the polar crane wall from the cross sectional area of the containment vessel



$A_1 = \pi r_2^2 - \pi r_1^2$
$= 3.14 (65)^2 - 3.14 (48)^2$
$= 6034 \text{ ft}^2$

A_2 can be approximated as half of the area between the missile shield and the inside of the polar crane wall



A_2 (area at 228' level)

$$A_2 = \frac{1}{2}\left(\pi r_3^2 - \pi r_4^2\right)$$

$$= \frac{1}{2}\left[3.14(45)^2 - 3.14(18)^2\right]$$

$$= 2671\ ft^2$$

$$V_{cv} = V_s + (A_1 + A_2)\,H_w$$

$$= 9009\ ft^3 + (6034 + 2671)\ ft^2\,(1.04\ ft)$$

$$= 18062\ ft^3$$

$$V_{cv} = 18062\ ft^3 \times 7.48\ gal/ft^3$$

$$= 135105\ \text{gallons}$$

Calculation 2

Calculating the Quantity of Water Removed From the Containment Vessel

Trucked Off-Site	27,900 gal
Emptied into A CVCS Holdup Tank (0% to 96%)	49,000 gal
Emptied into Refueling Water Storage Tank (67% to 83%)	54,500 gal
Emptied to Waste Holdup Tank (39.5% to 45%)	1,600 gal
Total Water Removed From Containment	133,000 gallons

Calculation 3

Calculating the Quantity of Water Emptied Into the Containment Vessel

From the Refueling Water Storage Tank (92% to 67%)		86,000 gal
318,000 gal		
-232,000 gal		
86,000 gallons		
From the Boric Acid Blender		23,238 gal
21,986 gal primary water		
+ 1,252 gal boric acid		
23,238 gallons		
From the Reactor Coolant System		19,762 gal
Operating Level Compensated to 200°F=	48,186 gal	
Drain Down Level at 200°F÷	-28,424	
Amount spilled onto floor	19,762 gal	
Total Quantity Spilled		129,000 gallons

