Nuclear

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February 18, 1983
4410-83-L-0037

TMI Program Office
Attn: Mr. L. H. Barrett, Deputy Program Director
US Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, PA 17057-0191

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Polar Crane Refurbishment

Enclosed for your approval is the Safety Evaluation Report (SER) for the Polar Crane Load Test. This document concludes that the load test does not constitute an Unreviewed Safety Question and can be accomplished without undue risk to the health and safety of the public. Your cooperation in expediting the review and approval of this document in a timely manner is greatly appreciated.

If you have any questions or desire further information, please feel free to contact Mr. J. J. Byrne of my staff.

Sincerely,

B. K. Kanga

B. K. Kanga
Director, TMI-2

BKK/RBS/jep

Attachment

CC: Dr. B. J. Sndyer, Program Director - TMI Program Office

D009

1983 FEB 18 PM 3 51
REGULATORY COMMISSION
U.S. NUCLEAR

8302240260 830218
PDR ADOCK 05000320
P PDR

Safety Evaluation Report

for the

Polar Crane Load Test

February 1983

Three Mile Island

Unit 2

REVISION 0
FEBRUARY 14, 1983

8302240263 830218
PDR ADOCK 05000320
P PDR

2.0 MAJOR ACTIVITIES

The polar crane load test may be characterized by four sequential major activities. Each of these activities is discussed below.

2.1 RELOCATING INTERNALS INDEXING FIXTURE

In order to perform the movements associated with the polar crane load test, the Internals Indexing Fixture must first be relocated out of the area to be used for the load test. Prior to moving any of the missile shields, the Internals Indexing Fixture will be moved from its storage location on El. 347' and stored on the Head Storage Stand. Please see Figure 2.1-1.

2.2 ASSEMBLY OF TEST LOAD

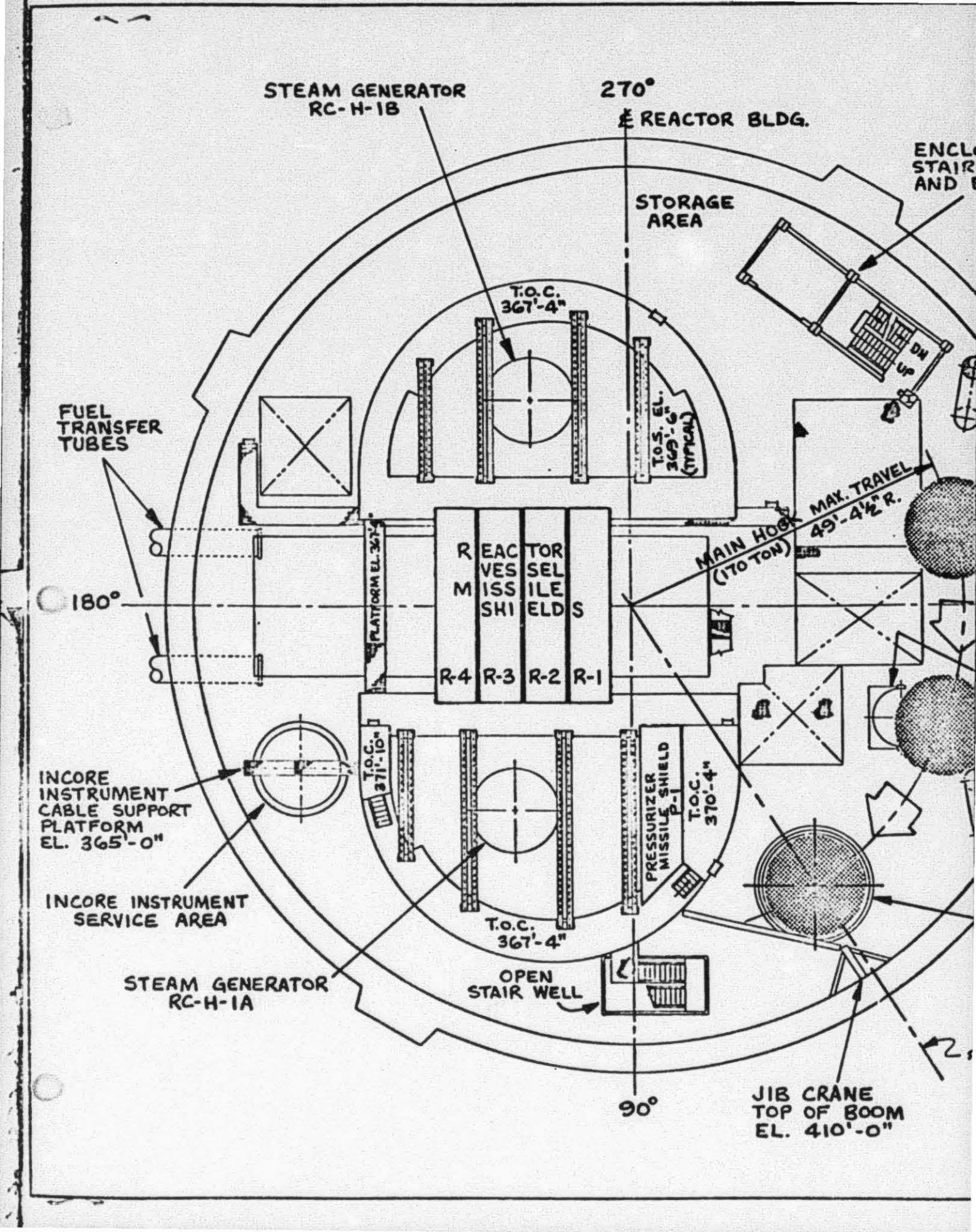
The test load will consist of a load test frame, the missile shields from over the reactor vessel and the pressurizer, the reactor vessel head lift rigging, and a load indicating device. This assembly is shown in Figure 3.3-1. The load test frame will be constructed of steel members brought into containment through the airlock doors in the equipment hatch. These members will then be moved to the floor hatch and lifted up to El. 347', where the load test frame will be assembled as shown in Figure 2.2-1. The test load will then be assembled by moving the reactor vessel missile shields from over the reactor vessel and stacking them on the load test frame as shown in Figure 2.2-2. Finally, the pressurizer missile shield will be moved from over the pressurizer and stacked on the load test frame as shown in Figure 2.2-3, and the lift rigging attached.

2.3 LOAD TEST

The actual load test is presented in step-by-step detail in the test procedure covered by reference b. This procedure will be approved by the USNRC prior to implementation. Basically, the test load will be lifted so it is supported by the crane and held by the hoist brakes. The load will then be transported by the trolley and bridge a distance sufficient to allow at least one full revolution of the trolley and bridge gearing. The bridge will then be rotated 180° and the aforementioned steps repeated. This evolution is shown in Figure 2.3-1.

2.4 DISASSEMBLY OF TEST LOAD

After completion of the load test, the test load will be disassembled. This will be accomplished by removing the load test rigging and then moving the missile shields to their storage location. The pressurizer missile shield will be moved back to its storage location over the pressurizer as shown in Figure 2.4-1. The reactor vessel missile shields will then be moved from the load test frame to their storage location on the B D-ring as shown on Figures 2.4-2 and 2.4-3. Prior to moving the reactor vessel missile shields, spacers will be set on the B D-ring walls to avoid interference of the missile shields with existing components. The missile shields will be stacked two high with spacers between them. Finally the load test frame will be disassembled.


STEAM GENERATOR RC-H-1B
270°
℄ REACTOR BLDG.
ENCL
STAIR
AND
STORAGE AREA
T.O.C. 367'-4"
T.O.S. EL. 369'-6" (TYPICAL)
DN
UP
FUEL TRANSFER TUBES
MAIN HOOK (170 TON)
MAX. TRAVEL 49'-4½" R.
180°
REACTOR VESSEL MISSILE SHIELDS
PLATFORM EL. 367'-6"
R-4
R-3
R-2
R-1
T.O.C. 371'-10"
PRESSURIZER MISSILE SHIELD P-1
T.O.C. 370'-4"
INCORE INSTRUMENT CABLE SUPPORT PLATFORM EL. 365'-0"
INCORE INSTRUMENT SERVICE AREA
T.O.C. 367'-4"
STEAM GENERATOR RC-H-1A
OPEN STAIR WELL
90°
JIB CRANE TOP OF BOOM EL. 410'-0"

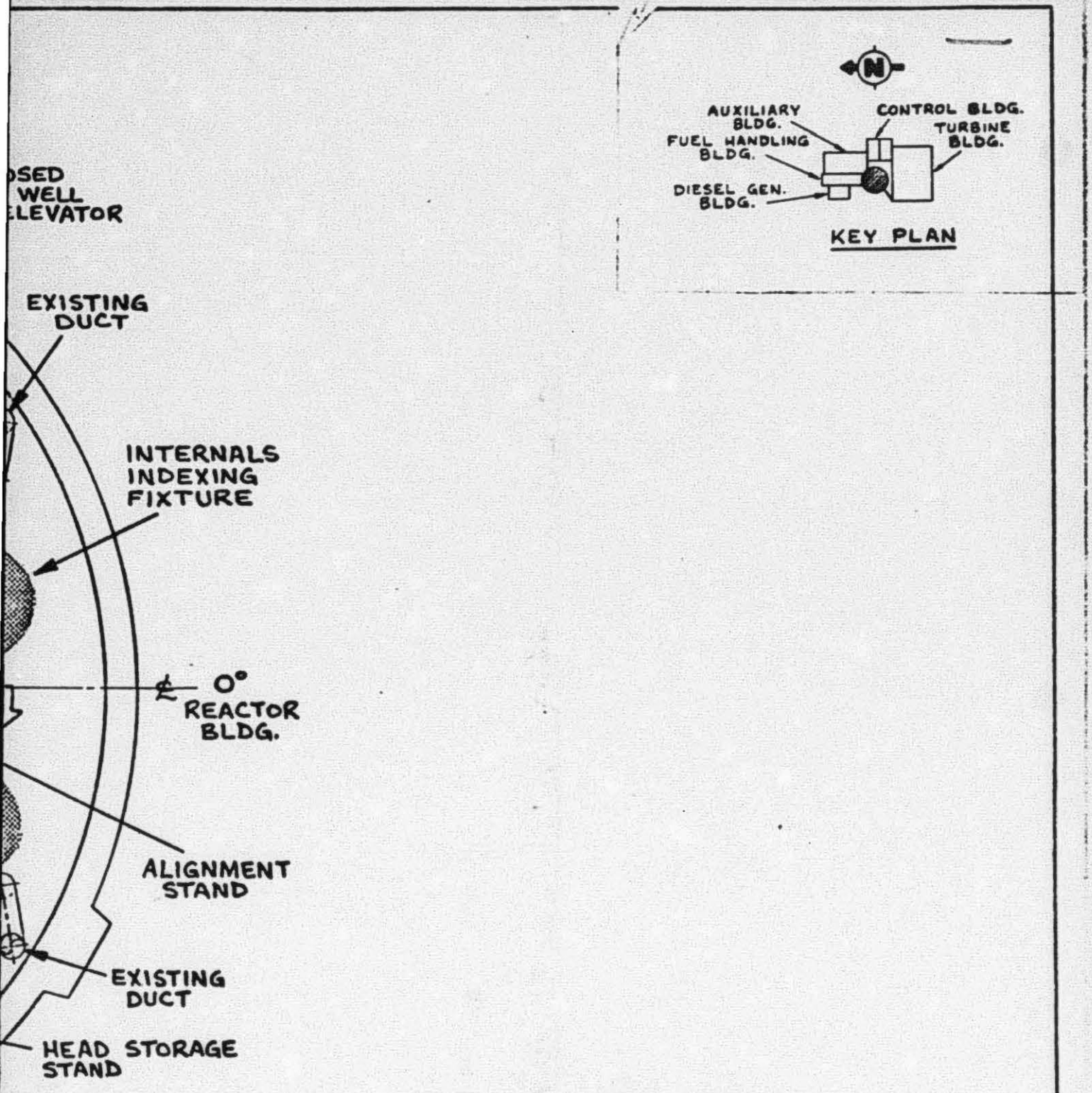
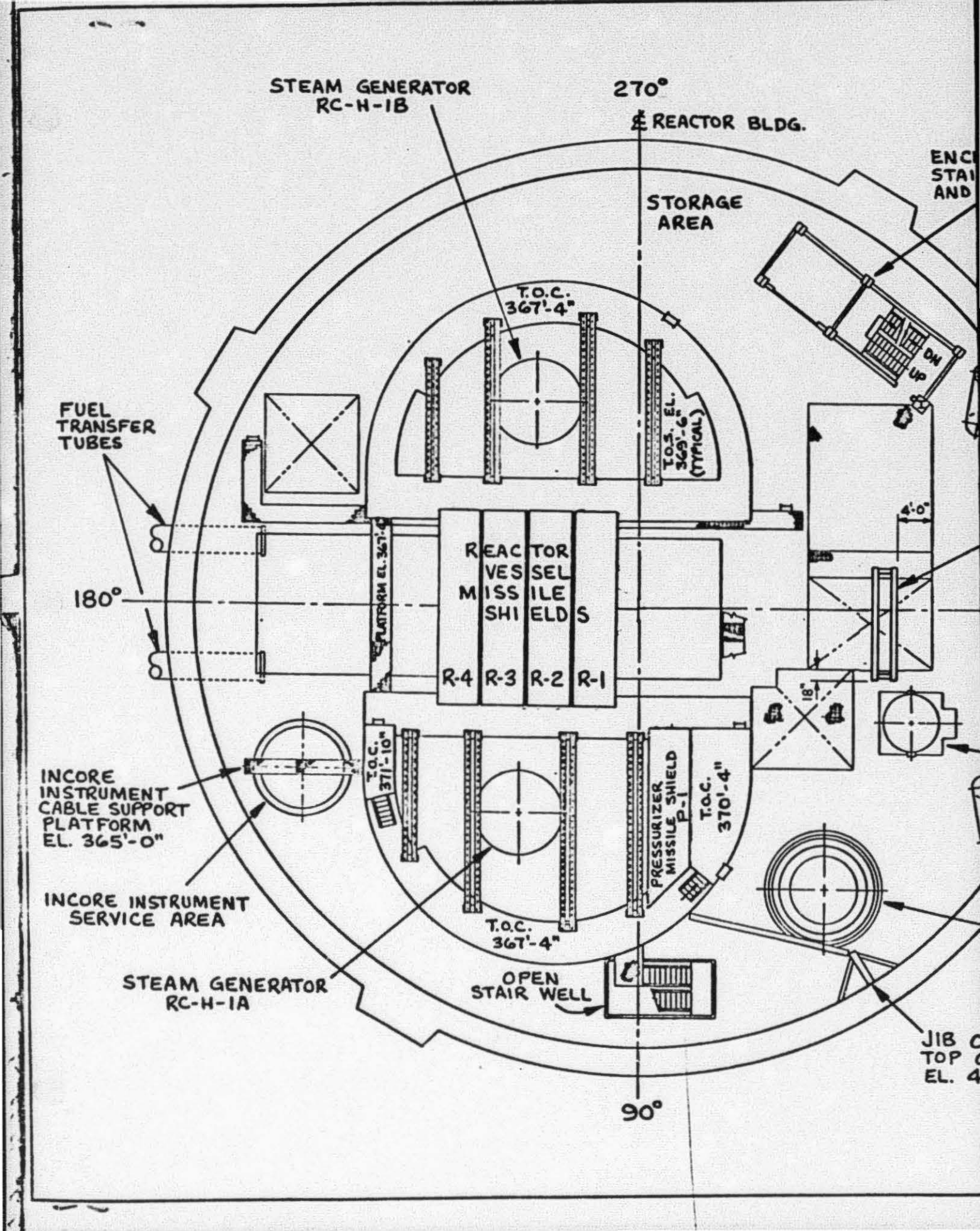


FIGURE 2.1-1

MOVEMENT PATH OF INTERNALS INDEXING FIXTURE


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
STORAGE
AREA
T.O.C.
367'-4"
T.O.S. EL.
369'-6"
(TYPICAL)
FUEL
TRANSFER
TUBES
180°
REACTOR
VESSEL
MISSILE
SHIELDS
R-4
R-3
R-2
R-1
PLATFORM EL. 367'-4"
4'-0"
18"
T.O.C.
371'-10"
PRESSURIZER
MISSILE SHIELD
P-1
T.O.C.
370'-4"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
90°
DN
UP

