NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-220

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NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. George Lear

FROM: Niagara Mohawk Power Corp
Syracuse, N.Y. 13202
Gerald K. Rhode

DATE OF DOCUMENT 03-16-77

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DESCRIPTION

Ltr. Ref. our 03-14-77 phone conversation....Trans The Following:

(1 page)

ENCLOSURE

Consists of Add. Info. supporting th proposed water level change...Reactor Vessel Internal Inspection...Also info. concerning the shielded platform to be used for feedwater nozzle inspection.....

(12 pages)
40 encl rec'd

PLANT NAME: NINE MILE POINT UNIT # 1

jcm

ACKNOWLEDGED

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FOR ACTION/INFORMATION

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NRC FORM 195 (2-76)

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD, WEST
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REGULATORY DOCKET FILE COPY

March 16, 1977

Director of Nuclear Reactor Regulation
Attn: Mr. George Lear, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Gentlemen:

Our submittal of March 8, 1977 requested an amendment to the operating license for Nine Mile Point Unit 1. This amendment would provide for lowering of reactor vessel water level to below the low-low-low level set point in order to perform inspection and/or maintenance on the reactor vessel.

The attached report contains additional information supporting the proposed water level change. Information concerning the shielded platform to be used for feedwater nozzle inspection is also included. This report is being provided in response to requests made by your staff during telephone discussions on March 14, 1977.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

GERALD K. RHODE
Vice President - Engineering

MGM/sz

Attachment

7708103600

REACTOR VESSEL INTERNAL INSPECTION

REACTOR VESSEL INTERNAL INSPECTION

I. INTRODUCTION

In order to perform planned inspections and maintenance on the feedwater spargers and nozzles during the current Spring 1977 outage the vessel water level must be lowered to below that allowed by the Technical Specifications. Future reactor vessel maintenance (i.e. core spray sparger) may also require the lowering of the reactor vessel water level below that presently allowed. Therefore, a change to the Technical Specifications is required. During the inspection and maintenance, a shielded inspection platform will be placed inside the reactor vessel and supported by the shroud.

II. REACTOR VESSEL WATER LEVEL CHANGE

During this refueling outage it is desired to lower the reactor water level to within 50 inches above the top of the active fuel to perform inspection and/or repair work on the feedwater spargers and nozzles.

This corresponds to a level of approximately 8 feet 5 inches below the normal water level at elevation 302 feet 9 inches. It is anticipated that future reactor vessel maintenance may require lowering water level to approximately 8' 9' below normal level. The current Technical Specification limit is 7' 11" below normal level. It is proposed that the Technical Specification limit be changed to 9' below normal level to accommodate current and future maintenance plans.

Since water level will be lowered below the set points for the automatic initiation of the primary coolant and containment isolation, emergency cooling, core spray and high pressure coolant injection, automatic initiation of these systems will be bypassed. However, the core spray system, the only bypassed system which may be required, will be available for manually initiation.

The present monitoring equipment is not capable of indicating levels lower than 7' 11" below normal level. Therefore, it is necessary to have in place additional monitoring equipment while water level is more than 7'11" below normal level. This will assure the active fuel is covered during the proposed work and that rapid water level changes will be detected and core spray will be manually initiated if required. Figures 1 & 2 show the existing and proposed level indicating systems. Figure 3 shows various reactor vessel elevations and their relation to normal water level.

The proposed monitoring installation consists of adding a temporary reference pot and temporary instrument lines to each of the two reactor water level indicating systems. The instrument lines will connect the reference pots to level transmitters 1D13A and 1D13B. Additional instrument lines will connect the level transmitters to an instrument line below the core at differential pressure indicating switch RV-30.

The two level transmitters 1D13A and 1D13B will not require a calibration change or relocation. The output from the level transmitters is connected to existing level meters located on F-Panel in the Control Room. The level meters are readable over the range from zero to 8', with zero being the top of the fuel. A chart recorder which is switchable to either channel will be used to aid in monitoring water level. It will have an annunciator set to alarm if the water level drops to 8' 11" below the normal water level.

The temporary reactor vessel water level monitoring installation will provide redundant instrumentation for monitoring the water level below the low-low-low level set point. With the exception of two manually-operated isolation valves, this system will meet single failure criteria. These two valves will be marked up and locked open with a temporary locking device.