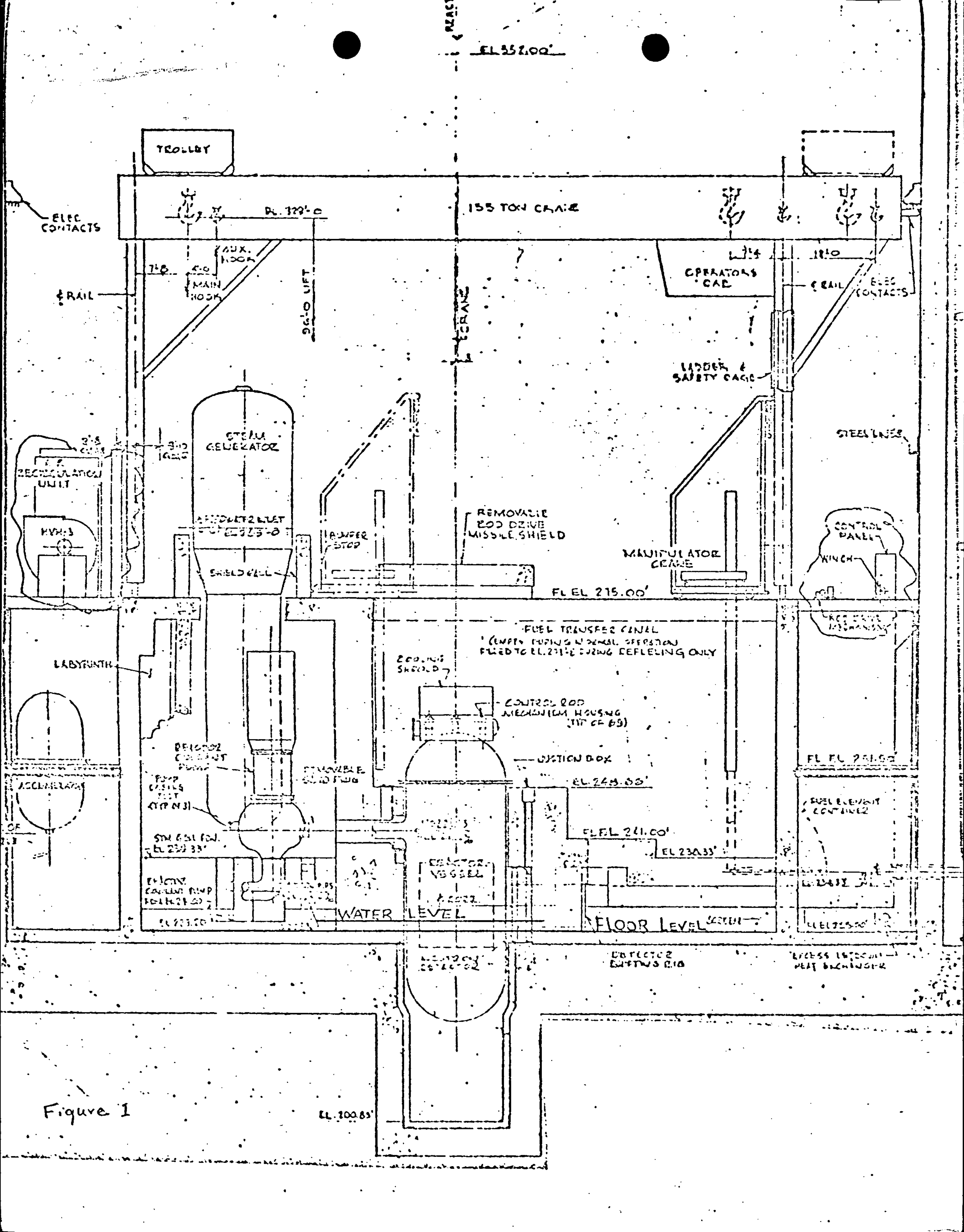
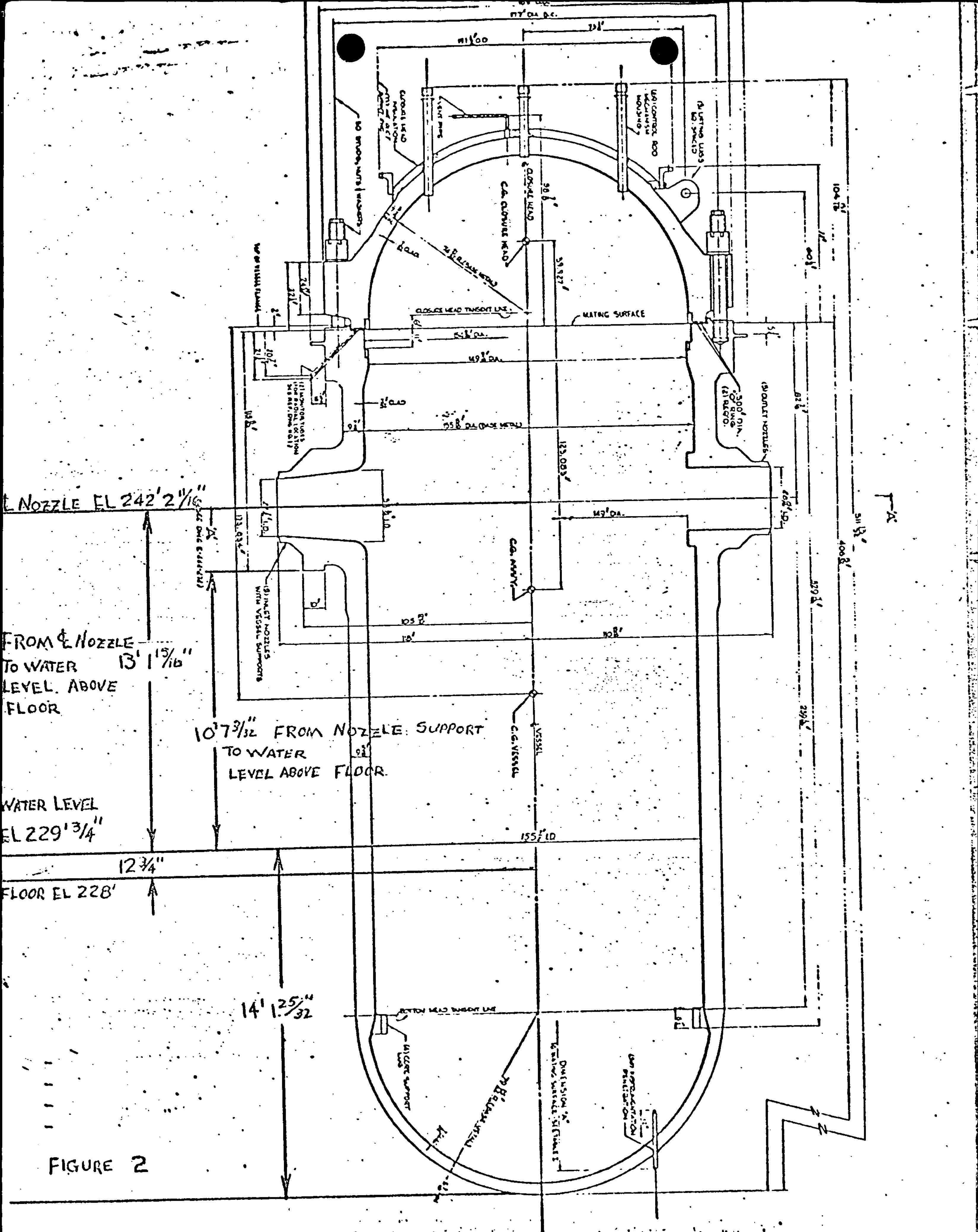

EL 352.00'
TROLLEY
155 TON CRANE
ELEC CONTACTS
AUX. HOOK
MAIN HOOK
96'-0 LIFT
℄ RAIL
℄ CRANE
OPERATORS CAB
LADDER & SAFETY CAGE
STEEL TRUSS
STEAM GENERATOR
REMOVABLE ROD DRIVE MISSILE SHIELD
BUFFER STOP
SHIELD WALL
MANIPULATOR CRANE
CONTROL PANEL
WINCH
FL EL 275.00'
FUEL TRANSFER CANAL
LABYRINTH
CONTROL ROD MECHANISM HOUSING
JUNCTION BOX
REACTOR COOLANT PUMP
REMOVABLE SHIELD PLUG
REACTOR VESSEL
FUEL ELEMENT CONTAINER
FL EL 241.00'
EL 230.33'
WATER LEVEL
FLOOR LEVEL
ACCUMULATOR
DETECTOR SUPPORT RIB
EXCESS LETDOWN HEAT EXCHANGER
EL 200.83'

Figure 1



℄ NOZZLE EL 242' 2 11/16"
FROM ℄ NOZZLE TO WATER LEVEL ABOVE FLOOR
13' 1 15/16"
10' 7 3/32" FROM NOZZLE SUPPORT TO WATER LEVEL ABOVE FLOOR
WATER LEVEL EL 229' 3/4"
12 3/4"
FLOOR EL 228'
14' 1 25/32"
MATING SURFACE
C.G. ASS'Y
C.G. VESSEL

FIGURE 2