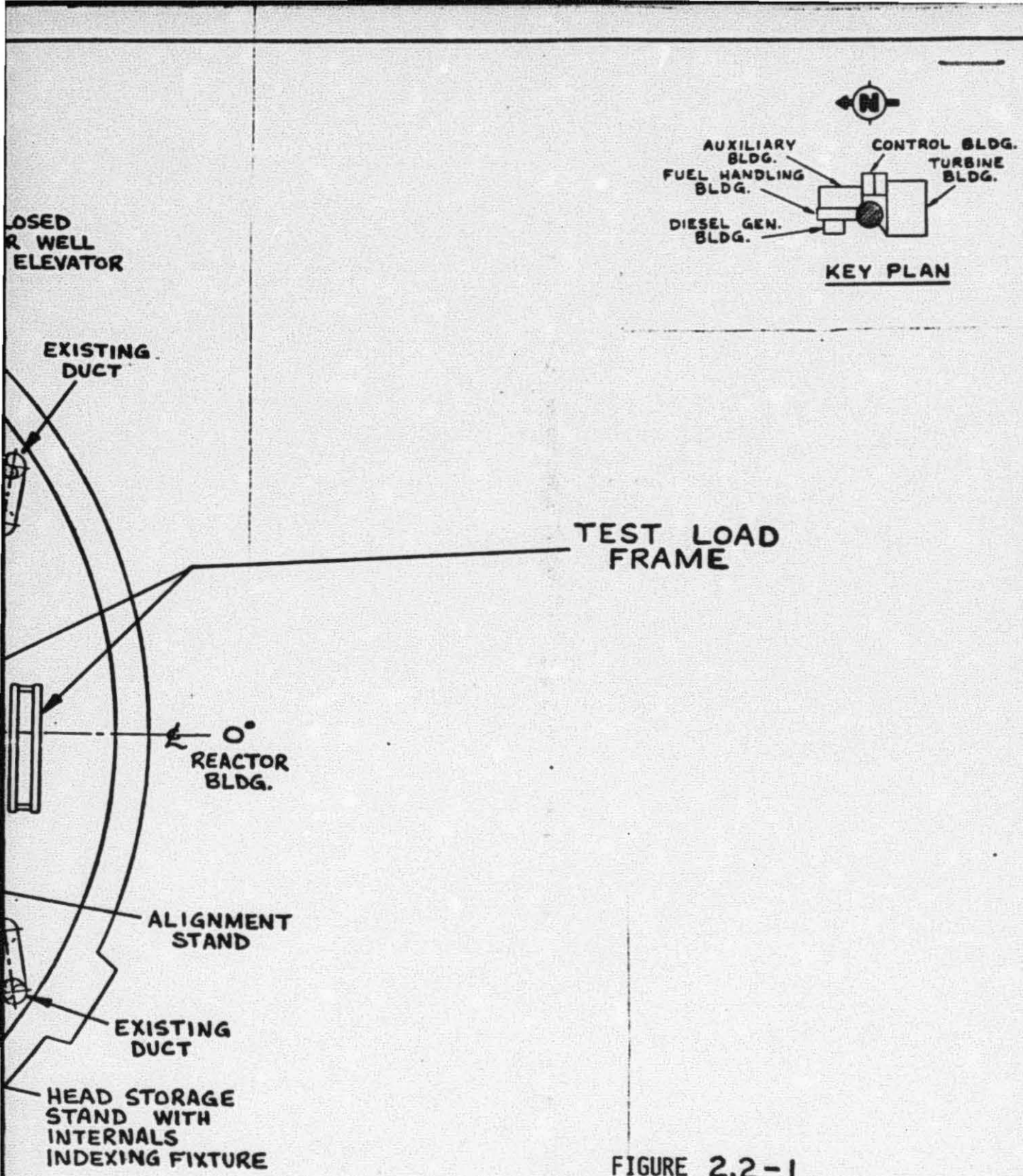
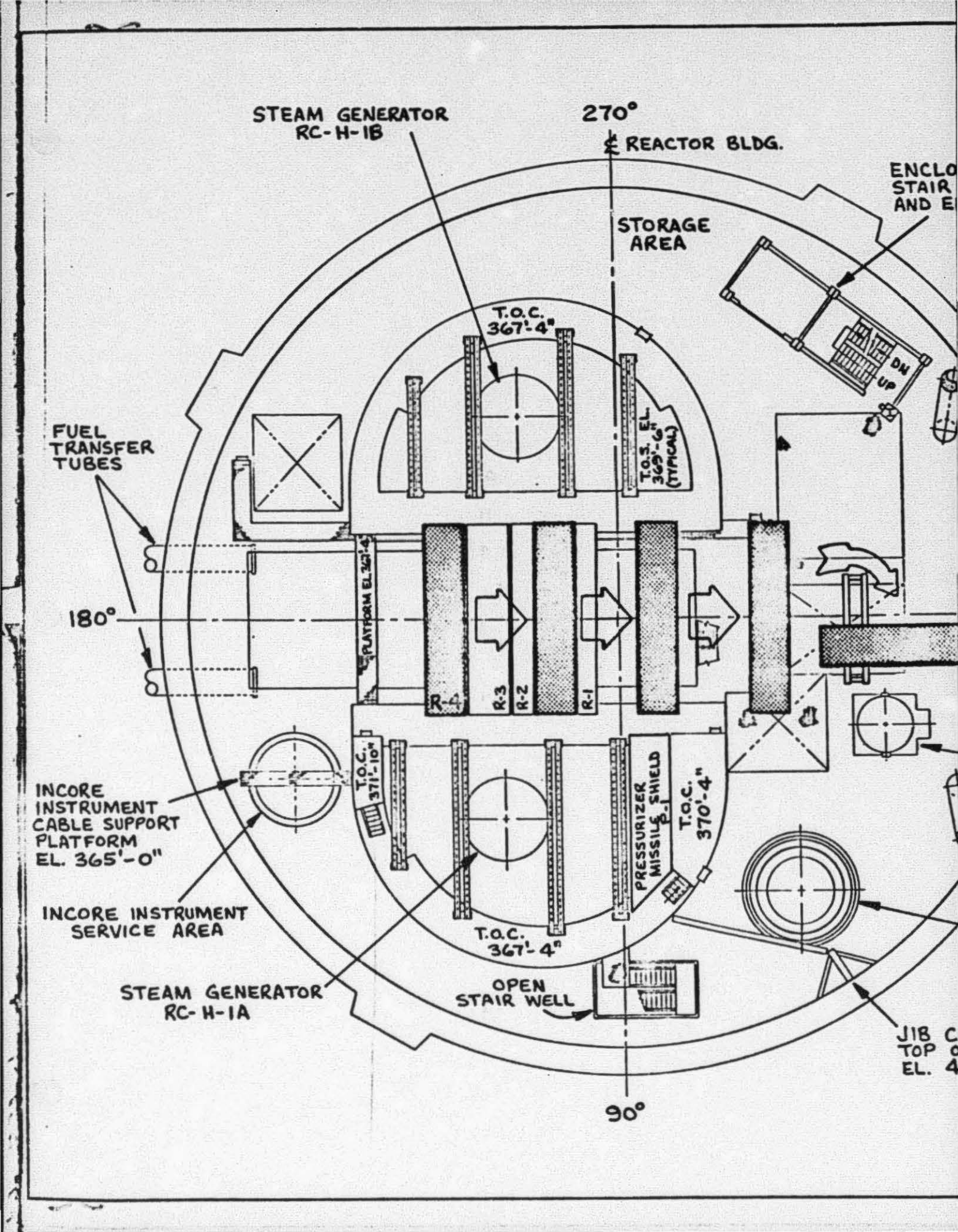
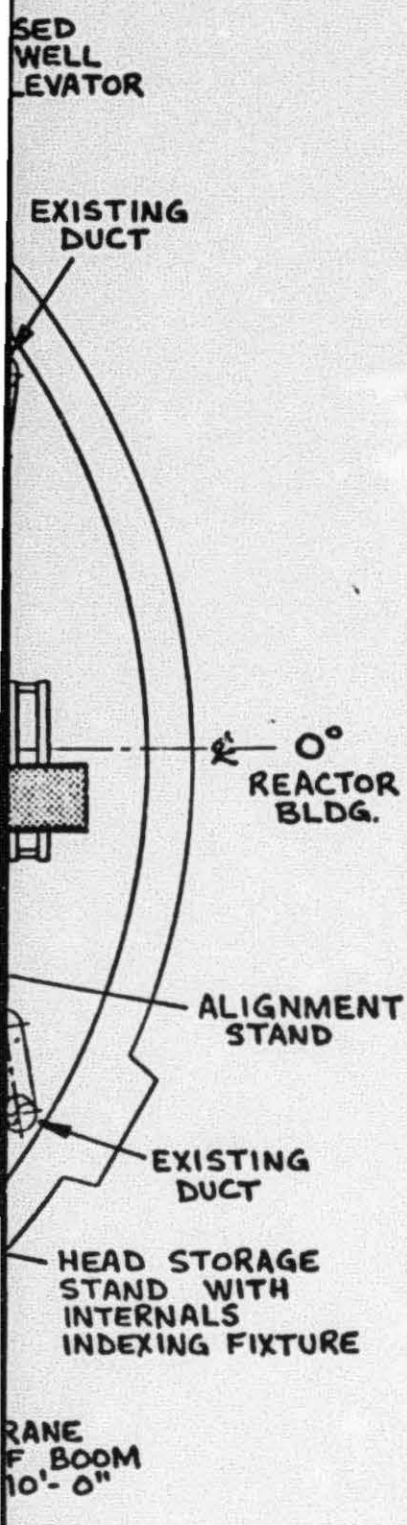
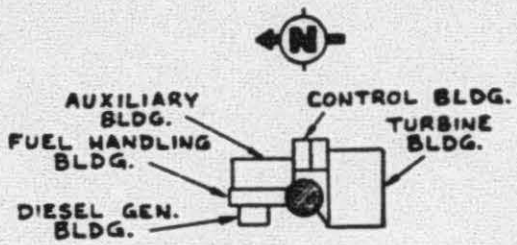


FIGURE 2.2-1

PLACEMENT OF TEST LOAD FRAME


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
ENCLO
STAIR
AND E
STORAGE
AREA
T.O.C.
367'-4"
T.O.S. EL.
369'-6"
(TYPICAL)
DN
UP
FUEL
TRANSFER
TUBES
180°
PLATFORM EL. 367'-4"
R-4
R-3
R-2
R-1
T.O.C.
371'-10"
PRESSURIZER
MISSILE SHIELD
T.O.C.
370'-4"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
J1B C
TOP
EL. 4
90°



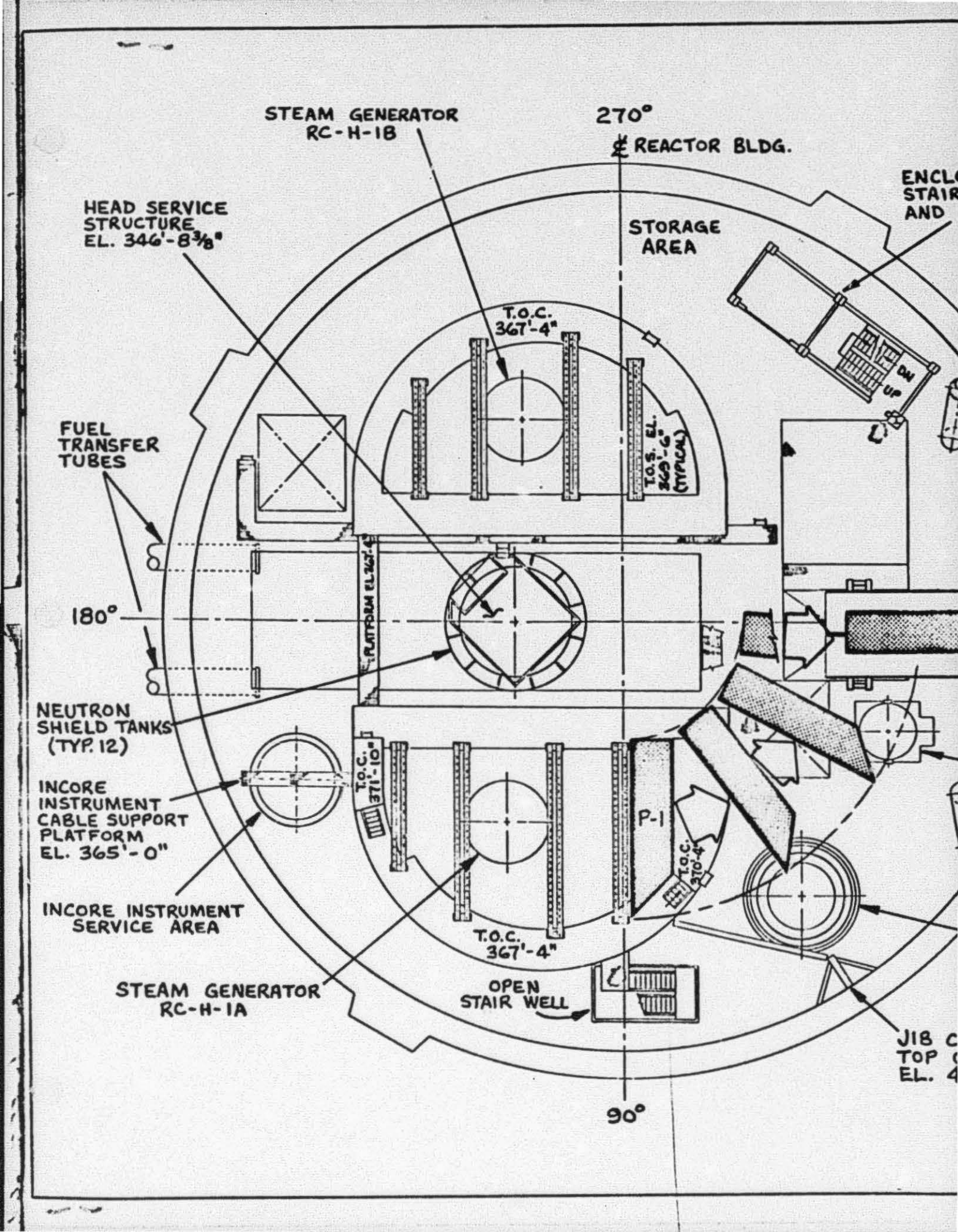


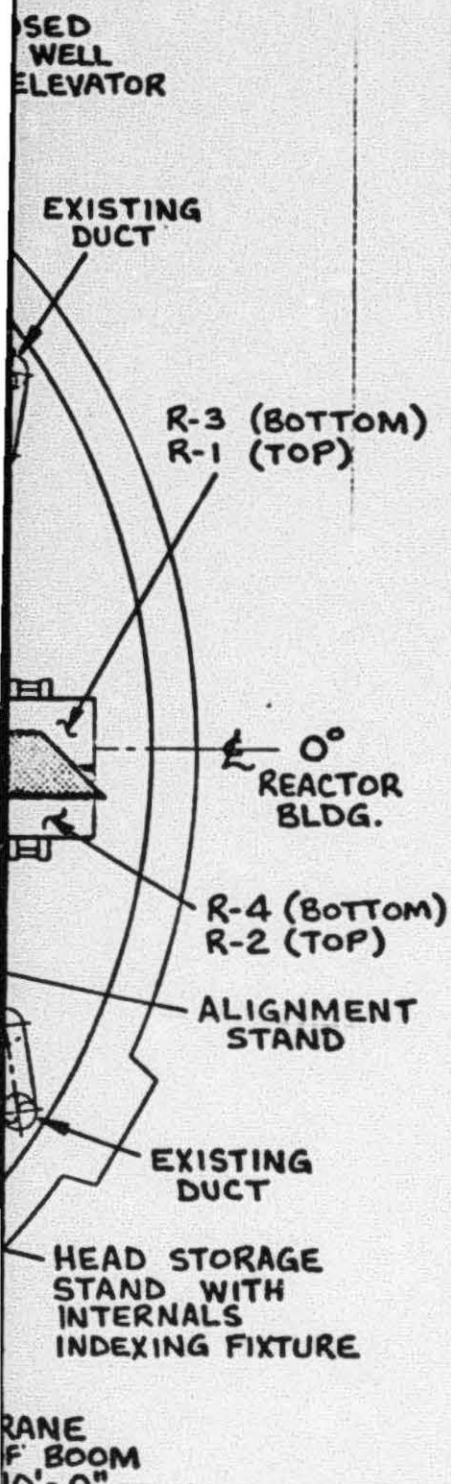
KEY PLAN

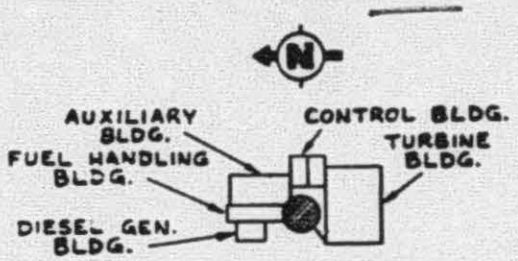
NOTE: R-4, WHICH LIES FARTHEST FROM TEST FRAME, IS MOVED UP AND OVER OTHER 3 SHIELD BLOCKS. THEN R-3, THE NEXT FARTHEST FROM TEST FRAME, IS MOVED OVER REMAINING 2 SHIELD BLOCKS. AFTER THAT, R-2 IS MOVED OVER R-1 AND FINALLY R-1 IS MOVED. THIS SEQUENCE PROVIDES MAXIMUM PROTECTION TO REACTOR VESSEL IN CASE ANY OF THE SHIELD BLOCKS WERE DROPPED

FIGURE 2.2-2

MOVEMENT PATH OF REACTOR MISSILE SHIELD BLOCK R-4


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
ENCL
STAIR
AND
HEAD SERVICE
STRUCTURE
EL. 346'-8 3/8"
STORAGE
AREA
T.O.C.
367'-4"
T.O.S. EL.
365'-6"
(TYPICAL)
DN
UP
FUEL
TRANSFER
TUBES
180°
PLATFORM EL 367'-4"
NEUTRON
SHIELD TANKS
(TYP. 12)
T.O.C.
371'-10"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
P-1
T.O.C.
370'-4"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
J18 C
TOP
EL. 4
90°