Vessel water level will be maintained by two independent sources. The primary source will be from the demineralized water storage tank through the liquid poison system with the explosive valves' internals removed. The secondary source of temporary make-up to the vessel will be from a hose connected to the condensate transfer system and run into the top of the vessel. The former source is capable of supplying a minimum of 60 gpm, while the later can supply 100 gpm through a 1-1/2" hose. Any drainage required to maintain vessel level will be controlled through remotely operated valves between the clean-up system and the equipment drain tank.

The shutdown cooling system will be available and in service, as required for removing heat from the reactor vessel water.

With the level meters located in the Control Room, the reactor water level is continuously monitored by the Control Room operator. Procedures and/or instructions will be developed for the operator to manually initiate core spray should the water level fall to 9' 3" below the normal water level. This elevation is 3' 4" above the top of the active fuel. Manual initiation of the core spray will be by the normal manual switch located on the Control Room Panel. Manual initiation of the Core spray may be slower than automatic initiation due to the human element lag time. However, this is compensated for by a substantial reduction of heat deposition resulting from a postulated Loss of Coolant Accident(LOCA) while in the shutdown condition. Reductions in heat deposition are due to the following:

1) The lowering of the reactor water level and the by-passing automatic initiation of the core spray will not be performed until approximately twenty days after shutdown at which time the core decay heat is greater than a factor of 10 less than it would be immediately after shutdown.

2) At the time of the core spray by-passing, the stored energy in the core will be insignificant.

Calculations show that after 20 days of shutdown, the decay heat in the core has decreased to a level where if cooling would be provided within 15 minutes after the core is uncovered, the peak clad temperature would be limited to less than 1500F. Once the operator responds to the lowering of the reactor water level and initiates core spray (human response time of turning a switch), the core spray will reach full flow in less than 35 seconds, as indicated in Addendum 5 of the Nine Mile Point Unit 1 FSAR.

Operability requirements for core spray will be such that after considering the worst single failure or operator error, at least one core spray system will be available to deliver rated flow during the proposed maintenance.

Furthermore, the probability of having a LOCA or other accident requiring the core spray system is significantly lower during shutdown and refueling when the reactor is at ambient temperatures and pressures.

Finally, no other work that has the potential of decreasing the reactor vessel water level will be performed while the reactor vessel water level is below the low-low-low level set point.

It is anticipated that this work or similar maintenance in the future, while requires lowering the reactor water level below the low-low-low level set point, will be completed within six weeks.

III. Shielded Inspection Platform

The Shielded Inspection Platform was designed under the restraints of a formal "Design Control Program" meeting the requirements of Criterion III of 10CFR50 Appendix B. This Program was imposed to assure that the platform structure and related components would perform under the intended design load combinations.

Design criteria for safe handling and installation of the platform require the structure to sustain primary stresses (service loads and dead loads) combined with secondary stresses due to temperature and seismic acceleration. The resultant cumulative stresses are within working limits as determined by the American Institute of Steel Construction Code, latest edition.

Design test loading required that the structure and related lifting tools be load tested with an equivalent uniform load of 1.25 times the sum of the dead and live loads.

Material procurement, fabrication, assembly and test acceptance were all monitored and controlled under a formal Quality Control program subject to audit by Niagara Mohawk Power Corporation.

The platform structure rests directly on the core shroud land area requires no vertical restraint from the core shroud due to the high dead load (28 tons). This core shroud normally supports both the steam dryer and separator assemblies which have a combined total dead load of 77 tons. Horizontal restraint is provided by two 2.75 in. pins which secure the platform to the core shroud through the guide pin holes normally used for the steam separator. See Figure 4 attached.

A. Safety Factors

The 125 ton reactor building crane and reactor head lifting assembly will be used to lift the shielded inspection platform into position inside of the reactor vessel. The crane is used during refueling to lift the reactor cavity shield plugs (122 tons), reactor head (80 tons) (with lifting assembly) and steam separator assembly (39 tons). The shielded platform and lifting tool (30.5 tons) is less than any of the above-mentioned items.

The following tabulation indicates the calculated stress levels and associated safety factors based on minimum yield. The safety factors are addressed for all principle load bearing members and connections

which are directly related to the safe handling and installation of the platform. It should be noted that this analysis considers the maximum stress developed in either the suspended or inplace modes. See Figure 5 attached for the lifting arrangement.

