FEB 5 1992

Docket No. 50-285
License No. DPR-40

Omaha Public Power District
ATTN: W. G. Gates, Division Manager
Nuclear Operations
444 South 16th Street Mall
Mail Stop 8E/EP4
Omaha, Nebraska 68102-2247

Gentlemen:

This refers to the management meeting conducted at Region IV's request in Omaha, Nebraska, on January 28, 1992. This meeting related to activities authorized by NRC License DPR-40 for Fort Calhoun Station and was attended by those on the attached Attendance List.

The subjects discussed at this meeting are described in the enclosed Meeting Summary.

It is our opinion that this meeting was beneficial and has provided a better understanding of your management controls planned for the refueling outage. In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter will be placed in the NRC's Public Document Room.

Should you have any questions concerning this matter, we will be pleased to discuss them with you.

Sincerely,

Original Signed By:

L. A. Yandell for

A. Bill Beach, Director
Division of Reactor Projects

Enclosure:
Meeting Summary w/attachments

cc w/enclosure:
LeBoeuf, Lamb, Leiby & MacRae
ATTN: Harry H. Voigt, Esq.
1875 Connecticut Avenue, NW
Washington, D.C. 20009-5728

*RIV/C:DRP	D:DRP
PHHarrell;bh	ABBeach
2/ /92	2/4/92

IE45

*Previously concurred

9202120048 920205
PDR ADOCK 05000285
P PDR

Washington County Board
of Supervisors
ATTN: Jack Jensen, Chairman
Blair, Nebraska 68008

Combustion Engineering, Inc.
ATTN: Charles B. Brinkman, Manager
Washington Nuclear Operations
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Nebraska Department of Health
ATTN: Harold Borchert, Director
Division of Radiological Health
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509

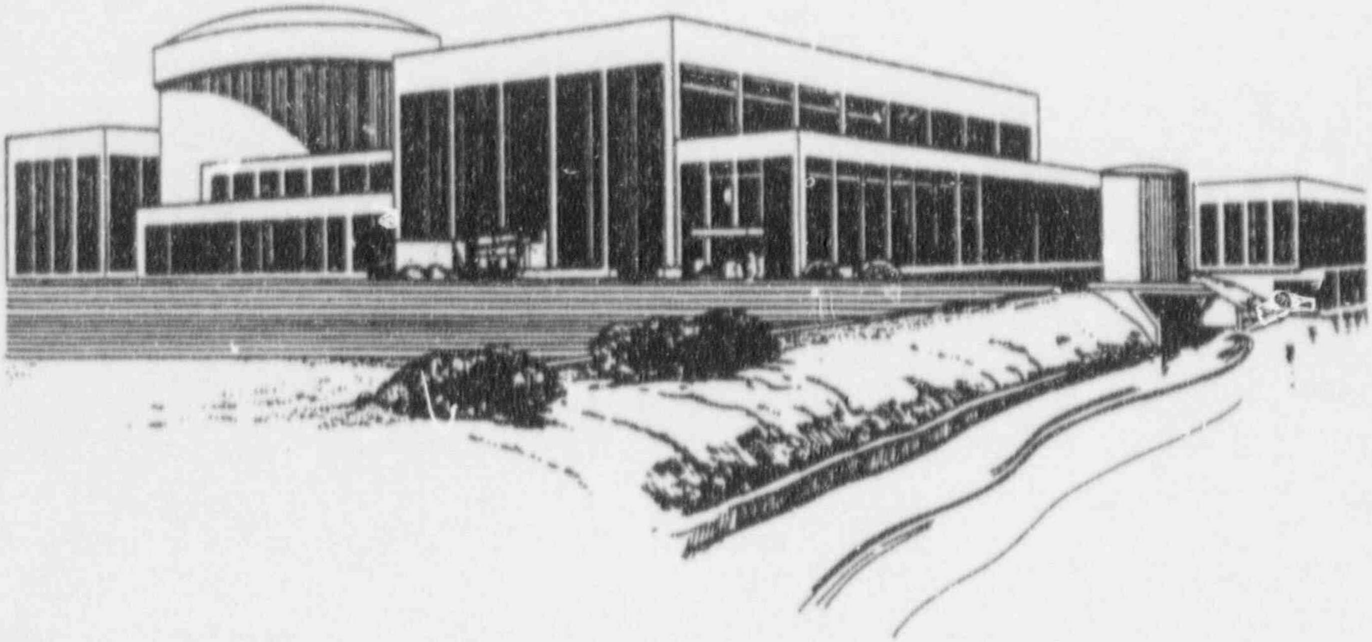
Fort Calhoun Station
ATTN: T. L. Patterson, Manager
P.O. Box 399
Fort Calhoun, Nebraska 68023

bcc to DMB (IE45)

bcc distrib. by RIV:

R. D. Martin
DRSS-RPEPS
RIV File
DRP
Project Engineer (DRP/C)
Senior Resident Inspector - Cooper
Senior Resident Inspector - River Bend
Resident Inspector
Section Chief (DRP/C)
Lisa Shea, RM/ALF
DRS