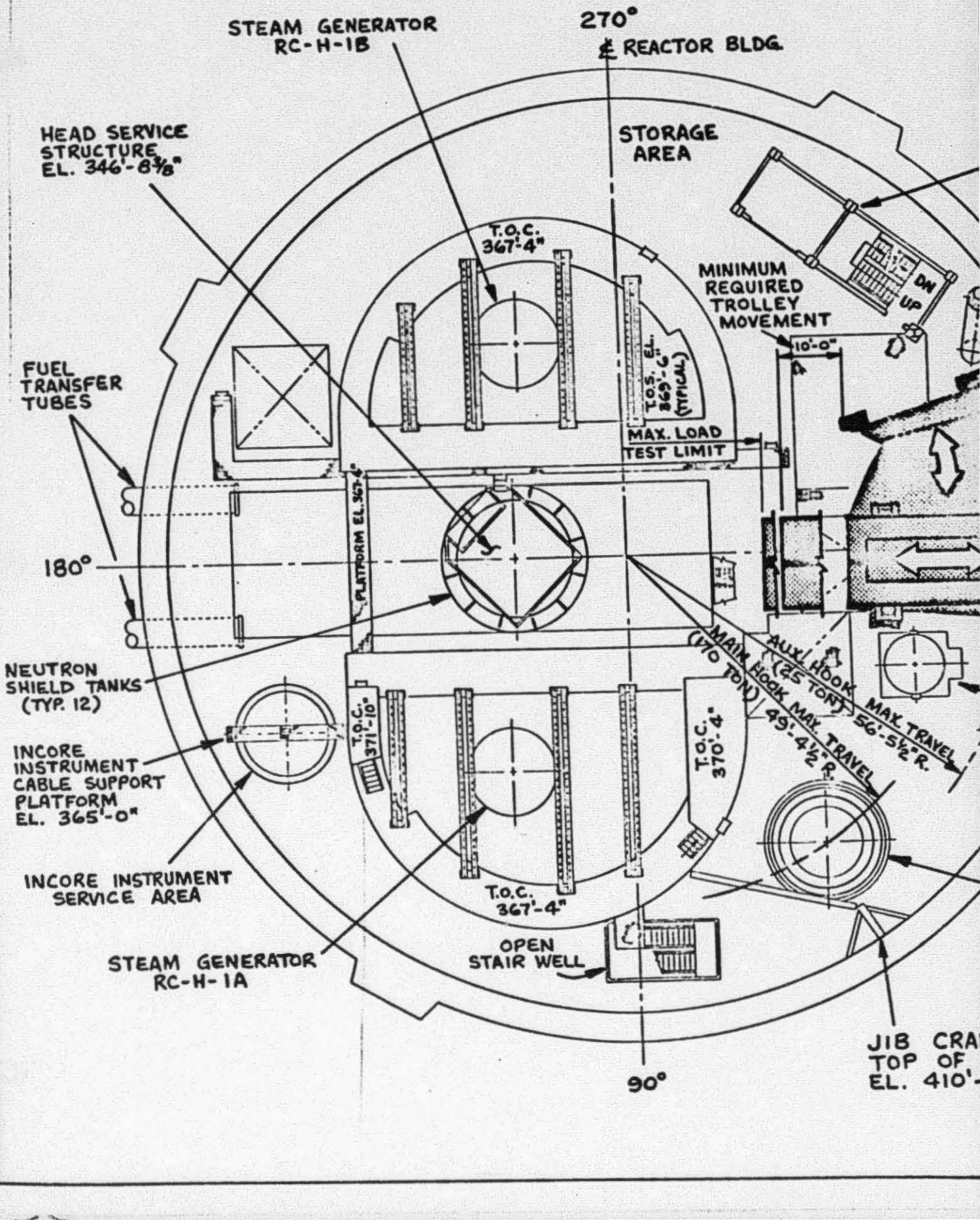




KEY PLAN

FIGURE 2.2-3

PATH OF PRESSURIZER MISSILE SHIELD BLOCK P-1


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
HEAD SERVICE
STRUCTURE
EL. 346'-8 3/8"
STORAGE
AREA
T.O.C.
367'-4"
MINIMUM
REQUIRED
TROLLEY
MOVEMENT
DN
UP
10'-0"
FUEL
TRANSFER
TUBES
T.O.S. EL.
369'-6"
(TYPICAL)
MAX. LOAD
TEST LIMIT
PLATFORM EL. 367'-4"
180°
NEUTRON
SHIELD TANKS
(TYP. 12)
AUX. HOOK
(25 TON)
MAX. TRAVEL
56'-5½" R.
MAIN HOOK
(170 TON)
MAX. TRAVEL
49'-4½" R.
T.O.C.
371'-10"
T.O.C.
370'-4"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
OPEN
STAIR WELL
STEAM GENERATOR
RC-H-1A
90°
JIB CRA
TOP OF
EL. 410'-

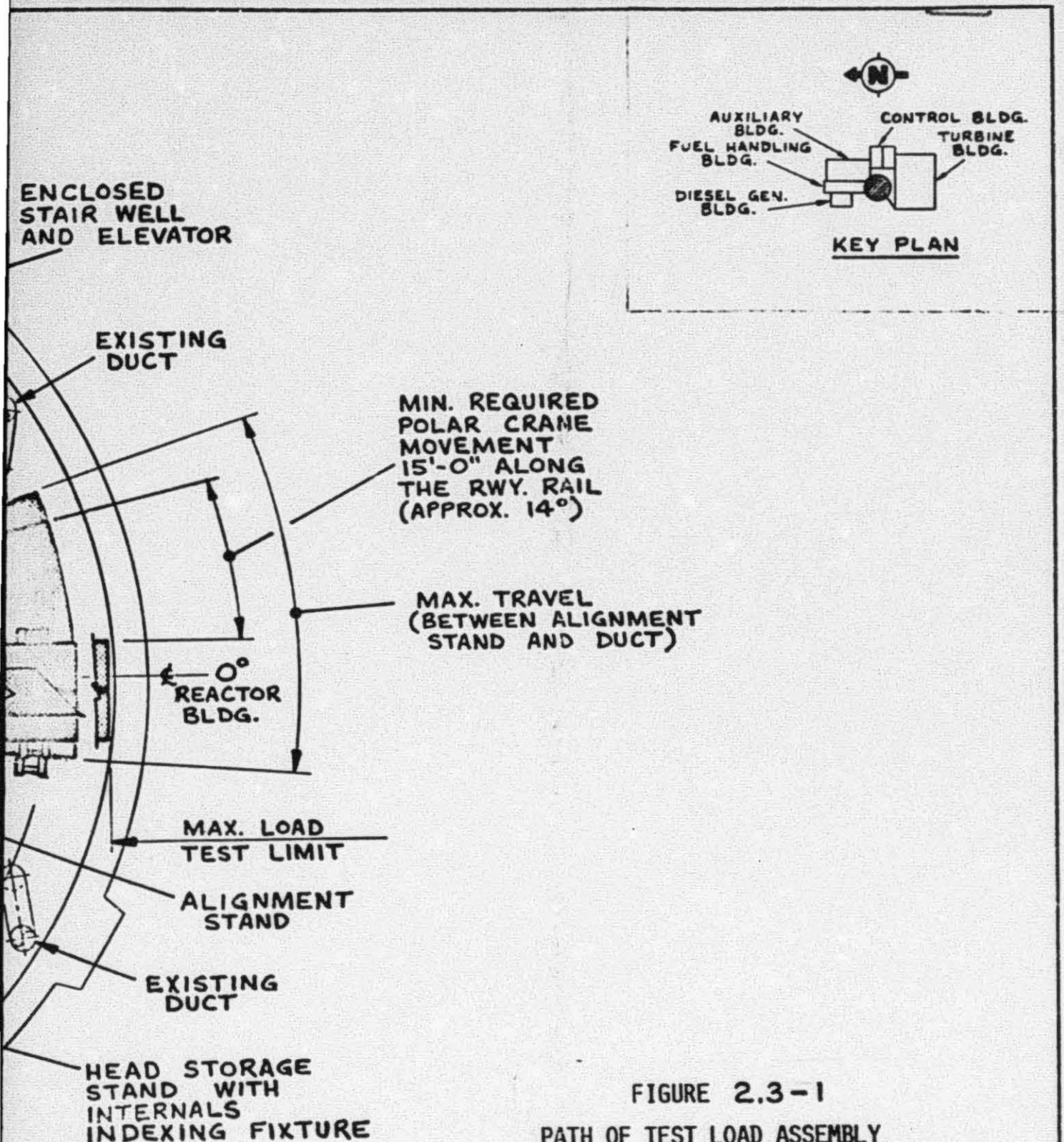
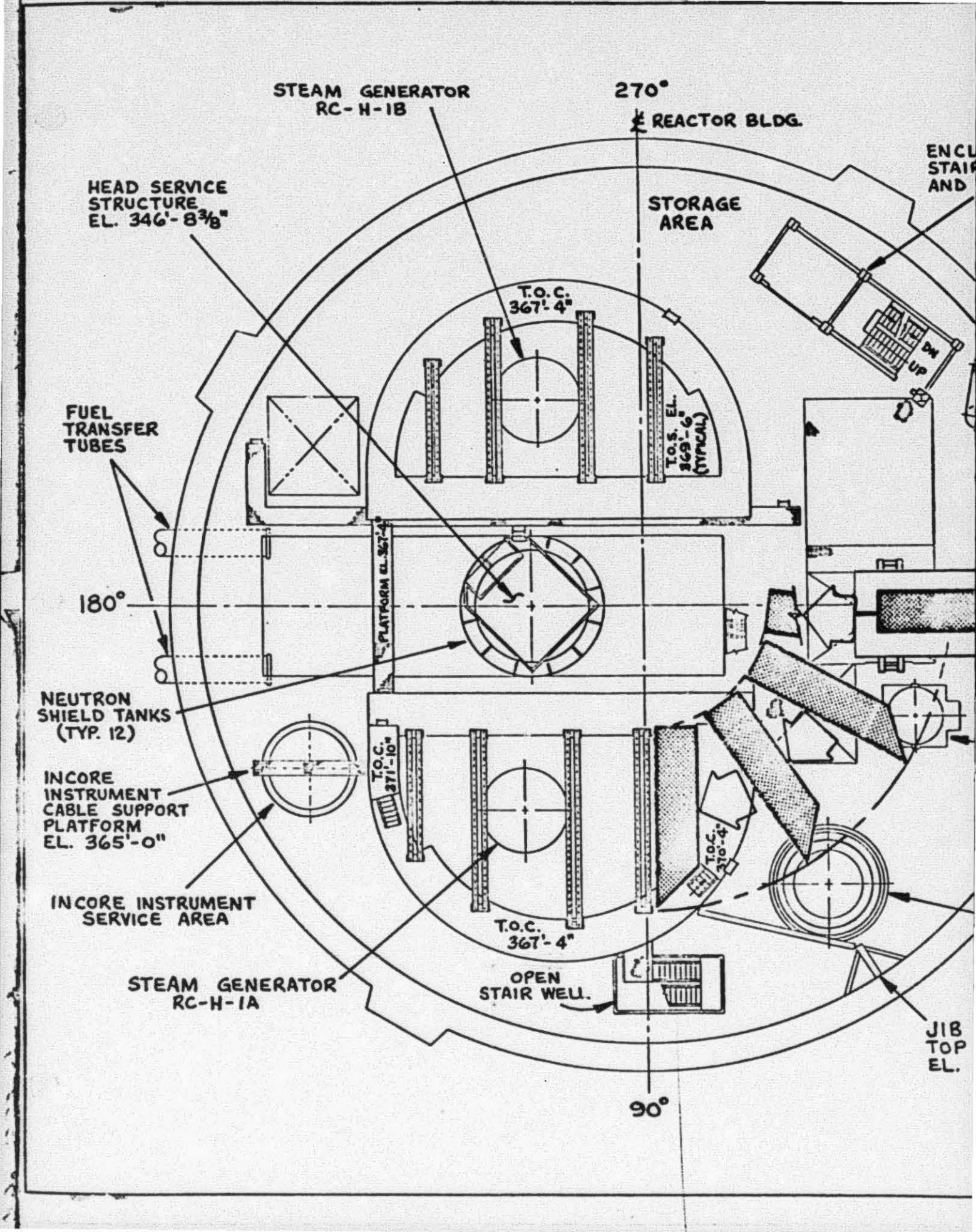


FIGURE 2.3-1

PATH OF TEST LOAD ASSEMBLY


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
ENCL
STAIR
AND
HEAD SERVICE
STRUCTURE
EL. 346'-8 3/8"
STORAGE
AREA
T.O.C.
367'-4"
T.O.S. EL.
369'-6"
(TYPICAL)
DN
UP
FUEL
TRANSFER
TUBES
PLATFORM EL. 367'-4"
180°
NEUTRON
SHIELD TANKS
(TYP. 12)
T.O.C.
371'-10"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
370'-4"
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
JIB
TOP
EL.
90°

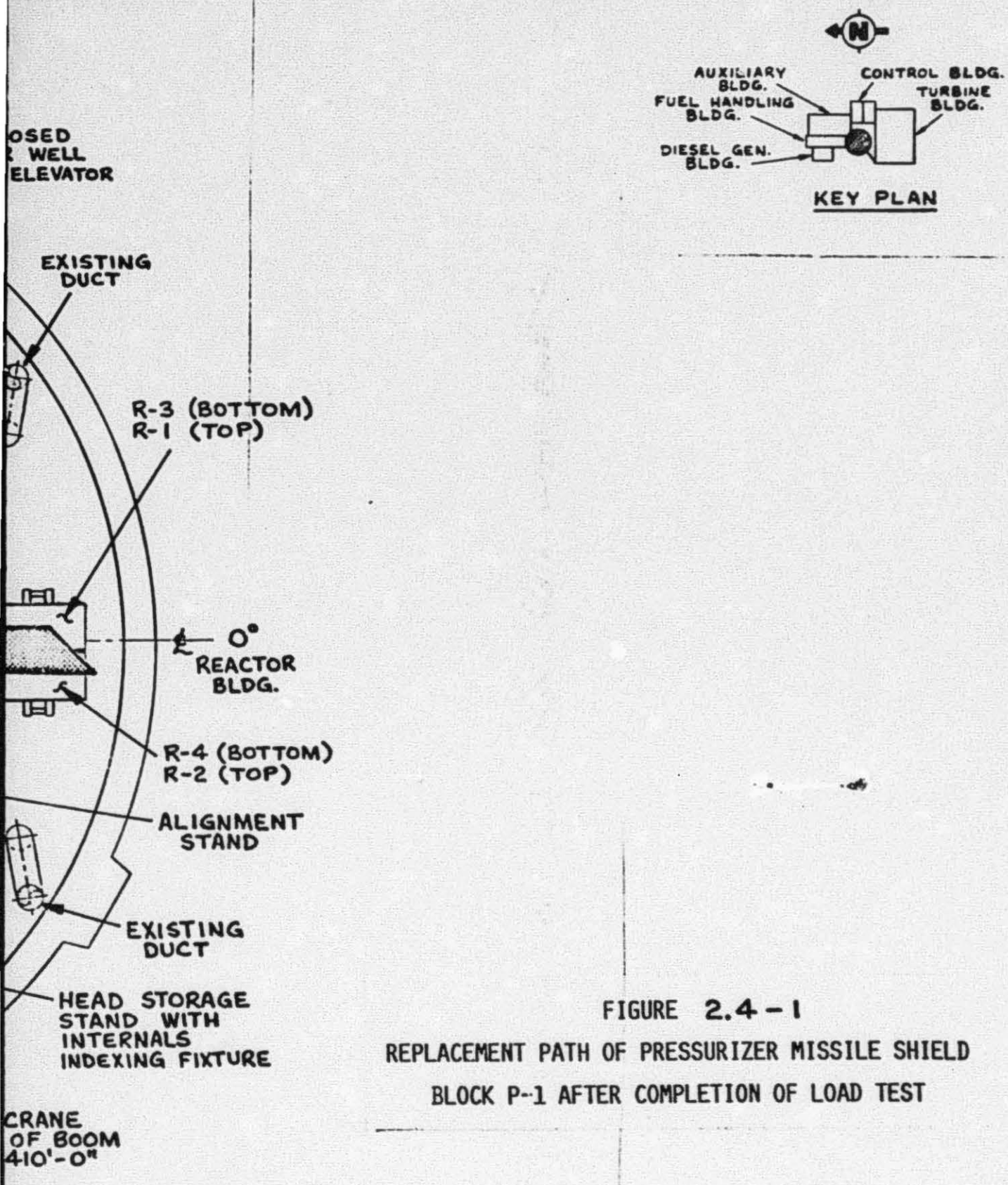
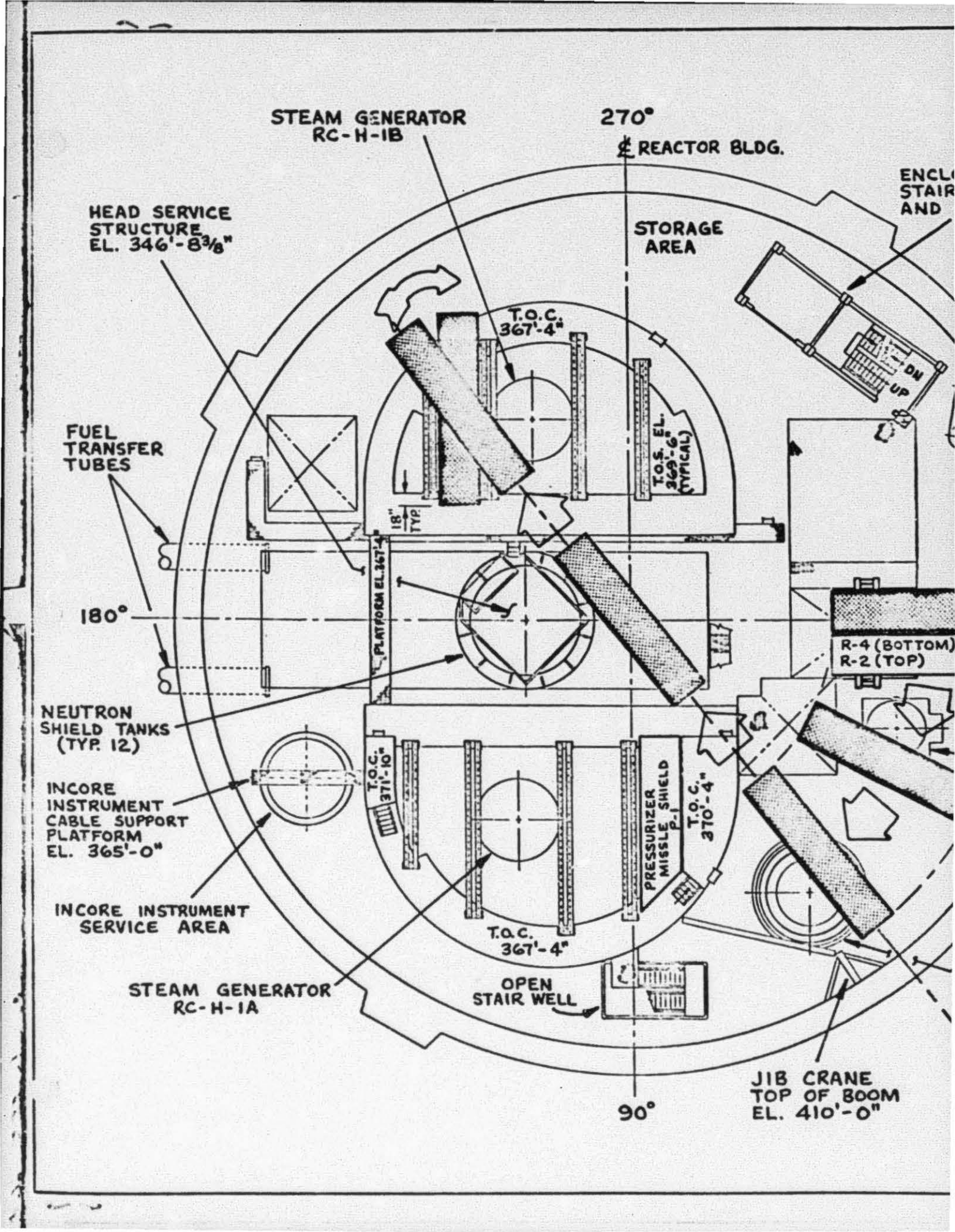
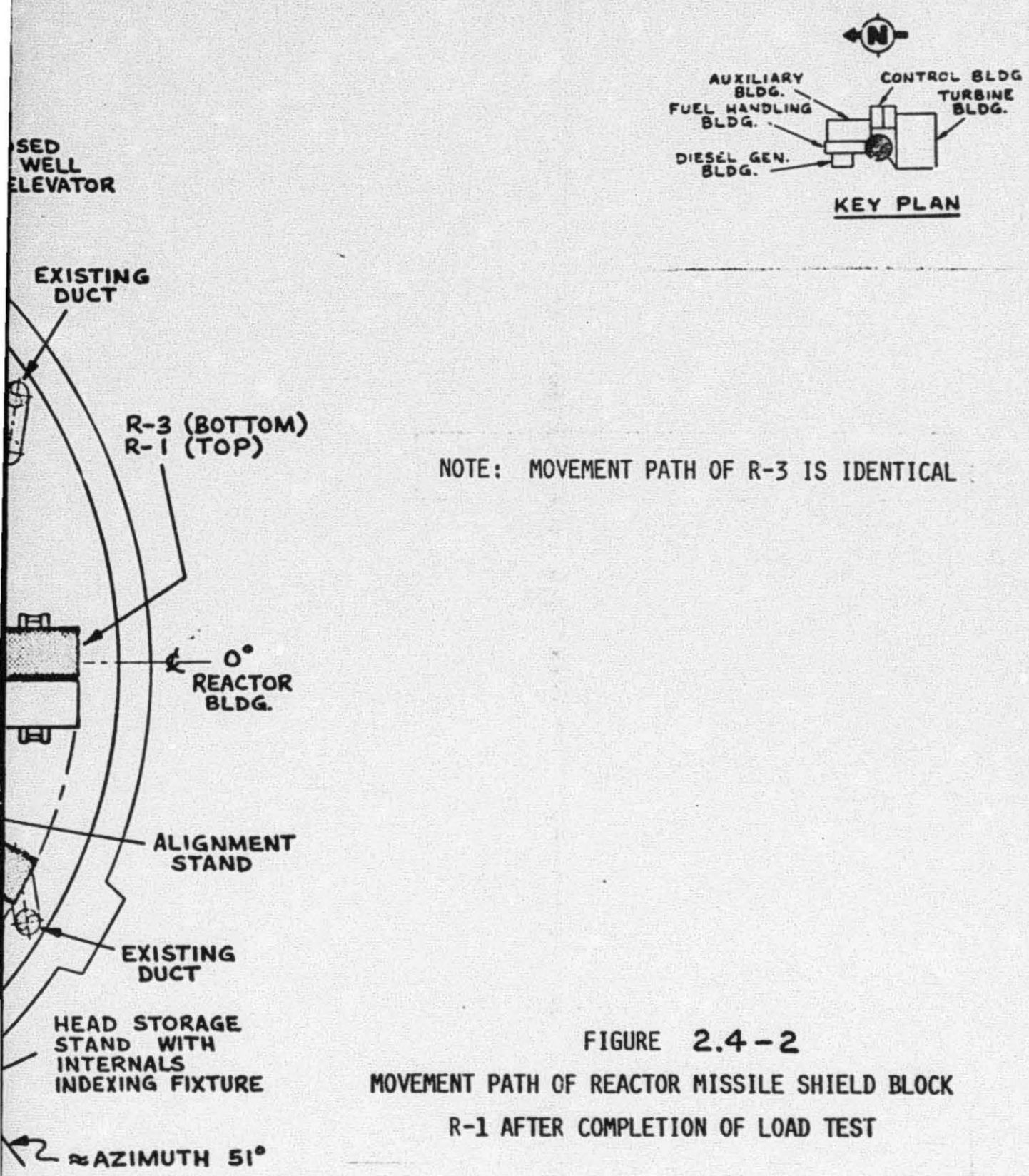