1) <u>125 Ton Reactor Building Crane</u>:

Rated Load:	125	tons
Test Loaded:	156.3	tons
Safety Factor (30.5 tons) (based on rated load)	4.1	

2) <u>Reactor Head Lifting Assembly</u>:

Rated Load:	100	tons
Test Loaded:	127	tons
Safety Factor (based on minimum yield strength)	9.84	

Note: This assembly interfaces with the 125 ton crane hook via three 6 7/8" diameter pins.

3) <u>Platform Lifting Tool</u>:

Rated Load:	28	tons
Dead Load:	2.5	tons
Test Loaded: (125% Dead Load and Live Load)	36	tons
Maximum Stress: (occurs in primary strong back sections)	(Bending)	
Minimum Yield Strength:	36	KIPS
Safety Factor (based on minimum yield strength)	6.10	

NOTE: This assembly interfaces with the bottom of the reactor head lifting assembly via three pin connected lugs which mate with the above assembly.

4) <u>Shielded Inspection Platform:</u>

Dead Load:	28	tons
Distributed Live Load (Design)	100#/ft^2	
Total Rated Capacity	9.74	tons
Test Loaded: (125% Dead Load and Live Load)	47.18	tons
Minimum Yield Strength:	36	KIPS

Stresses in Primary Members - "Suspended"

a). Main Deck members — 4.25 KSI (combined stresses)

Safety Factor (based on minimum yield strength — 8.47

b) Circular Channel Sections: (periphery of platform) — 1.86 KSI (Bending), 0.24 KSI (Shear)

Safety Factors (Bending) (based on minimum yield strength) — <u>19.35</u>

c) Recessed Circular Lifting Lugs to Main Deck Members (4) Weldments (F_y = 45 KSI weldmetal) — 2.7 KSI (tensile)

(based on A-36 base metal) — <u>13.3</u>

d) 4-ATSM A307-2-1/2" Lifting Studs — 3.93 KSI (tensile)

(based on tensile strength of 55 kips) — <u>14</u>

Stresses in Main Deck Members - "In Place"