FIGURE 2.4-1

REPLACEMENT PATH OF PRESSURIZER MISSILE SHIELD BLOCK P-1 AFTER COMPLETION OF LOAD TEST


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
ENCL
STAIR
AND
HEAD SERVICE
STRUCTURE
EL. 346'-8 3/8"
STORAGE
AREA
T.O.C.
367'-4"
DN
UP
T.O.S. EL.
369'-6"
(TYPICAL)
FUEL
TRANSFER
TUBES
18"
TYP.
PLATFORM EL. 367'-4"
180°
R-4 (BOTTOM)
R-2 (TOP)
NEUTRON
SHIELD TANKS
(TYP. 12)
T.O.C.
371'-10"
PRESSURIZER
MISSLE SHIELD
P-1
T.O.C.
370'-4"
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
JIB CRANE
TOP OF BOOM
EL. 410'-0"
90°

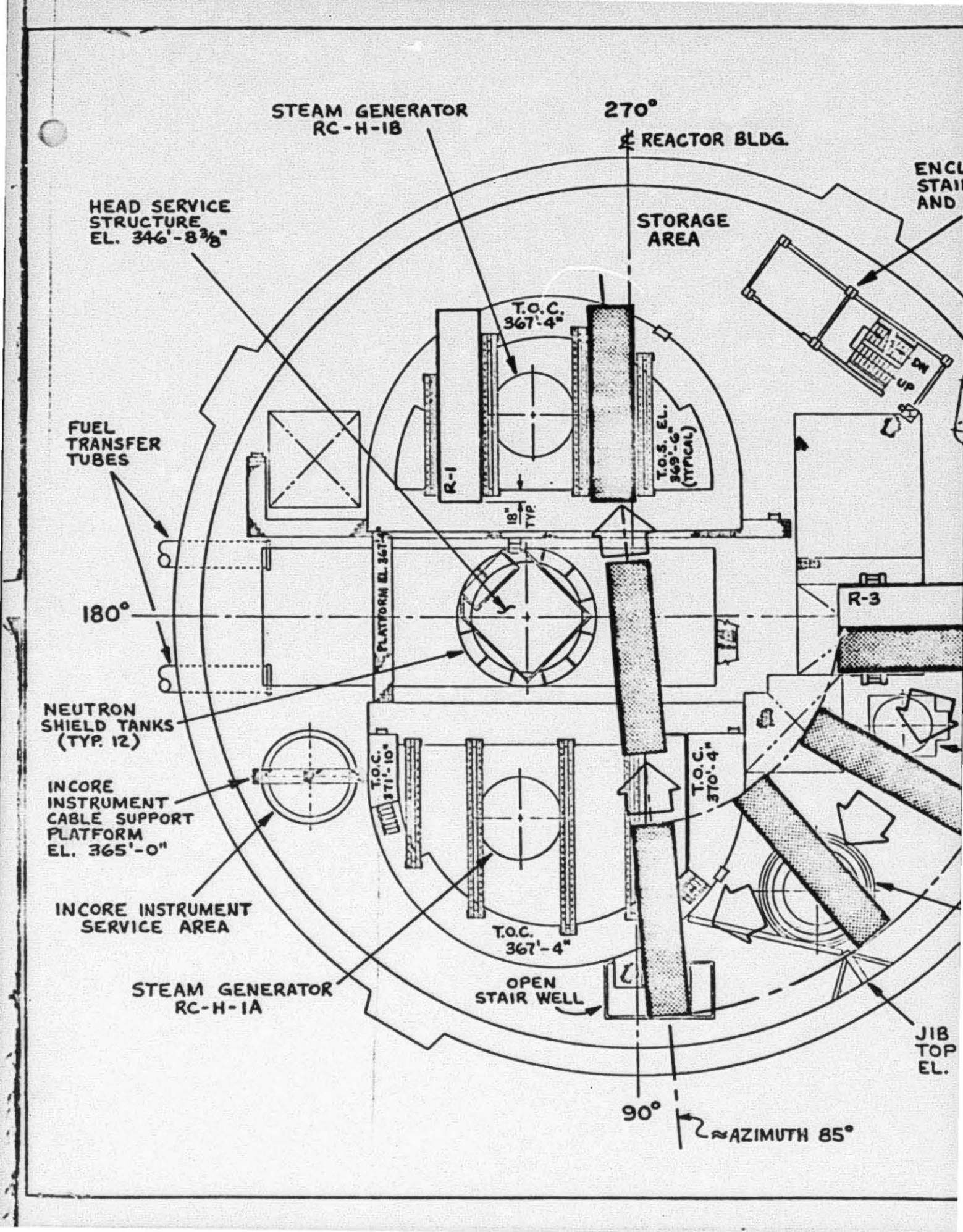


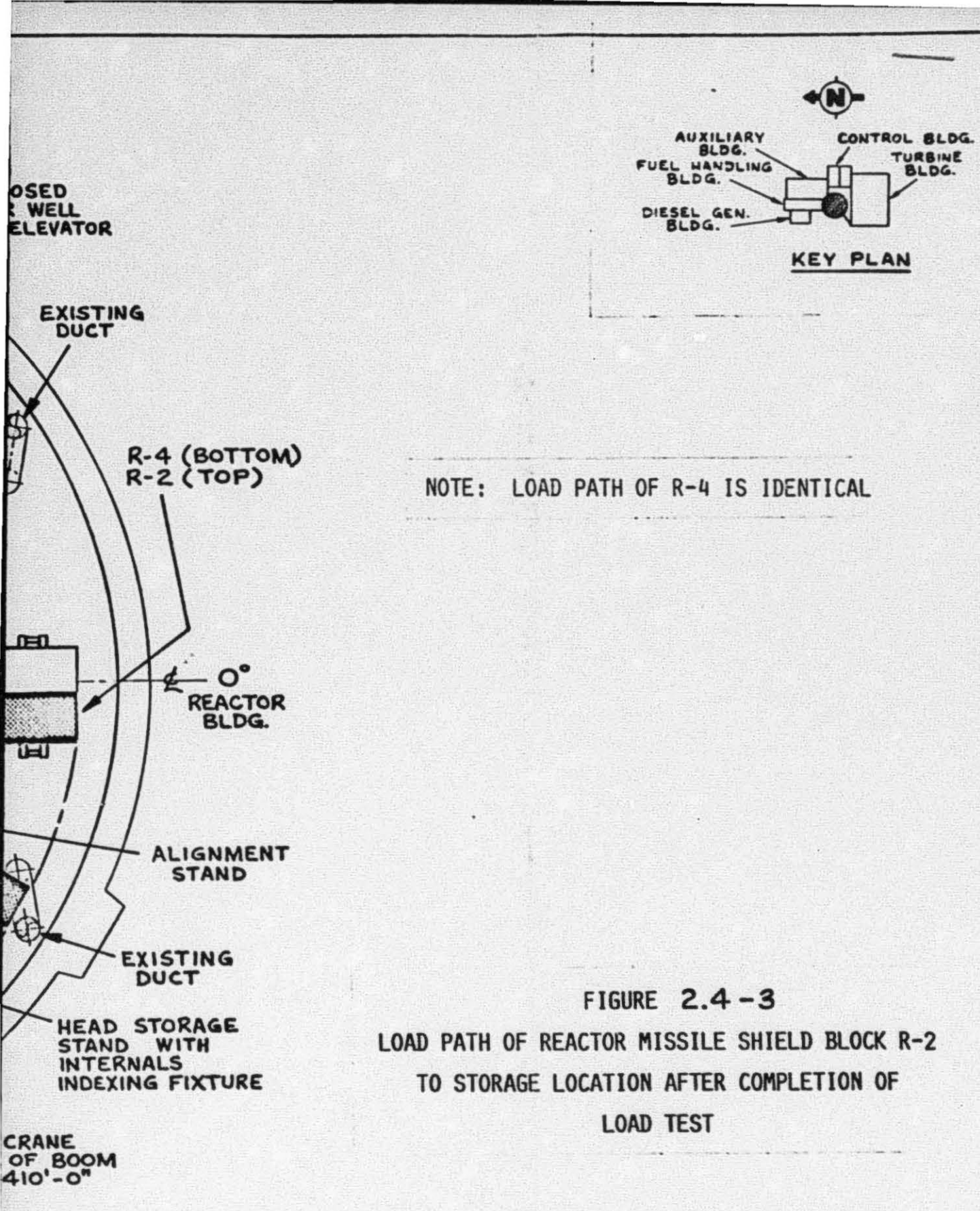
NOTE: MOVEMENT PATH OF R-3 IS IDENTICAL

FIGURE 2.4-2

MOVEMENT PATH OF REACTOR MISSILE SHIELD BLOCK

R-1 AFTER COMPLETION OF LOAD TEST


STEAM GENERATOR
RC-H-1B
270°
℄ REACTOR BLDG.
ENCL
STAI
AND
HEAD SERVICE
STRUCTURE
EL. 346'-8 3/8"
STORAGE
AREA
T.O.C.
367'-4"
DN
UP
FUEL
TRANSFER
TUBES
R-1
T.O.S. EL.
369'-6"
(TYPICAL)
18" TYP.
PLATFORM EL. 367'-4"
180°
R-3
NEUTRON
SHIELD TANKS
(TYP. 12)
INCORE
INSTRUMENT
CABLE SUPPORT
PLATFORM
EL. 365'-0"
T.O.C.
371'-10"
T.O.C.
370'-4"
INCORE INSTRUMENT
SERVICE AREA
T.O.C.
367'-4"
STEAM GENERATOR
RC-H-1A
OPEN
STAIR WELL
JIB
TOP
EL.
90°
≈AZIMUTH 85°



NOTE: LOAD PATH OF R-4 IS IDENTICAL

FIGURE 2.4-3

LOAD PATH OF REACTOR MISSILE SHIELD BLOCK R-2 TO STORAGE LOCATION AFTER COMPLETION OF LOAD TEST

3.0 COMPONENTS AFFECTED

3.1 The components affected by this test are the reactor building polar crane, the lifting devices associated with the load test, and the test load. Load testing of the polar crane will provide main hoist lift qualification to a maximum of 170 tons, which is sufficient to lift the reactor vessel head and service structure. Qualifying the polar crane to lift loads up to 170 tons is also sufficient for all planned recovery activities. In addition, the polar crane load test will serve as a load test for the reactor vessel head lift rigging.

3.2 The refurbished condition of the polar crane meets or exceeds the requirements stated in the Polar Crane Functional Description (Reference c). Further, an extensive inspection and maintenance program will have been completed before the load test. This program is described in detail in the document entitled 'Clarification and Exceptions to the "Crane Maintenance Inspection Specification" Checklist' (References c and e).

The reactor building polar crane was designed to allow for a rating of 500 tons. This includes the crane structure and main hoisting system, including load block with sister hook, brakes and two individual, though not redundant, wire rope systems. The main hoist braking system includes a load controlling eddy current brake and two main hoist (150 percent each) load holding brakes. The normal position of the holding brakes is engagement. Whenever the main hoist "up" or "down" circuits are energized the brake shoes are lifted from the brake drums by the action of direct-current clapper type magnets. Brake engagement is automatically accomplished by spring action. Whenever the hoist motors or the crane is deenergized the brakes are set. The crane will be controlled using a push button pendant station. To stop a load during lowering, the operator need only release the "down" button, or push the crane power button to "off", or have someone open a crane power breaker. Redundant upper limit switches are provided to prevent possible two-blocking.

Visual examinations and static electrical testing, as applicable, of the crane components necessary for the recovery effort will be completed prior to lifting any loads. These inspections encompass the critical mechanical, electrical and structural components and are directed by experienced crane inspectors. Also, a certified welding inspector will visually inspect critical, load bearing welds. The main hoist ropes will be virtually 100 percent inspected to ANSI B30.2.0-1976, paragraphs 2-2.4.1 and 2-2.4.2 criteria to determine if rope replacement is necessary.

Functional tests of the bridge, trolley, and main hoist control cabinets, and component parts of each drive system will precede operational testing. These tests prove the schematic function of each tested control cabinet and drive system component.

As necessary, defective components will be replaced prior to operational testing. The replacement components whose failure could result in a load drop, including the brakes, are replacement in kind for the originals and therefore are sized for the 500 ton rating.

An operational test will be performed in accordance with ANSI B30.2.0-1976, paragraph 2-2.2.1. As a minimum, this test will verify operation of all crane functions necessary for head lift. These functions are the bridge, trolley, and main hoist motions including the associated trolley and redundant main hoist upper limit switches.

Highlights of significant refurbishment activities are summarized as follows:

- Main hoist brakes replaced. Break drums cleaned and brakes fully adjusted.
- Structure of crane inspected and accepted (including weld & bolt inspection).
- Crane completely relubricated where required. All fluid levels checked and corrected where necessary.
- Oil samples from gear cases analyzed and considered in good condition.
- All motors slip rings refurbished.
- All clutch plates cleaned and all clutches adjusted.
- All couplings checked.
- All circuits meggered, or checked for continuity.
- Electrical contactors and relays replaced where defective.
- All bridge motor and trolley resistor banks replaced.
- New pendant station installed.
- Festoon system for pendant station refurbished.
- Cab controllers verified operational.
- New power system to crane and trolley installed.
- New fire extinguishers installed on crane.
- Runway rail inspected for alignment, gaps, loose bolts, level and roundness.
- Wire rope sample tested by laboratory and reported to meet or exceed original certification.

- o Wire rope completely lubricated.
- o Wire rope inspection indicates rope in good condition.
- o All crane functions verified operational except under load.

Additional and supplemental documentation of the results of the inspection and refurbishment program have been and will continue to be submitted as they are accomplished.

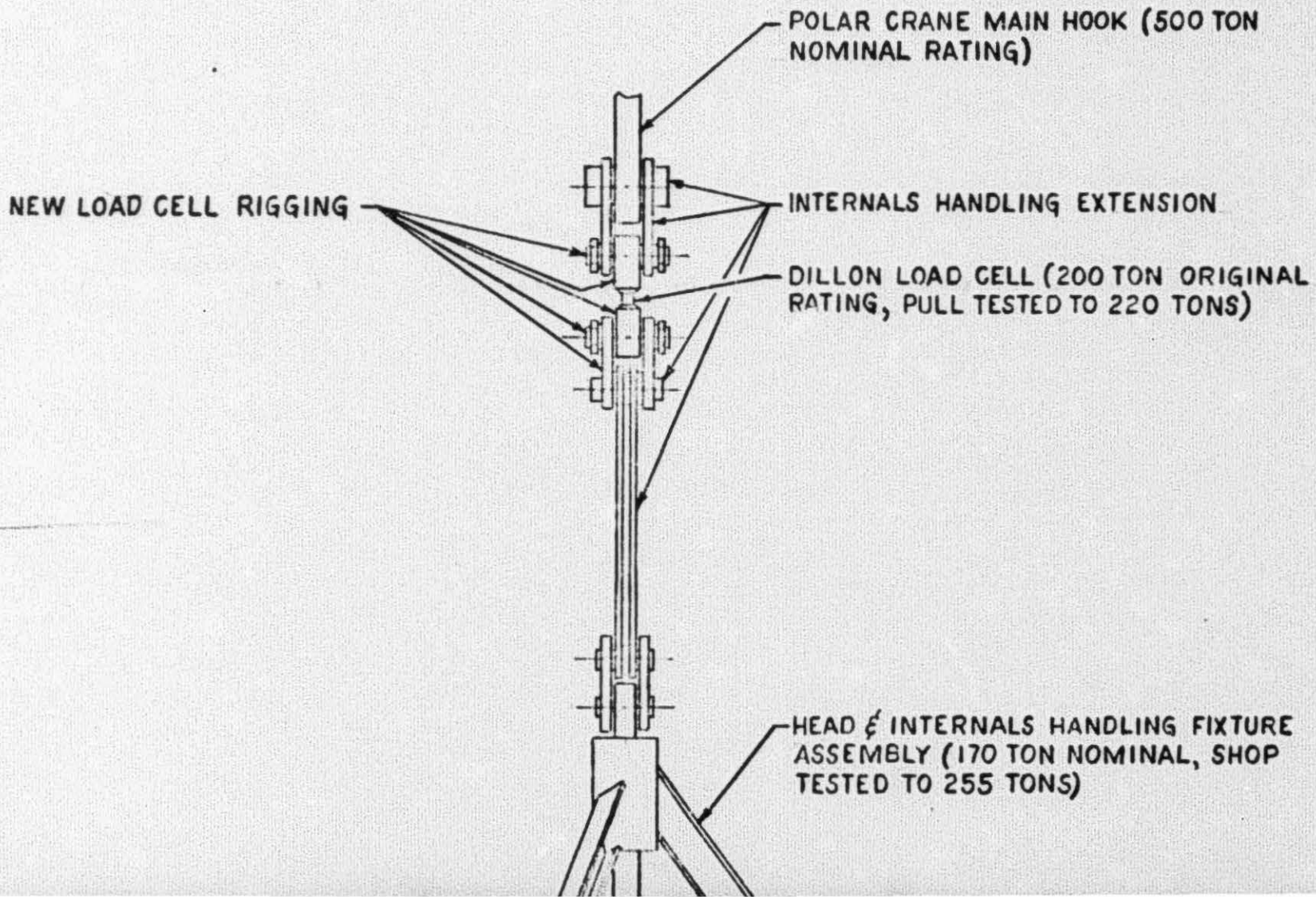
3.3 The lifting device associated with the load test is composed of several components: hook attachment shackles, a load cell, connecting shackles to the head lift rig tripod, the head lift rig proper, the load test frame, and associated load bearing connectors. This lifting device is shown and described in Figure 3.3-1.

The lifting device to be used for the missile shield block is shown and described in Figure 3.3-2.

Because the reactor vessel head lift rig will be concurrently tested as a part of the polar crane load test, it will receive special attention in that a rigorous visual examination will be conducted after it has lifted the test load. A more detailed and comprehensive presentation regarding the requalification of reactor head lift rigging is contained in reference d.

The test load frame component of this lifting rig assembly has been specially designed for its purpose and will be used only during load testing. As such, it is designed in accordance with the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings". Design load for this component is 220 tons.

3.4 The test load is composed of the four reactor missile shields weighing 40 $\pm$ 1 tons each and one pressurizer missile shield weighing 32 $\pm$ 1 tons. The combined weight of the discrete test load is, therefore, 192 $\pm$ 5 tons. The total lifted load also includes the contribution from the weight of the test frame and associated rigging, bringing the total lift to a minimum of 200 tons and a maximum of 220 tons. The actual test load is estimated as accurately as possible at 212 tons.


POLAR CRANE MAIN HOOK (500 TON NOMINAL RATING)
NEW LOAD CELL RIGGING
INTERNALS HANDLING EXTENSION
DILLON LOAD CELL (200 TON ORIGINAL RATING, PULL TESTED TO 220 TONS)
HEAD & INTERNALS HANDLING FIXTURE ASSEMBLY (170 TON NOMINAL, SHOP TESTED TO 255 TONS)

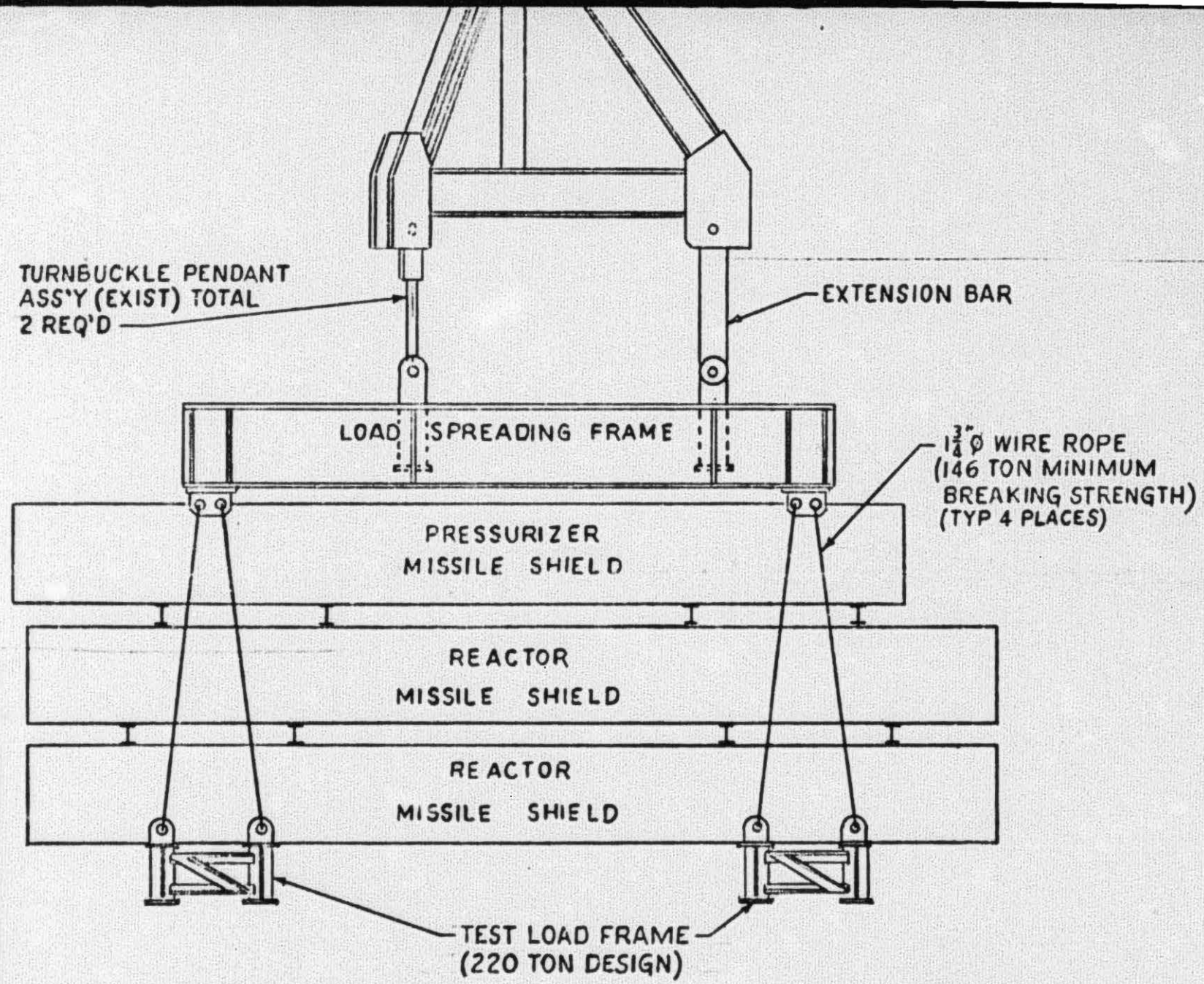
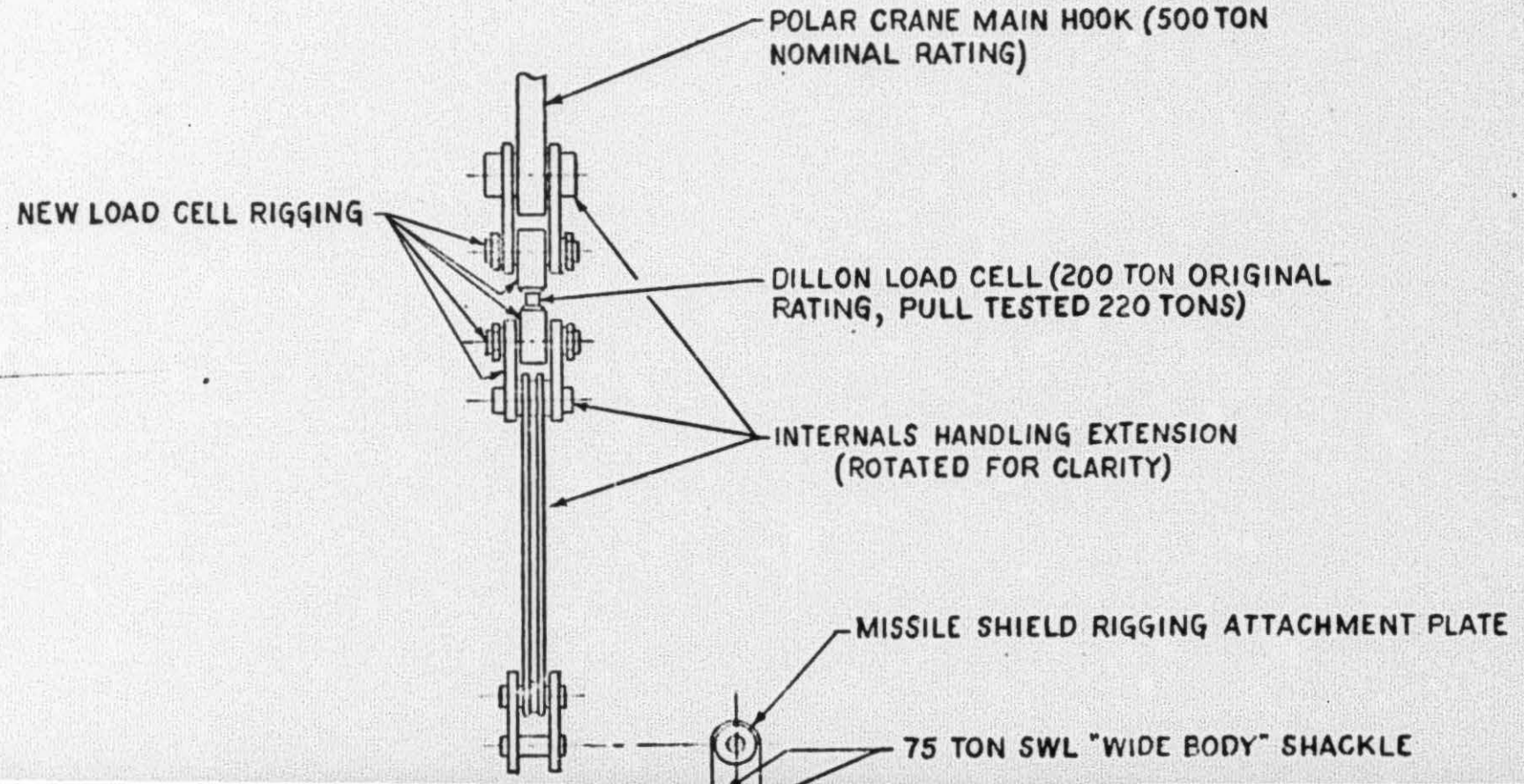


FIGURE 3.3-1
LOAD TEST RIGGING
(COMPLETE ASSEMBLY)

NOTE: THE MISSILE SHIELD BLOCK LIFTING PAD EYES HAVE A SAFETY FACTOR OF 3 TO ULTIMATE (BREAKING). THE DESIGN BASIS FOR THIS SAFETY FACTOR IS:

1. ONLY TWO CABLES SUPPORT LOAD (INSTEAD OF 4).
2. CABLE LIFT ANGLE WITH THE VERTICAL DOES NOT EXCEED 30°.



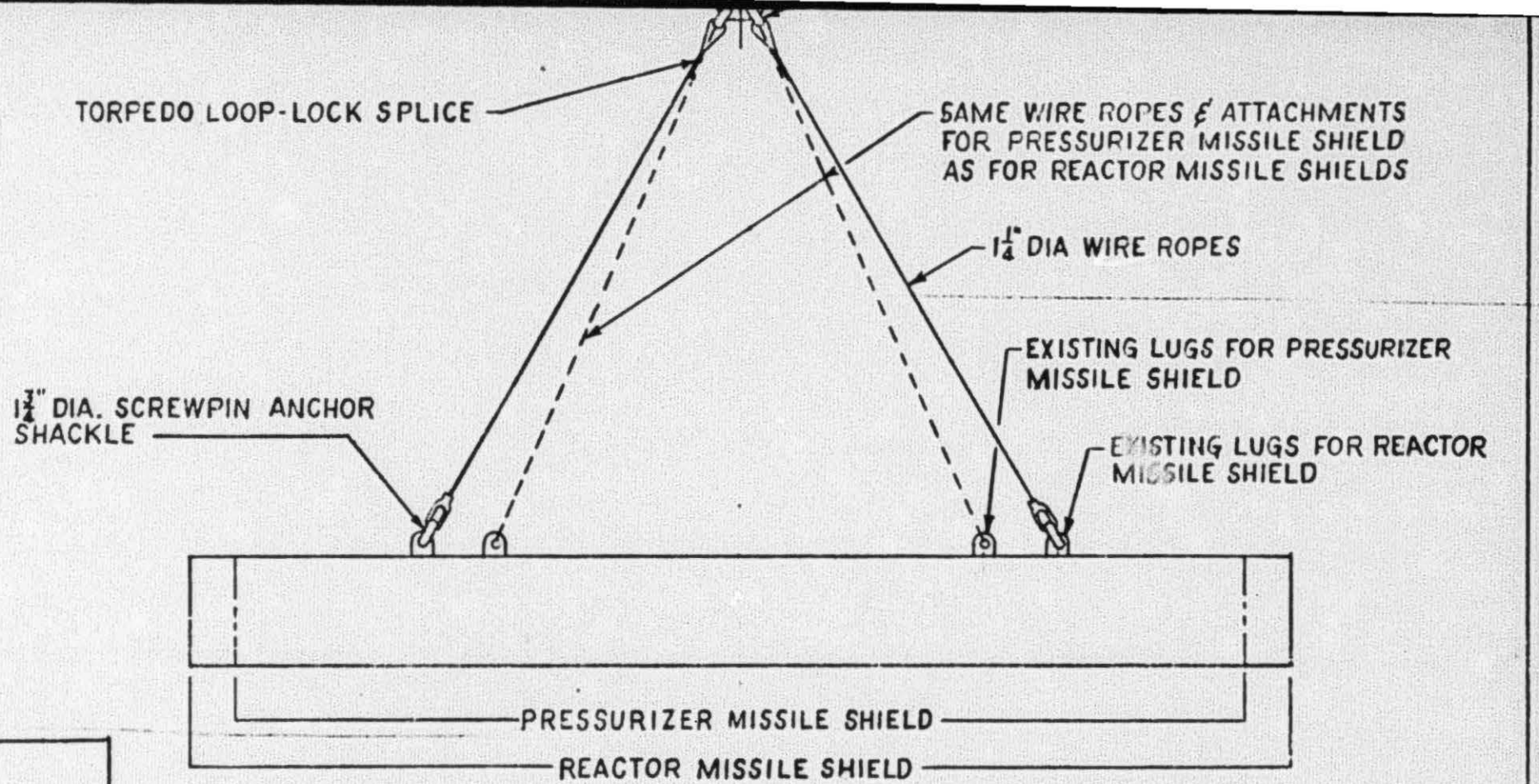


FIGURE 3.3-2
MISSILE SHIELD RIGGING
(COMPLETE ASSEMBLY)

APPROX. SCALE 1/4" = 1'-0"

4.0 HEAVY LOAD DROP ANALYSIS - BACKGROUND AND INTRODUCTION

The generic letter on the control of heavy loads requires licensees to address the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Section 2.3 of this letter requires information be provided which is sufficient to demonstrate that adequate measures have been taken to ensure that in the vicinity of the reactor core, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria I-III of NUREG-0612, section 5.1.

Section 2.4 of the generic letter requires information to be provided which is sufficient to demonstrate that in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment will be limited so as not to result in the loss of these safety-related functions (Criterion IV).

Even though TMI-2 polar crane is not a single - failure - proof crane in the classic sense as defined in NUREG 0554, this does not totally negate the argument that the probability of dropping a certain load on a certain target at a specific time is "extremely small".