5.97 KSI (Combined)

Safety Factor (based on yield strength - 36 KIPS) — <u>6.03</u>

EXISTING LEVEL INDICATING SYSTEM

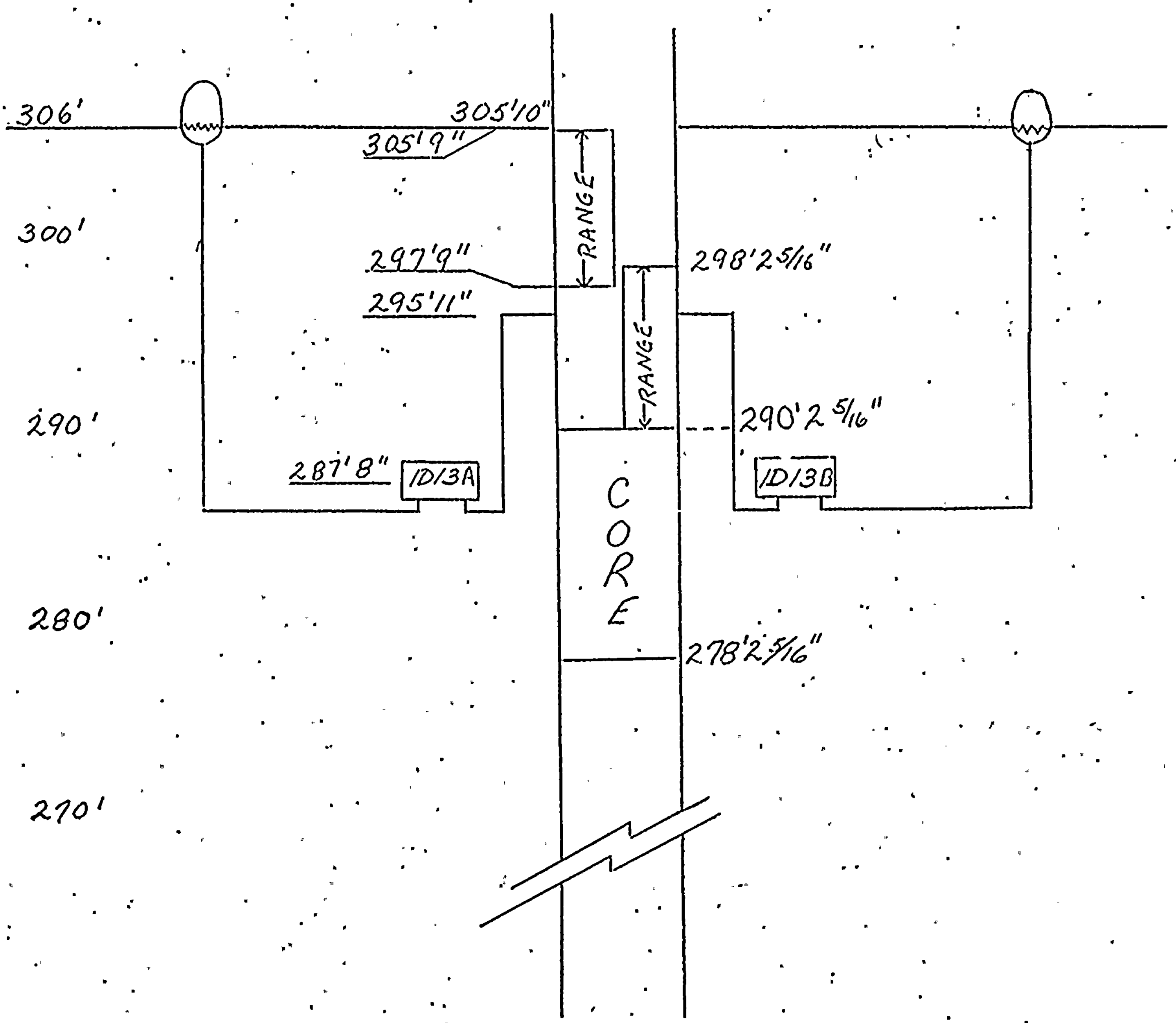


FIGURE 1

PROPOSED LEVEL INDICATING SYSTEM

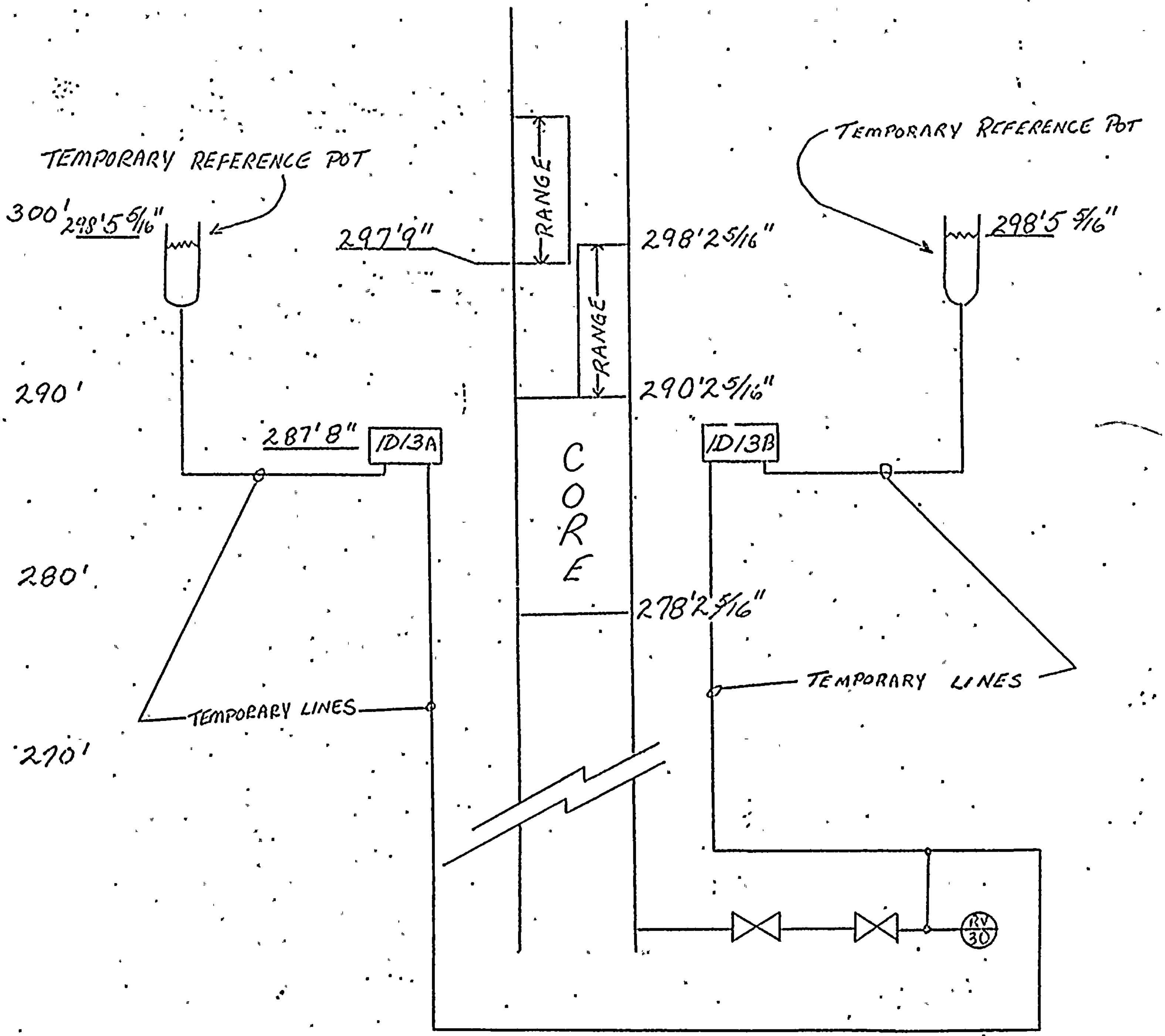


FIGURE 2

Normal Water Elevation	302'9"	
Low Water Level Set Point	301'9"	(1' below normal water level)
Low Low Water Level Set Point	297'9"	(5' below normal water level)
Low Low Low Water Level Set Point	294'10"	(7'11" below normal water level)