Approaching the particular case of a missile shield drop from a more mechanistic standpoint, it can be concluded that the probability is indeed extremely small by taking into account the following factors:

- The Polar Crane factor of safety is greater than 10 compared to the original design rating and approximately 5 compared to the requalification rating.
- The missile shield blocks are moved sequentially starting with the one farthest from the load test frame using the intervening blocks to protect the reactor from drops of the block being moved.
- Shield block lift rigging incorporates a significant degree of conservatism as may be seen in figure 3.3-2.
- The amount of time during which these loads are lifted in the vicinity of the reactor vessel will be minimized.

In the case of the test load similar arguments regarding conservatism of crane and rigging capacities can be made. However, since the test load is approximately five times greater than a missile shield, the following additional conservatisms are introduced:

- The test load area was carefully reviewed and selected on the basis that a minimum amount of equipment was located directly beneath it in comparison to other areas of the contaminent.
- Lifting time of the test load has been minimized to the greatest extent compatible with crane requalification.

Procedural conservatisms have also been considered and introduced. For example, the initial loads to be lifted by the polar crane will be lighter than later loads. As described in section 2, the 6 ton Internals Indexing Fixture will be lifted prior to the movement of a missile shield. Each time a load to be lifted is heavier than any previously lifted load, the procedure will require that the new load be lifted in steps to insure that, should a failure occur, consequences would be minimized. The procedure will require that the new load be lifted initially only a small distance, held in place to verify that no problems are encountered, and then complete the lift. An example of this is the initial lift of a reactor vessel missile shield. The missile shield will be lifted only a small distance and held in place while still on the guide studs. A load drop of the missile shield would have no unacceptable consequence as it would merely settle back into place on the D-rings across the refueling canal.

Further, no reliance has been placed on the installation and use of electrical interlocks or mechanical stops to keep the load in its prescribed load path. The movement of the load will be controlled by the Test Director, who will be equipped with a voice actuated headset as will the other in-containment personnel associated with the load test. The Test Director can communicate with a person stationed by the main power supply breaker, 480V MCC-2-32A, in the Auxiliary and Fuel Handling Building. In case of an emergency, the Test Director can have the main power supply interrupted. This will freeze the crane in the position it was in when the power was cut off.

The above points, taken in conjunction with the refurbished condition of the crane as delineated in section 3, lead to the conclusion that the probability of a load drop is in fact so small that it approaches the drop probability of the so-called single failure proof cranes, which is to say, incredible.

Notwithstanding this low probability of a drop in the first place, a detailed analysis has been conducted to examine the potential consequences of load drops in the vicinity of important equipment. These analyses encompassed all equipment in the load path down to the level of detail of individual valves and instrument lines. These analyses are summarized below, taking selected major equipment and systems or examples.

4.1 HEAVY LOAD DROP - CORE VICINITY

4.1.1 Identification of load

For the performance of the polar crane load test, the loads to be moved in the vicinity of the reactor core are the reactor vessel missile shields. Figures 2.2-2, 2.4-2 and 2.4-3 show the paths for the movement of the missile shields to the load test frame south of the D-rings on the 347' elevation and their movement from the test frame to their storage location on the "B" D-ring after completion of the test.

These missile shields are constructed in the shape of oblong blocks of concrete and rebar, weighing approximately 40 tons each.

4.1.2 Load/Target Interaction

In the event that a shield block were to fall onto the reactor head and service structure, damage to the Control Rod Drive Mechanism (CRDM) motor tubes would result causing leakage of reactor coolant into the reactor building. The maximum leakage would be the draining of the RCS to the top of the level of the reactor vessel closure head, since penetration of the 8" thick steel closure head is not credible.

A shield block striking the reactor vessel (RV) closure head could also bring about some physical redistribution of loose core debris within the reactor coolant system.

It should be noted that much of the kinetic energy of the falling shield block would be absorbed in physical deformation of the service structure and CRDM apparatus above the head and that an instantaneous impact directly on the RV head proper would not occur.

4.1.3 Criteria - Specific Evaluations (NUREG 0612)

4.1.3.1 Criterion I:

Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

Evaluation:

The impact of a missile shield block dropping onto the reactor vessel head and service structure might cause leakage of reactor coolant through the CRDM motor tubes into the reactor building as described in Section 4.1.2. This liquid would be contained in the reactor building; thus, the containment building would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any gaseous releases caused by this postulated drop would be physically contained, since the containment integrity will be set and maintained throughout the load test. Containment integrity is further assured since there is no longer any energy source capable of producing a driving pressure which could transport this activity across the containment boundary.

Any gaseous activity released in the containment would be directed through the high efficiency particulate air (HEPA) filters and containment purge exhaust system and eventually released in a controlled manner, so as not to exceed the limits established in Criterion I.

Further, any releases which might occur in spite of the factors presented above would be only a small fraction of the calculated release presented for a Loss-of-Coolant Accident (LOCA) in Chapter 15 of the TMI-2 FSAR; thus meeting Criterion I.

4.1.3.2 Criterion II:

Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95.

Evaluation:

The precise configuration of the fuel is presently unknown; therefore, the exact k_{eff} resulting from the potential redistribution of the fuel due to the impact of a missile shield block on the reactor vessel head and service structure cannot be calculated. Despite the inability to calculate the exact k_{eff}, bounding analyses performed for the 100% fuel damage case, conclude that the fuel debris will not be critical when: it is in its most reactive condition, the effects of structural material are accounted for in the analysis and the reactor coolant boron concentration is 3000 ppm. (References f & g) In view of this conclusion and the fact that the concentration of boron in the reactor coolant is over 3500 ppm, a recriticality is precluded.

4.1.3.3 Criterion III:

Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated).

Evaluation:

As stated in section 4.1.2, the maximum leakage resulting from a drop of a missile shield block onto the reactor head and service structure would be to drain the reactor coolant system to the level of the top of the reactor vessel closure head. Drainage to this level does not uncover the fuel; thus, crition III is met.

In addition, at least one make-up train capable of delivering water with a 3500 ppm boron concentration to the reactor vessel would be available.

4.2 HEAVY LOAD DROP ANALYSIS - SAFE SHUTDOWN AND DECAY HEAT REMOVAL EQUIPMENT

4.2.1 Identification of loads

For performance of the polar crane load test, the loads to be moved in the vicinity of safe shutdown and decay heat removal equipment (or other equipment specifically important at TMI-2 due to unique site considerations) are: the internals indexing fixture (a six ton metal cylinder as shown on figure 2.1-1), the missile shields (including the 32 ton pressurizer shield), and the load test assembly composed of the missile shields stacked on a frame or shown on figure 3.3-1.

The indexing fixture is not specifically addressed in the analyses because the results of a drop of this item is enveloped and bounded by the missile shield loads. The assembled test load is specifically addressed.

4.2.2 Identification of targets

A list of systems and components including valves and instrumentation, which were considered essential functions for TMI-2 was compiled. The criteria by which this equipment was selected are:

1) that equipment within the reactor coolant pressure boundary which is required for decay heat removal and reactivity control,

2) that equipment required to be operable by the TMI-2 Recovery Technical Specifications, and

3) that equipment required by plant procedures which were approved in accordance with Technical Specification 6.8.2.

The following is a list of major fluid systems which were examined as possibly presenting important targets:

- Reactor Coolant
- Make-up & Purification
- Decay Heat Removal
- Mini-Decay Heat Removal
- Standby Reactor Coolant Pressure Control System
- Core Flood
- Decay Heat Closed Cooling Water
- Reactor Building Spray
- Chemical Addition
- Nuclear Services Closed Cooling Water
- Feedwater and Condensate
- Main Steam
- Demineralized Service Water
- Intermediate Closed Cooling Water
- Nuclear Services River Water
- Reactor Building Ventilation
- Reactor Building Purge
- Fire Protection
- Steam Generator Secondary Side Vent and Drains

4.2.3 Load/Target Interactions

The effects of a heavy load drop in areas over which heavy loads are expected to be moved have been analyzed to ascertain the worst credible consequence. (These areas are shown on Figures 4.2-1, 4.2-2 and 4.2-3.) In determining the consequences of a heavy load drop, it was determined that the floors on elevations 347'-6" and 305' would locally collapse when impacted. The impact areas and load/equipment combinations have been presented in matrix format in Table 4.2-1.

Components were not considered to be functional after a heavy load drop which was assumed to occur directly over the components. (The availability of an unaffected alternative was ascertained in view of why the component was required. A component was considered to be an alternative only if it performed the same safe shutdown function as the component subjected to a heavy load drop.) The location of each piece of essential equipment was determined from the latest available drawings.

The consequences of the loss of each component lying within the designated areas was determined and is presented in the following subsection 4.2.4.

4.2.4 Criterion IV - Specific Evaluation (NUREG-0612)

Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

Evaluation:

Criterion IV refers to "required safe shutdown functions" which are defined as those required to: maintain the reactor coolant pressure boundary, reach and maintain subcriticality, remove decay heat, and maintain the integrity of components whose failures could result in excessive off-site releases.

The required safe shutdown functions that apply to the TMI-Unit 2 reactor in its current cooling mode and core configuration are:

1) the capability to maintain subcriticality.

2) decay heat removal.

3) the capability to maintain the integrity of components whose failures could result in excessive off-site releases.

The reactor coolant pressure boundary needs only to be maintained insofar as reactor coolant must be maintained in the RCS for decay heat removal and reactivitiy control. Radiation shielding is discussed and evaluated in section 5.2, part 4.

Required safe shutdown functions unique to TMI-2 are addressed in the same order as above:

1) The capability to maintain subcriticality in the core due to load impact on the reactor itself is addressed in section 4.1.3.2. Due to the configuration of TMI-2, the only credible mechanism by which criticality control could be compromised is deboration of the water in the reactor coolant system. Systems within load impact areas which contain unborated water have been investigated and found to fail in such a way as to drain their contents onto the reactor

building floor and not into the RCS. For example, the boron concentration of the RCS could be reduced by gross leakage from the unborated secondary side into the primary side as a result of a postulated load drop. Damage to the OTSGs severe enough to cause such leakage would undoubtably cause damage to the outer surface of the OTSGs allowing the unborated water to drain to the containment sump.

Further, systems capable of injecting highly borated water into the RCS are available and it is not feasible with one load drop to reduce the functional capability of these systems to such a point that boron injection could not be done.

2) Decay heat removal capability is ensured by maintaining water in the reactor vessel. Analysis shows the water could be drained to the bottom of the cold leg nozzles (elev. 314'-4") and no adverse consequences such as boiling would be experienced. The only way to drain the vessel below this level would be through damage to the incore instrument tubes. (See item C below.) In addition, if damage to the RCS causing leakage of reactor coolant were to occur, make-up capability will exist at least through one loop since a drop in both D-rings at the same time is not credible by physical separation. Damage to make-up system penetrations will not occur since they are located on the northern side of the building away from the load paths.

3) Off-site releases are prevented by the containment pressure boundary. The containment integrity, as required by the Technical Specifications, will be set during the load test. All containment penetrations which have the potential for damage in the event of a load drop have closed isolation valves outside the reactor building.

Please refer to section 5.2, part 3, for additional discussion.

Additionally, a number of extraordinary accident scenarios have been postulated as having some finite although remote occurrence potential. These are:

A) Criticality in a steam generator.

B) Criticality in the reactor building sump.

C) Impact induced failure of incore instrument piping.

Extraordinary accident scenarios are addressed in the same order as they are stated above:

A) A sequential series of low probability events must occur before the concern of criticality in the steam generators can be viewed as a ligitimate safety issue. The following set of low probability events must occur in sequence, each conditioned on the occurrence of all the prior low probability events, before a criticality in the steam generator could occur.

1. A missile shield must be dropped above one of the D-rings.

2. The missile shield must travel far enough into the D-rings to impact a reactor coolant pump or cold leg piping. It should be noted in this regard that there are massive structural beams crossing the D-ring above the reactor coolant pump elevation from which the reactor coolant pumps are vertically supported.

3. The missile shield must impact the reactor coolant pump or other structure in such a way as to rupture the pump suction line at a point well below the secondary side water level.

4. An amount of fuel sufficient to raise criticality concerns must have been transferred to the steam generator and found its way into the steam generator tubes and lodged there during the accident.

5. That fuel must be in a high density close pack configuration within the tubes in such a manner that would allow criticality if the borated water were drained from the tubes during the period of time that they were surrounded by unborated secondary water.

As can be seen from the above description of events, even the application of conservative probabilities to each event of the required sequence will result in a probability of criticality occurrence which is below any reasonable threshold for safety concern. This extraordinary accident is, therefore, appropriately dispatched.

B) In regard to subcriticality in the containment sump, our evaluation indicates that the only point of potential concern would be the drop of a heavy load onto the systems listed on Table 4.2-2, which might, in turn, provide a source of unborated water to the sump and raise concerns related to potential criticality within the sump. This problem may be addressed in two ways: First, to limit the amount of unborated water available for leakage to the sump from these systems, the water supply to these systems will be isolated for the period of time that the load test is being performed. (Note: The operable fire protection system is normally isolated.) Our evaluation shows that a significant quantity of unborated water would be required to lower the sump water concentration from current values to a level below the 1700 ppm value which has been specified as the reasonable point for sump reactivity problem avoidance.

Secondly, several low probability events would have to have occurred before valid concerns regarding sump criticality could arise, regardless of the amount of unborated water delivered to the sump. First, an amount of fuel sufficient to create a critical mass would have to have been washed to the sump during the TMI-2 accident. Secondly, this fuel would have to be in a configuration

which could induce criticality if a global deboration of the sump were to occur. Our qualitative assessment indicates that the administrative controls described above (which limit the amount of unborated water which could be delivered to the sump) combined with the low probability of simultaneous occurrence of the initial conditions regarding fuel deposition in the sump which could lead to a criticality problem effectively eliminate this issue as a legitimate safety concern.

C) The third special consideration which has been evaluated is that regarding the potential consequences of a load drop which damaged the incore instrument lines.

The reactor vessel lower head is penetrated by incore instrument lines which run from beneath the very bottom of the vessel through a tunnel in the base mat of the containment to terminations at the seal table area. This termination area is shown on Figure 4.2-3. Shown in phantom on the figure, the routing of these lines parallels a line drawn between the center of the reactor vessel and the center of the seal table. The width of the area occupied by these lines is just smaller than the diameter of the seal table.

As can be seen in Figure 4.2-3, the seal area is not located within the load path. A portion of the incore instrument lines is, however, physically located below the area of the load path for the reactor missile shield blocks. This portion of the lines is separated vertically from load impact surfaces by concrete and steel structures of such massive proportion as to render load penetration incredible.

The only remaining scenario by which a dropped load could damage an incore instrument line is as follows:

1) Drop of reactor missile shield into refueling canal.

2) Shield block to orient and reconfigure itself to fit between reactor vessel and primary shield wall. (Present dimensions preclude such an event).

3) Shield block to travel down to elevation of reactor vessel skirt, disintegrate into pieces small enough to fit through holes in the RV skirt, and travel horizontally far enough and with sufficient remaining energy to damage the stainless steel incore lines.

This scenario is judged not credible based on the improbability of these three items; especially item 2 which violates the physical laws of nature.

TABLE 4.2-1 SHEET 1 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	FUEL TRANSFER CANAL		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Reactor Missile Shield Blocks (4: R-1, R-2, R-3, R-4) 40 Tons each	346'-8" (elev. of top)	Reactor Vessel Closure Head and Service Structure (RC-T-1) (See Figure 4.2-1)	C, D, E (4.1.3.1, 4.2.4 Item 1,2)
	347'-6"	3" Intermediate Closed Cooling (IC) System Piping - APSR cooling water lines (See Figure 4.2-1) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 2 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF A D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Pressurizer Missile Shield Block (P-1) 33 Tons	347'-6"	2 1/2" Fire Service (FS) Line and Hose Reel (See Figure 4.2-1) (Unborated Water)	C, D, E (4.2.4 Item B)
	347'-6"	3" Demineralized Water (DW) line - spray line to Head Storage Stand (FH-Y-11) (See Figure 4.2-1) (Unborated Water)	C, D, E (4.2.4 Item B)
	305'	2 1/2" Fire Service (FS) Line and Hose Reel (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)
	305' (act. 342'-6")	3" Demineralized Water (DW) System Piping (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 3 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF A D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Pressurizer Missile Shield Block	305'	Core Flood Tank (CF-T-1B) (See Figure 4.2-2) (Drained)	C, D, E Note 1
	305'	Air Coolers - (AH-E-1C; AH-E-1D; AH-E-1E) (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)
	305'	Reactor Building Emergency Cooling - River Water (RR) System Piping (Unborated Water) (See Figure 4.2-2)	C, D, E (4.2.4 Item B)
	282'-6"	Reactor Coolant Drain Tank (WDL-T-3) (See Figure 4.2-3)	C, D, E Note 2
	282'-6"	Leakage Coolers (WDL-C-1A; WDL-C-1B) (See Figure 4.2-3)	C, D, E Note 2
	282'-6"	Leakage Transfer Pumps (WDL-P-9A; WDL-P-9B) (See Figure 4.2-3)	C, D, E Note 2
	282'-6"	20" Feedwater (FW) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 4 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF A D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Pressurizer Missile Shield Block	282'-6"	6" Emergency Feedwater (EF) Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
	282'-6"	29 1/4" O. D. Mainstream (MS) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
Polar Crane Load Test Assembly	347'-6"	None	
212 tons - nom.	305'	Air Coolers (AH-E-1C, AH-E-1D, AH-E-1E) (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)
	305'	Reactor Building Emergency Cooling - River Water (RR) System Piping (Unborated Water) (See Figure 4.2-2)	D, C, E (4.2.4 Item B)
	282'-6"	20" Feedwater (FW), System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
	282'-6"	6" Emergency Feedwater (EF), System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 5 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF A D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Polar Crane Load Test Assembly	282'-6"	29 1/4" O. D. Main Steam (MS) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
Reactor Missile Shield Blocks (4: R-1, R-2, R-3, R-4)	347'-6"	3" Decay Heat (DH) System Piping (See Figure 4.2-1) (PZR Aux. Spray)	C, D, E (4.2.4 Item 2)
40 Tons each	305'	14" Core Flood (CF) System Piping (See Figure 4.2-2)	C, D, E (Note 1)
	305'	Reactor Building Emergency Cooling - River Water (RR) (See Figure 4.2-2) (Unborated Water)	D, C, E (4.2.4 Item B)
	282'-6"	20" Feedwater (FW), System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
	282'-6"	29 1/4" O. D. Main Steam (MS) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 6 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF B D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Polar Crane Load Test Assembly	347'-6"	None	
212 Tons - nom.	305'	Air Coolers (AH-E-1A, AH-E-1B, AH-E-1C) (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)
	305'	Reactor Building Emergency Cooling - River Water (RR) System Piping (See Figure 4.2-2) (Unborated Water)	C, D, E (4.2.4 Item B)
	282'-6"	20" Feedwater, (FW) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)
	282'-6"	6" Emergency Feedwater (EF) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 7 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	SOUTH-WEST QUADRANT, SOUTH OF B D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Polar Crane Load Test Assembly	282'-6"	29 1/4" O. D. Main Steam (MS) System Piping (See Figure 4.2-3) (Unborated Water)	C, D, E (4.2.4 Item B)

TABLE 4.2-1 SHEET 8 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
IMPACT AREA / LOADS	INSIDE A D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Reactor Missile Shield Blocks	All	Reactor Coolant (RC) System inside D-ring	B,C,D,E (4.2.4 Item 1,2,A) Note 3
(4: R-1, R-2, R-3, R-4) 40 Tons each	All	Secondary Systems inside D-ring (Unborated Water)	C, D, E (4.2.4 Item B)
Pressurizer Missile Shield Block (P-1)	All	Reactor Coolant (RC) System inside D-Ring	B,C,D,E (4.2.4 Item 1,2,A) Note 3
33 Tons	All	Secondary Systems inside D-ring (Unborated Water)	C,D,E (4.2.4 Item B)

TABLE 4.2-1 SHEET 9 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

LOCATION	REACTOR BUILDING		
LOADS \ IMPACT AREA	INSIDE B D-RING		
	ELEVATION	EQUIPMENT	HAZARD ELIMINATION** CATEGORY (And Reference)
Reactor Missile Shield Blocks	All	Reactor Coolant (RC) System inside D-ring	B,C,D,E (4.2.4, Item 1,2, A)
(4: R-1, R-2, R-3, R-4) 40 Tons each	All	Secondary Systems inside D-ring	C,D,E (4.2.4 Item B)

** Hazard Elimination Categories are defined in the Generic Letter on the Control of Heavy Loads at Nuclear Power Plants as:

a. Crane travel for this area/load combination prohibited by electrical interlocks or mechanical stops. (Please refer to section 4.0 for a discussion)

b. System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations eliminate the need to consider load/equipment combination.

d. Likelihood of handling system failure for this load is extremely small (i.e. section 5.1.6 NUREG 0612 satisfied). (Please refer to section 4.0 for a discussion of failure probabilities.)

e. Analysis demonstrates that crane failure and load drop will not damage safety-related equipment. (Special Conditions for TMI-2 as stated in Section 4.2.4.)

TABLE 4.2-1 SHEET 10 OF 10

LOAD/IMPACT MATRIX

CRANE: REACTOR BUILDING POLAR CRANE MH-A-1

NOTE 1 This item is included as an example of systems and equipment which would be of importance to an operating power plant but, due to the special condition existing at TMI-2, this equipment is no longer important. In fact, both of these tanks are drained.

NOTE 2 These components no longer perform an important function but may contain water of an unknown boron concentration and are under the load path.

NOTE 3 Drop of a missile shield into the D-ring may damage the Make-Up and Purification (MU) system to which the Standby Pressure Control (SPC) System is connected. Specifically, the SPC system connects to two 2 1/2" MU lines in the Auxiliary and Fuel Handling Building. These MU lines enter the A D-ring on opposite sides and attach to the discharge of the corresponding reactor coolant pump. A single load drop is not postulated to simultaneously impact both trains trains of this piping.

TABLE 4.2-2

Potential Sources of Unborated Water

Reactor Building Fire Protection System

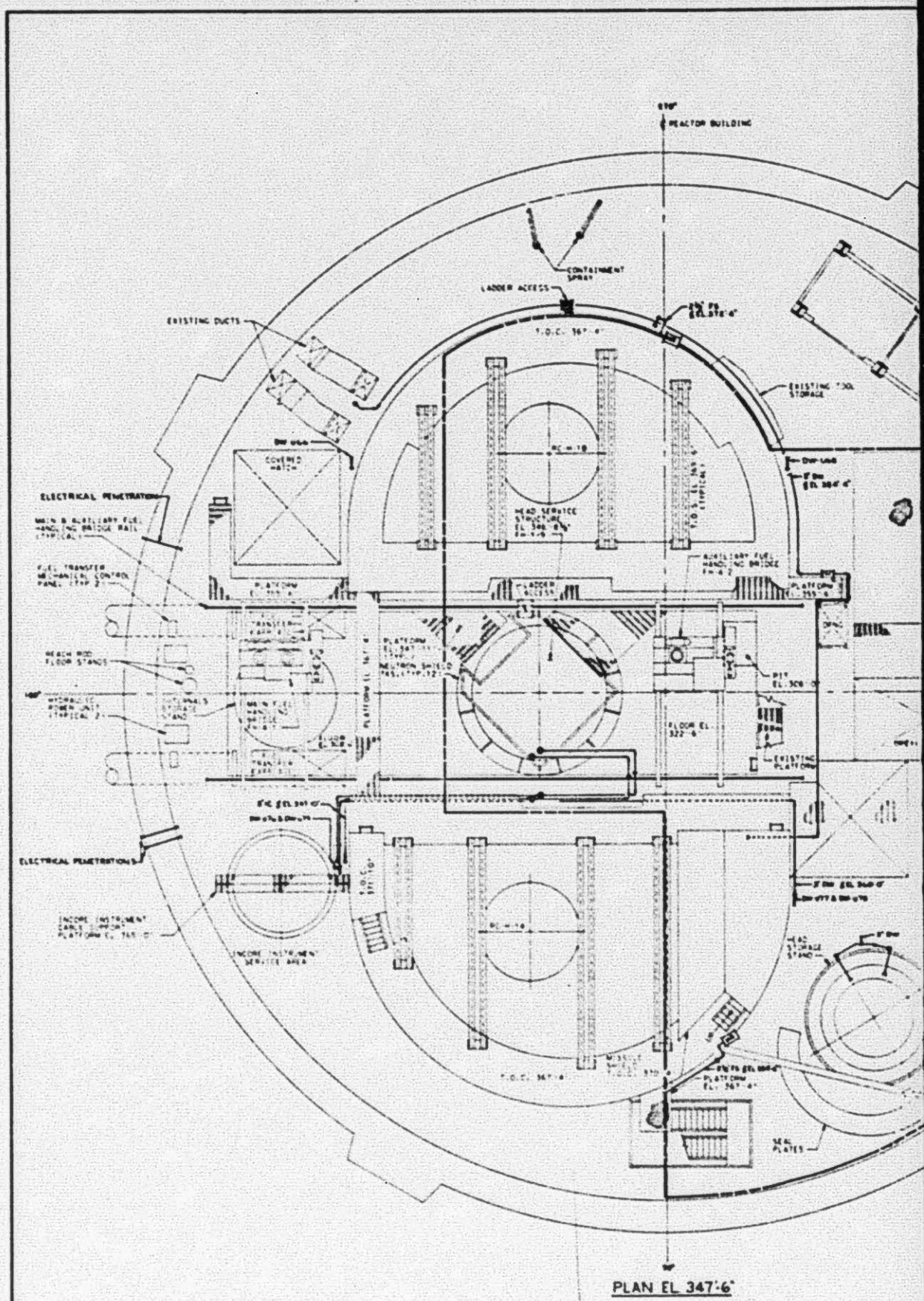
Reactor Building Demineralized Water System

Reactor Building Nuclear Services Closed Cooling Water System

Reactor Building Intermediate Closed Cooling Water System

Reactor Building Normal Cooling Water System

Once-Through Steam Generators (Secondary Sioe)


REACTOR BUILDING
CONTAINMENT SPRAY
LADDER ACCESS
EXISTING DUCTS
EXISTING TOOL STORAGE
COVERED HATCH
ELECTRICAL PENETRATION
MAIN & AUXILIARY FUEL HANDLING BRIDGE RAIL (TYPICAL)
FUEL TRANSFER MECHANICAL CONTROL PANEL (TYP 2)
HEAD SERVICE STRUCTURE
AUXILIARY FUEL HANDLING BRIDGE
LADDER ACCESS
PLATFORM
REACH ROD FLOOR STANDS
HYDRAULIC POWER UNIT (TYPICAL 2)
INTERNALS STORAGE STAND
MAIN FUEL HANDLING BRIDGE
NEUTRON SHIELD
FLOOR EL. 322'-6"
EXISTING PLATFORM
ELECTRICAL PENETRATIONS
INCORE INSTRUMENT CABLE SUPPORT PLATFORM EL. 765'-0"
INCORE INSTRUMENT SERVICE AREA
HEAD STORAGE STAND
SEAL PLATES

PLAN EL 347'-6"

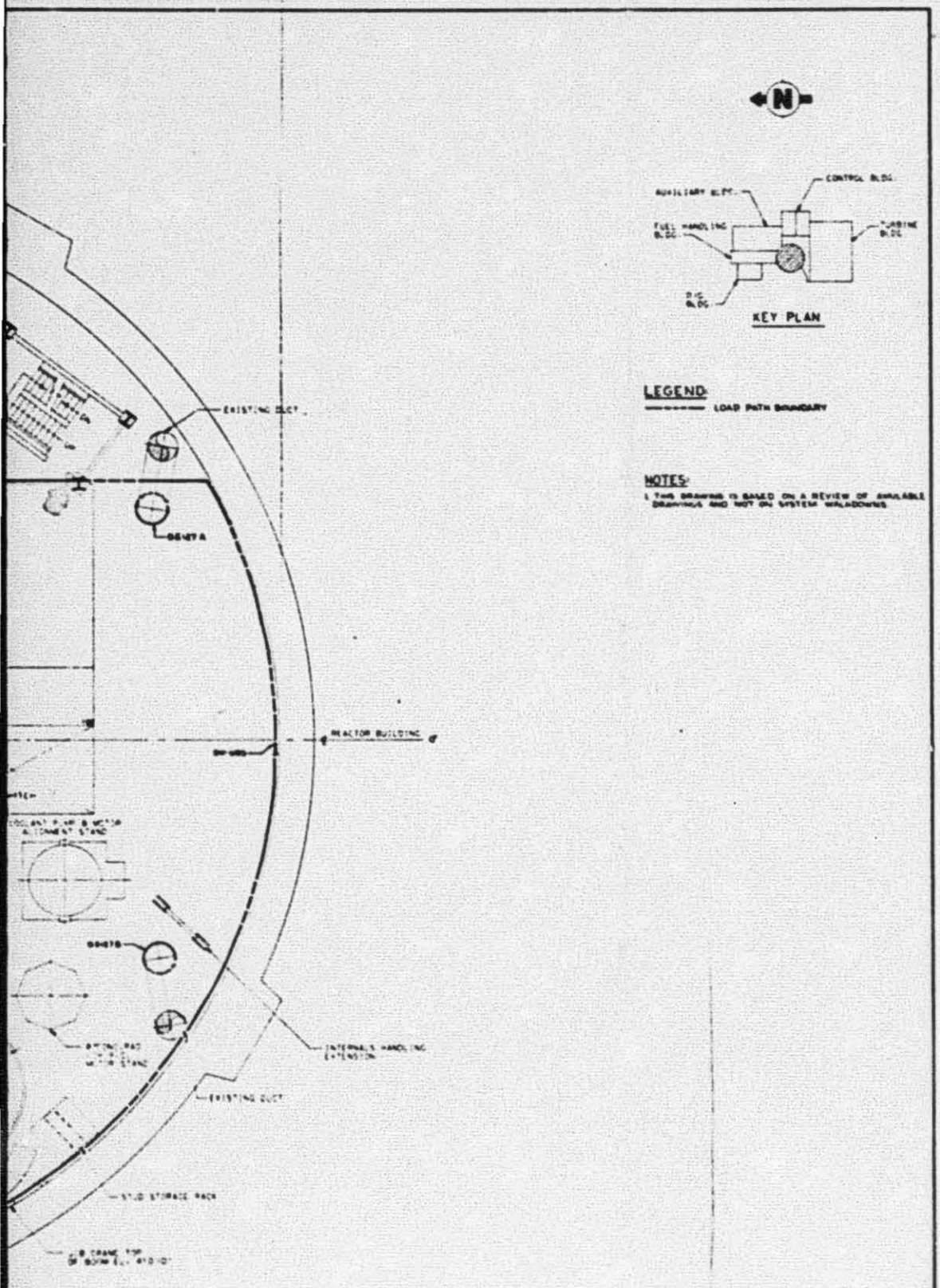
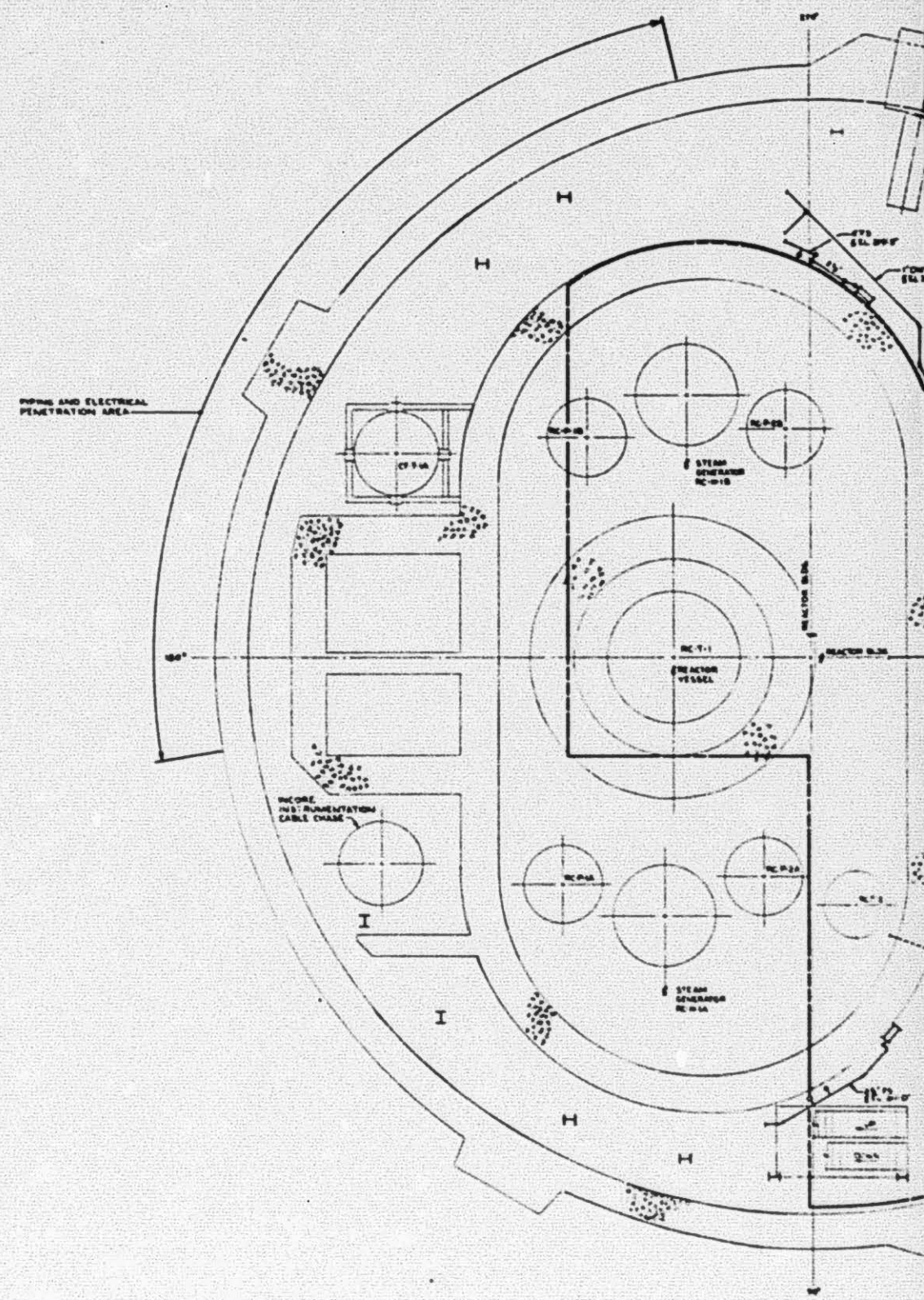


FIGURE 4.2-1
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 347'-6"


PIPING AND ELECTRICAL PENETRATION AREA
INCORE INSTRUMENTATION CABLE CHASE
REACTOR VESSEL
STEAM GENERATOR
STEAM GENERATOR

PLAN EL 305'-0"