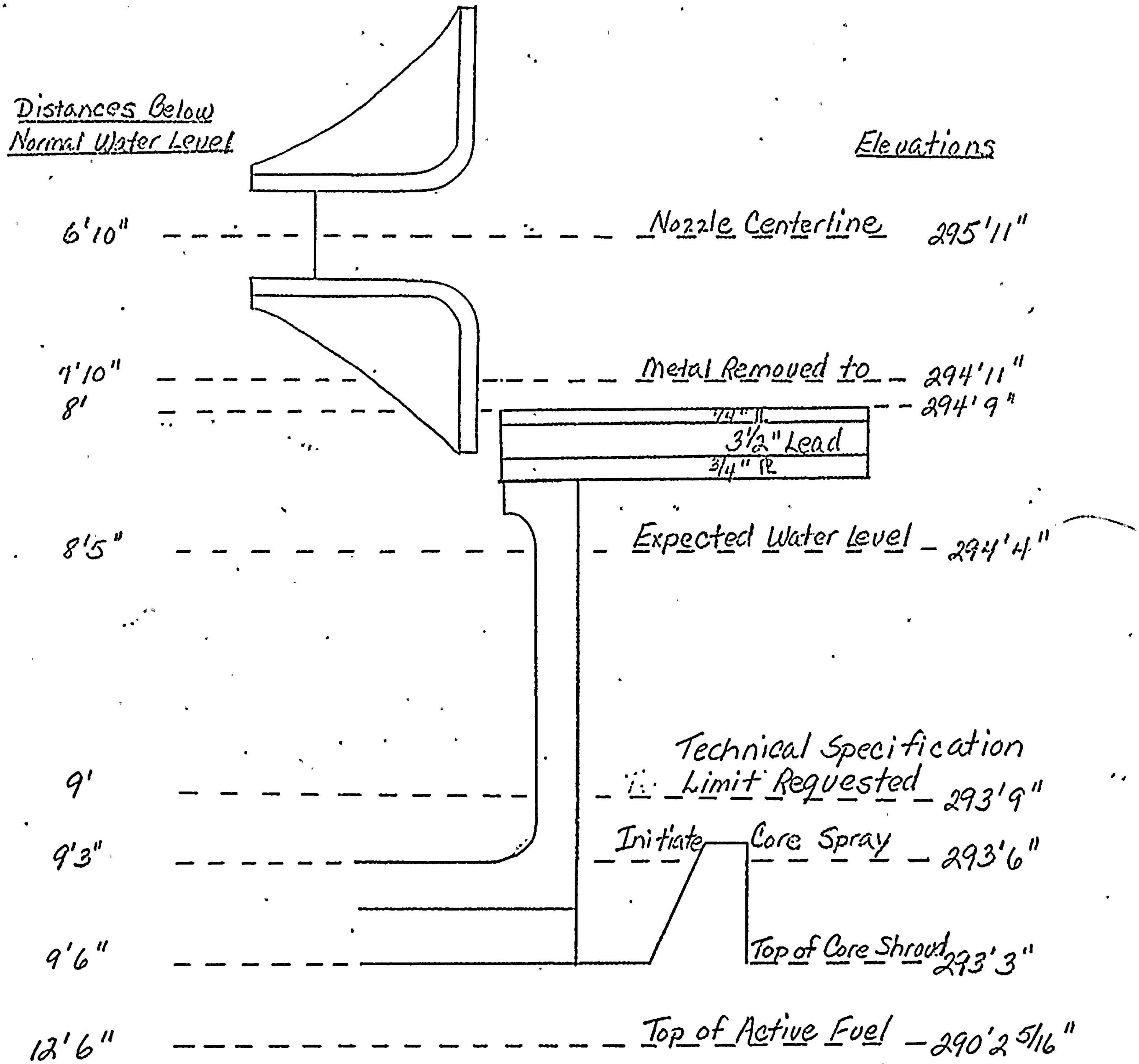
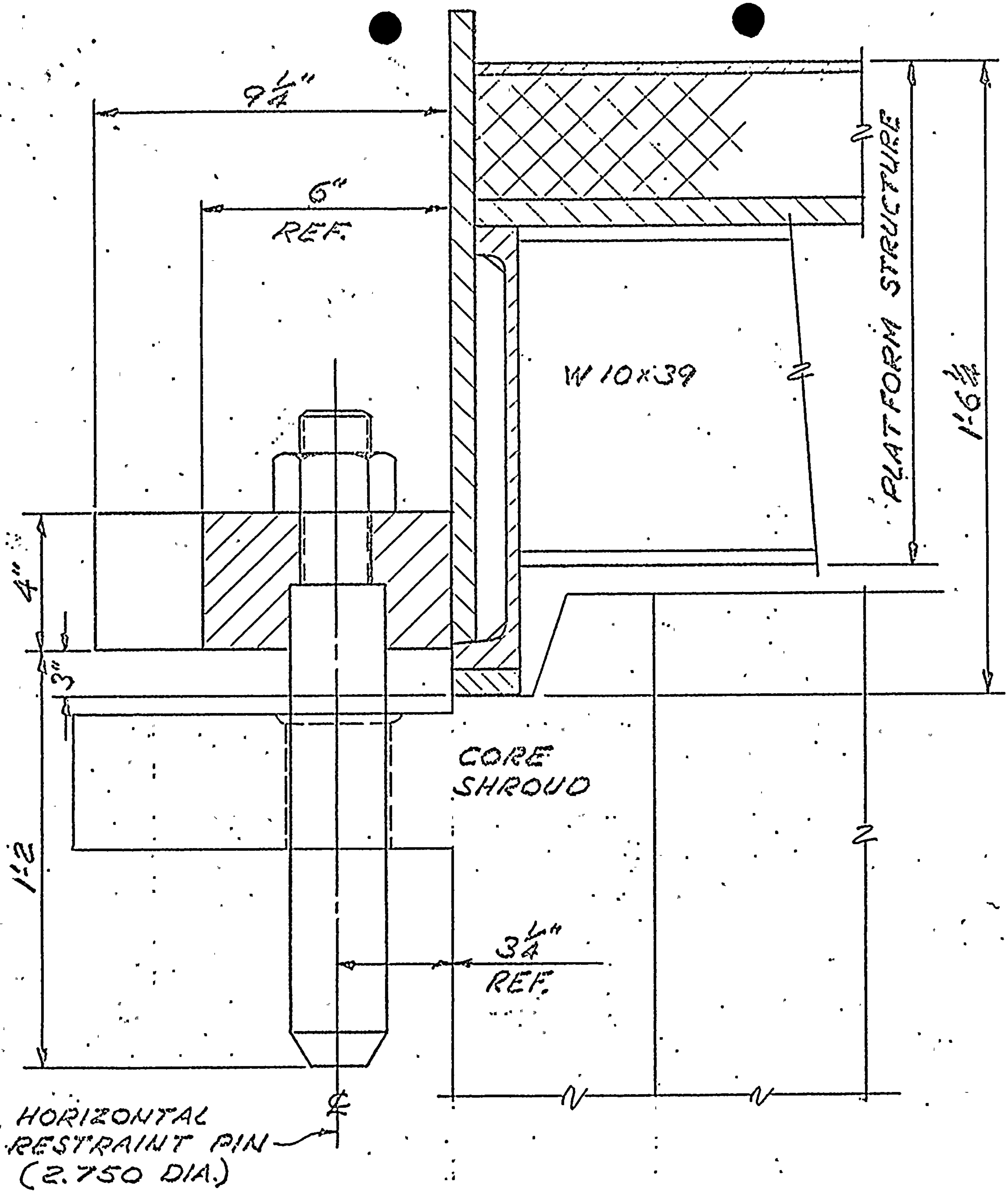