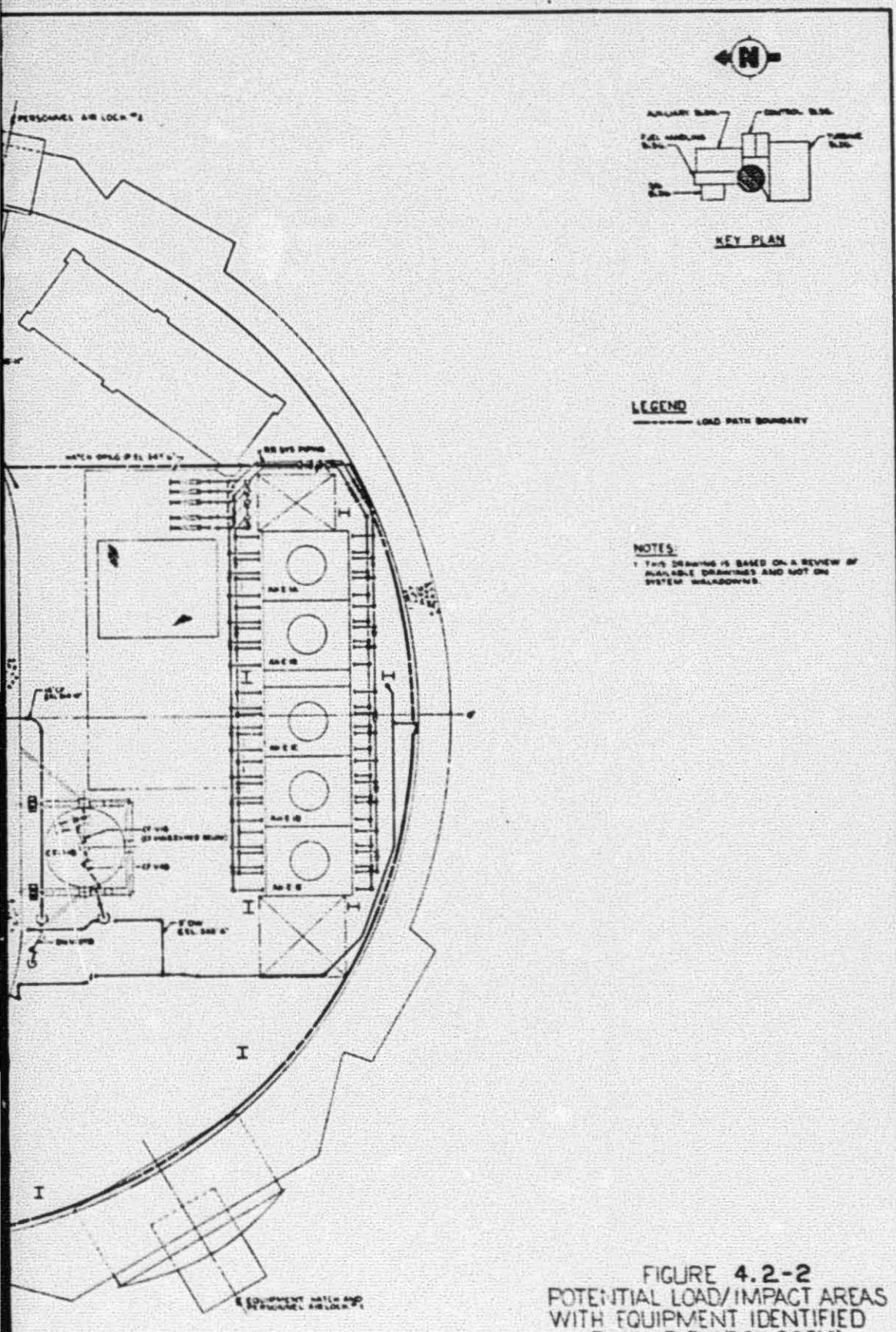
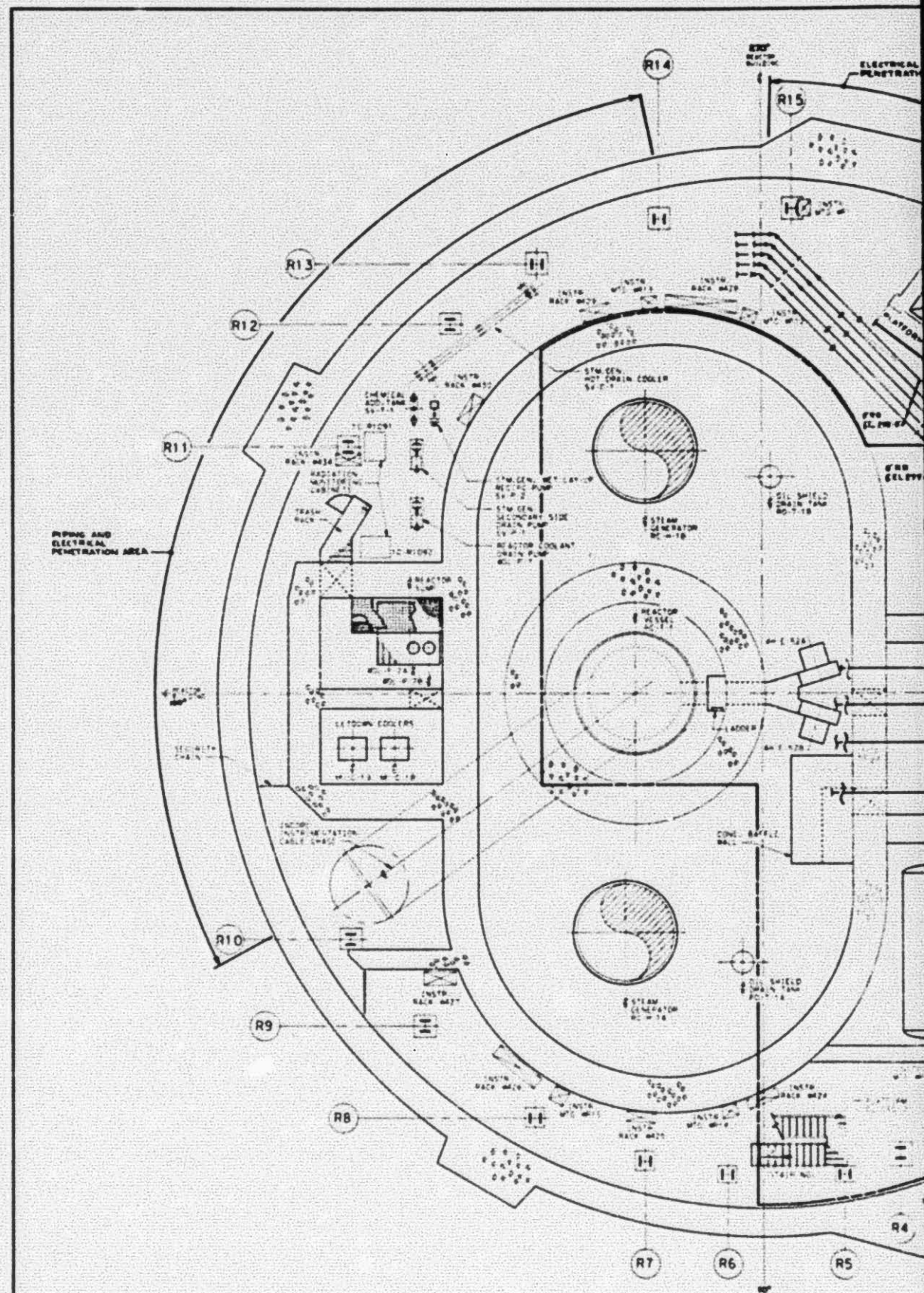


FIGURE 4.2-2
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 305'-0"


R14
R15
R13
R12
R11
R10
R9
R8
R7
R6
R5
R4
PIPING AND ELECTRICAL PENETRATION AREA
STEAM GENERATOR
REACTOR VESSEL
LADDER
LETDOWN COOLERS
CONC. BAFFLE WALL
STAIR NO.

PLAN EL. 282'-6"

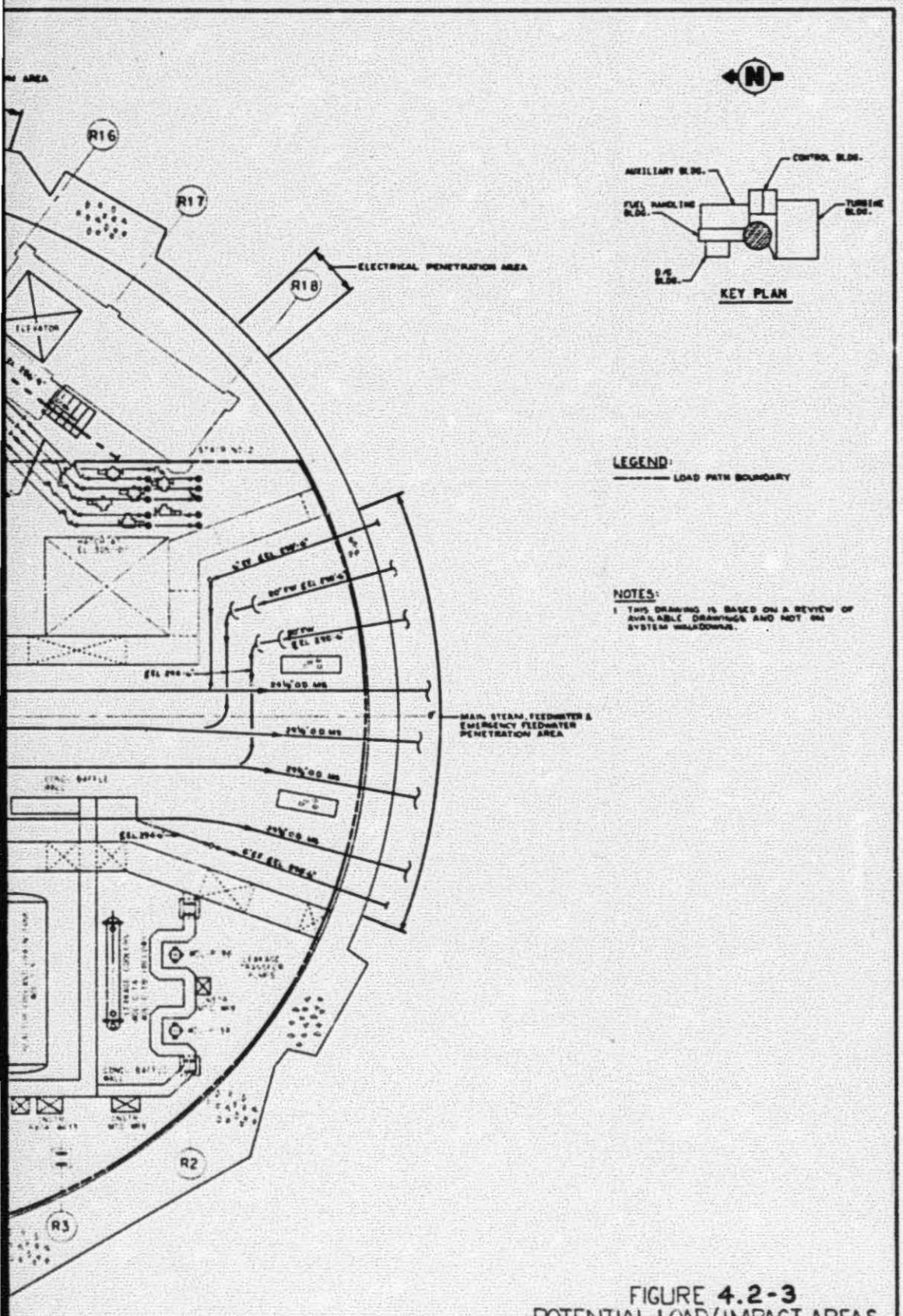


FIGURE 4.2-3
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 282'6"

5.0 10 CFR 50.59 EVALUATION FOR THE POLAR CRANE LOAD TEST

Changes, Tests, and Experiments, 10 CFR 50, paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test, or experiment is determined not to be an unreviewed safety question and does not involve a modification of the plant technical specifications.

A proposed change involves an unreviewed safety question if:

a) the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b) the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) the margin of safety, as defined in the basis for any technical specification, is reduced.

5.1 Due to the unique configuration of the TMI-2 reactor, i.e., a post-accident recovery mode situation, case (a) above is no longer strictly applicable. Accidents analyzed in the SAR are predicated upon an initial configuration being one of the normal operational modes as allowed by technical specification for power plants with an operating license.

Nevertheless, accidents presented and analyzed in the SAR address events which bound by a wide margin the potential consequences of any planned or postulated unplanned event that might occur in connection with the proposed test.

5.2 Case (b), above, more directly applies to the special situation at TMI-2, and an evaluation in this regard must focus attention on the following areas directly related to the maintenance of a safe and stable configuration of the reactor and its required support systems.

1) Criticality control

The reactor core is being maintained in a stable subcritical condition by virtue of the soluble poison (boron) present in high concentration in the reactor coolant. Since the activities associated with the load test as stated in section 2 do not in any way utilize or disturb systems which could effect the boron concentration of the reactor coolant, it is clear that subcriticality will be maintained. Postulated load drop accident scenarios and an evaluation of their potential effects on criticality are presented in section 4 with the conclusion that subcriticality will be maintained under all credible conditions.

2) Decay heat removal

The decay heat production rate has diminished greatly during the time since the accident to a point at which active heat removal systems are no longer required (and in fact have not been required for some time). The present mode of decay heat removal is by natural losses to ambient via the reactor coolant system with the main pumps idle. Recent analyses have shown that the decay heat rate is now so small that losses to ambient could be accomplished without undesirable sequelae (such as boiling) with a level of coolant in the reactor vessel lowered to the elevation of the bottom of the cold leg nozzles. In other words, decay heat removal could be maintained even with the main loop cold leg piping sheared off at the vessel nozzles.

Notwithstanding the above, planned activities of the load test do not include manipulation or use of any system associated with the maintenance of decay heat removal capability and, as elucidated in section 4, no credible unplanned occurrence could result in a loss of this capability.

3) Confinement of radioactive material

The mechanism for confinement of radioactive material presently consists of two major components: the physical barrier of the containment building and the lack of an energy source capable of moving radioactive material across this barrier.

The reactor coolant system and especially the vessel itself also contribute to confinement but more in the sense of preventing further escape and dispersion within the containment building.

As may be seen from section 2, no planned activity of the load test involves breaching the physical barrier of the containment or providing a source of energy capable of transporting radioactive material across this boundary and further, as presented in section 4, no postulated unplanned occurrence yields a credible mechanism by which confinement of radioactive material within the containment could be compromised.

4) Radiation shielding

Radiation shielding is presently composed of major contributions from 1) steel in the reactor vessel, 2) concrete and steel in the primary shield and D-rings, and 3) concrete and steel in the containment wall.

No planned activity or consequence of any credible unplanned occurrence associated with the load test has the potential to degrade major shielding components to a degree at which their shielding function would be negated. This is based on engineering judgement resulting from a comparison of load/target relative weights and strengths, as well as a review of load paths and

credible impact orientations. For example, it is not considered credible that a missile shield composed of concrete and rebar could penetrate the 8 inch thick steel reactor vessel head when falling from the lift height used in the load test.

5.3 A careful review of the bases (Section B 3/4) for the TMI-2 technical specification has been conducted. Results of this investigation show that none of the planned activities associated with the proposed test results in diminution of safety margins stated in these bases.

5.4 The information presented in the above subsections, taken in the aggregate, demonstrates that the proposed test does not constitute an unreviewed safety question and does not involve a modification of the plant technical specifications.

6.0 RADIOLOGICAL CONSIDERATIONS

6.1 OCCUPATIONAL EXPOSURE

All individuals entering the reactor building will be monitored for external exposure in accordance with GPU Radiological Control Procedures (RCP) to ensure personnel exposures are maintained within 10 CFR 20 dose equivalent guidelines. Administrative control points in accordance with GPU Procedures will be used in order to assure specified dose limits are not exceeded. Extremity monitoring will be performed in accordance with existing procedures.

The total exposure for the polar crane load test is estimated to be 30 to 90 man-rem. This is based upon the scope defined in Sections 1 and 2 of this safety evaluation, which results in an estimated in-containment man-hour figure of 540.

Because of the uncertainty in the dose rates and man-hours, the man-rem for the activities are estimated to vary by $\pm$ 50 percent. Considering the uncertainties associated with the man-rem estimate, 30 to 90 man-rem has been selected to be used as the estimate for the polar crane load test.

Personnel entering the reactor building will be protected against the inhalation of particulate radioactivity in accordance with Radiological Control Procedures.

As specified by Radiological Control Procedures analyses of expected airborne contamination levels will be performed in order to select appropriate respiratory protective devices.

Air sampling for particulate activity will be performed using devices such as lapel samplers and methods such as grab samples. Tritium air samples will also be taken as required.

6.2 ENVIRONMENTAL RELEASES

Since the polar crane load test does not involve the use of any system containing radioactivity and since containment integrity will be set and maintained throughout the test, no release of radioactivity to the environment is expected.

However, postulated occurrences identified and evaluated in Section 4 may result in some slight release. Since the release pathway to the environment is through the containment boundary, these postulated releases would be strictly controlled such that they would be bounded by the release estimate presented in Reference h.

7.0 SUMMARY AND CONCLUSION

The analyses, investigations, and other information contained in this SER show that:

- The polar crane has undergone an extensive refurbishment and inspection program and is ready in all respects to be finally load tested. Indeed, the maximum load to be lifted is less than one half of the originally designed capability of the crane.

- Special attention has been given to the hardware and software to be used in the load test itself, thus rendering the probability of a load drop extremely small.

- Notwithstanding the low probability of a load drop in the first place, an analysis has been performed which shows that the probability of unacceptable consequences arising from postulated load drop accident scenarios is extemely small as well.

- Occupational radiation exposure to personnel conducting the load test will be maintained as low as reasonably achievable.

- Environmental releases of radioactivity will not occur in connection with the planned test.

- The polar crane load test does not constitute an unreviewed safety question as stated in 10 CFR 50.59.

In view of the evaluation components summarized above, it is the conclusion of this SER that the polar crane load test may proceed without presenting undue risk to the health and safety of the public.

8.0 REFERENCES

(a) NRC letter of February 27, 1981 containing Generic Letter 81-07 regarding control of heavy loads.

(b) Polar Crane Load Test Procedure - UWI 4370-3891-83-PC0001.

(c) GPU letter 4410-82-L-0021 of October 8, 1982 containing the Polar Crane Functional Description and Maintenance Checklist as modified.

(d) GPU letter 4410-83-L-0004 of January 4, 1983 containing the Polar Crane Head Lift Rigging description of requalification.

(e) GPU letter 4410-83-L-0007 of January 12, 1983 containing additional clarification to the Polar Crane Maintenance Checklist.

(f) GPU letter 4400-82-L-0110 of July 6, 1983 containing "Methods and Procedures of Analysis for TMI-2 Criticality Calculations to Support Recovery Activities through Head Removal: as an attachment to the CRDM - Quick Look Safety Evaluation.

(g) NUREG-0557, Safety Evaluation and Environmental Assessment, Three Mile Island, Unit 2, February 11, 1980.

(h) GPU letter 4410-82-L-0007 of September 23, 1982, containing the Safety Evaluation Report for Ongoing Decontamination Activities.

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