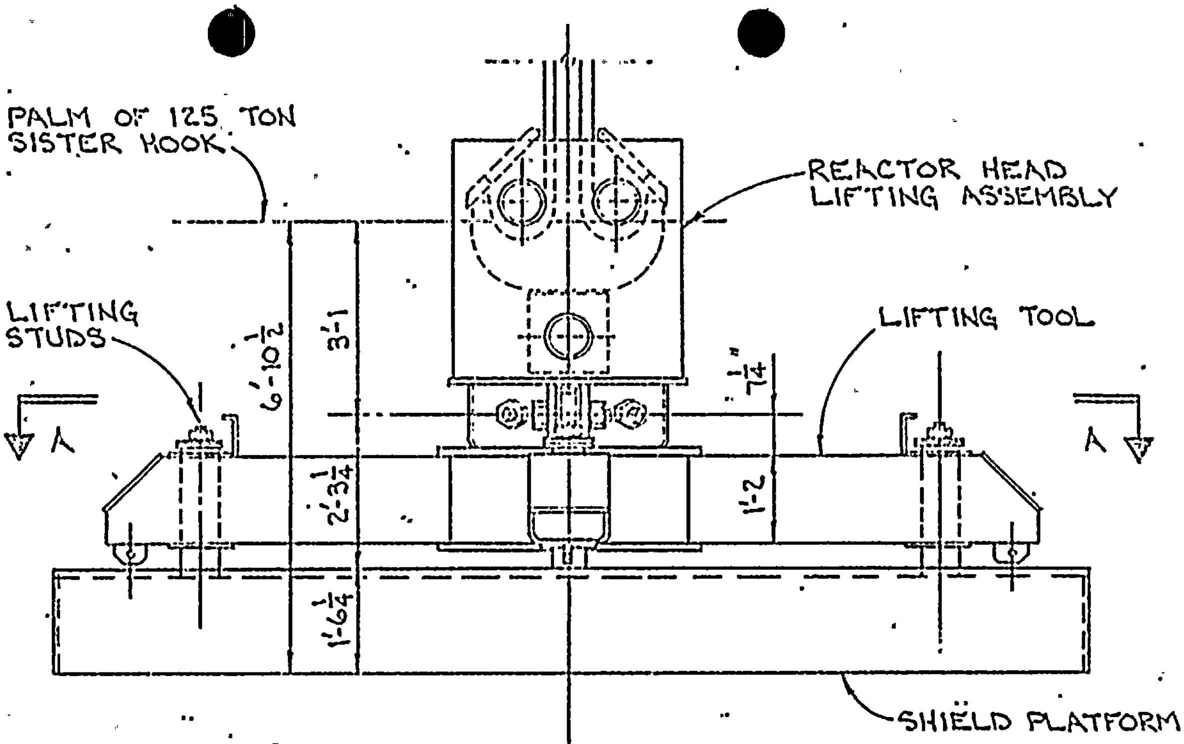


FIGURE 3.

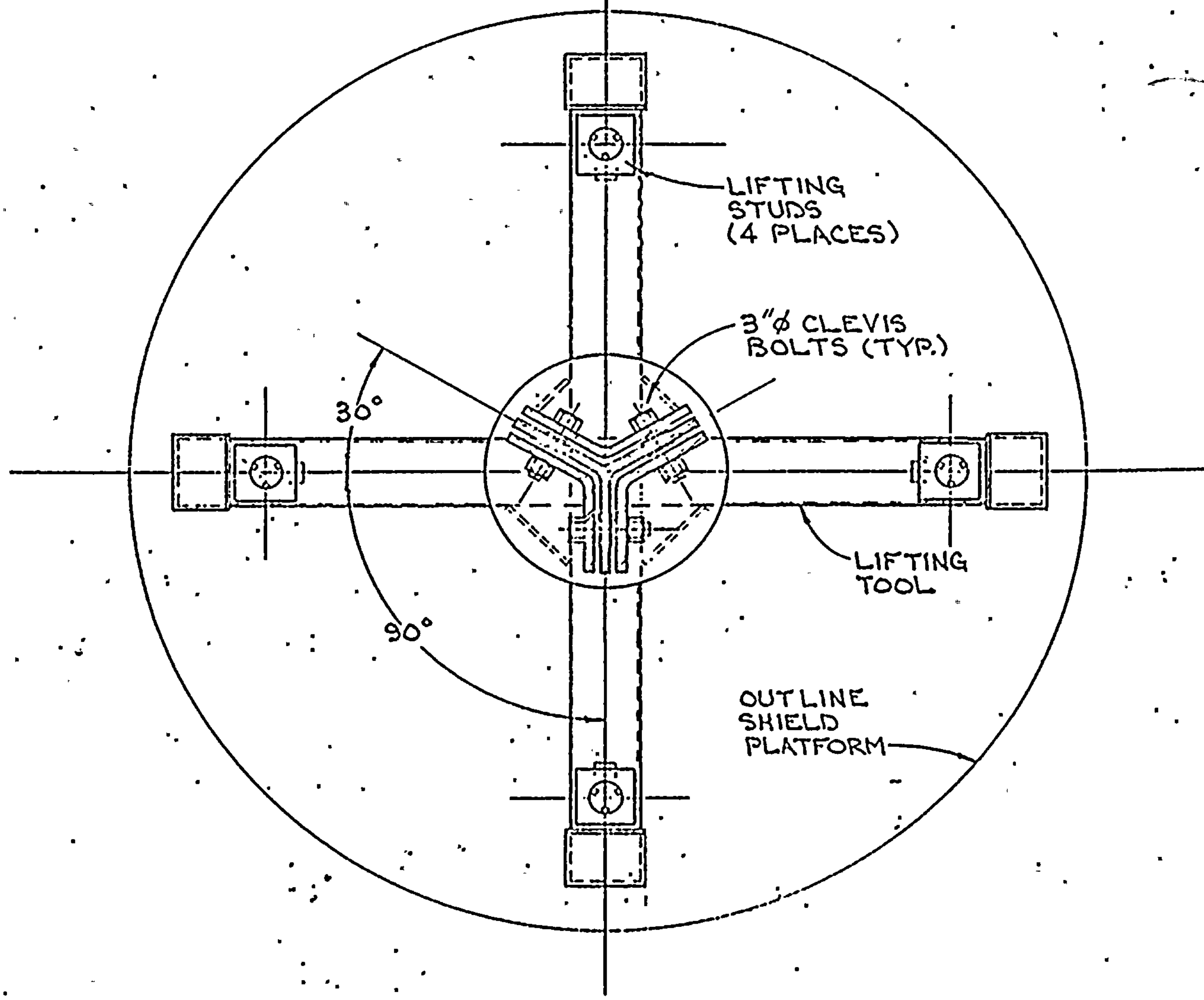


DETAIL SHOWS PLATFORM IN PLACE ON REACTOR CORE SHROUD. HORIZONTAL RESTRAINT PIN IS FOR SEISMIC FORCES ONLY.

FIGURE 4



ELEVATION - PLATFORM LIFTING ARRANGEMENT



PLAN - SECTION A-A

FIGURE No. 5