ATTACHMENT 2



OMAHA PUBLIC POWER DISTRICT
SHUTDOWN PLANT ISSUES MEETING
JANUARY 28, 1992

AGENDA FOR NRC / OPPD SHUTDOWN PLANT ISSUES MEETING

ENERGY PLAZA ATRIUM - JANUARY 28, 1992

INTRODUCTIONS

- Opening Remarks:	W. C. Jones

SHUTDOWN PLANT RISK MANAGEMENT

- OPPD Philosophy:	W. G. Gates
+ Shutdown Risk Lessons + Nuclear Policy 2.03	
- Specific Implementation:	T. L. Patterson
+ Planning + Procedural Guidance	
- Training	J.K. Gasper
- 1992 Refueling Outage:	J. W. Chase
+ Outage Management + Schedule/Milestones	
- Conclusion	W. G. Gates

LUNCH

AGENDA FOR NRC / OPPD SELECT TOPICS

ENERGY PLAZA ATRIUM - JANUARY 28, 1992

OVERVIEW	S. K. Gambhir
- Steam Generators:	J. M. Cate
- Thermal Shield Inspection/ Repair:	C. E. Boughter
- Reactor Vessel Inspections:	C. N. Bloyd
- System Report Card	R. L. Jaworski
- Pressurized Thermal Shock:	K. C. Holthaus
- PRA Level I:	H. A. Hackerott
- Total Quality Advantage:	M. A Ferdig

SHUTDOWN RISK LESSONS LEARNED

- Loss of shutdown cooling events continue to occur even though there have been over 44 documented Operating Experience lessons provided to the industry.

- There have been 74 loss of AC events during shutdown.

- There have been 52 loss of DHR events during midloop operations.

- Risks for core damage while shutdown can represent a substantial portion of the total core melt frequency.

NUCLEAR POLICY 2.03 - "SAFETY DURING SHUTDOWN"

ESTABLISHES CORPORATE POLICY

- Shutdown Safety Principles Defined

- Improved Outage Planning and Control is Key

- Independent

OUTAGE PRINCIPLES

- Minimized Vunerability
  + Maximize Power Supply Availability, Especially at Mid-Loop
  + Ensure Availability of DHR Capability

- Integrity of Fission Product Barriers

- Careful Adherence to Fuel Handling Procedures

SYSTEM WINDOW CONTROL

- Ensures Critical Equipment Redundancy

- At Reduced RCS Inventory
  + Both DG's Available
  + Two DHR Pumps, Two SDC Heat Exchangers available when RCS Level is 3 ft below Vessel Flange

OPPD NUCLEAR POLICY MANUAL

FUNCTION	Safety Assessment/Quality Verification	NUMBER	2.J3
SUBJECT	Safety During Shutdown	DATE	December 20, 1991
SUPERSEDES		PAGE	1 OF 2

ISSUED BY W. C. Jones APPROVED BY W C Jones

The Omaha Public Power District (OPPD), as the owner and operator of Fort Calhoun Station, has established a nuclear policy to maintain and operate the facility with due regard for public and plant safety.

Awareness of shutdown concerns is a prerequisite to enhancing shutdown safety. Vulnerabilities that certain systems and components have under shutdown plant conditions can challenge safety during shutdown such as loss of AC power and loss of decay heat removal. During refueling outages maintenance and surveillance activities can require the simultaneous opening of primary systems and/or containment, cessation of shutdown cooling, disabling of electrical systems or components, and movement of heavy equipment. Industry analysis of shutdown events (NUMARC "Guidelines to Enhance Safety During Shutdown") has "concluded that improved outage planning and control is the most effective means of reducing the likelihood and consequences of events during shutdown. The coordination of these activities with the objective to manage risk and maintain key safety functions is essential and goes beyond compliance with technical specifications requirements during shutdown."

OPPD nuclear management will regularly and critically assess current practices for planning and conducting outages. OPPD management is dedicated to preventing such events by providing management attention; ensuring adequate training; demanding compliance with procedures; effecting detailed planning, coordination and execution of operations; and managing risk. While the scope of activities for an unplanned or forced outage is far less than that of a refueling outage, the same awareness of vulnerabilities during shutdown conditions is required to safely conduct the outage. In addition outage scopes will be closely controlled; schedules will be fully planned, reviewed, and approved; an independent safety analysis of approved refueling schedules will be conducted; and activities affecting nuclear safety will be performed according to the approved schedules. Changes to the schedule or activities that were the basis for the safety evaluation will be re-reviewed and approved by appropriate personnel.

During refueling outages at Fort Calhoun Station, system, train and equipment outages are scheduled according to the System Window concept. The System Window concept takes into account the redundancy of safety systems, electrical power distribution circuits and fire detection and protection requirements in order to ensure the Station's ability to maintain the reactor in a safe shutdown condition.

At times when reactor vessel water inventory is reduced and fuel is in the vessel both Diesel Generators will be available for operation (if needed) and at least two pumps and both Shutdown Cooling Heat Exchangers for decay heat removal will be available prior to reducing water level to less than 3 feet below the vessel flange. All planned evolutions that require this reduced inventory condition will be controlled administratively via plant procedures.

In addition, the outage shall be conducted in accordance with the following principles:

- The periods of high vulnerability should be minimized. The availability of on-site and off-site electrical power supplies shall be maximized, particularly during periods of increased vulnerability to fuel damage, such as during mid-loop operation or periods of high decay heat generation prior to refueling cavity flood.

- The availability of systems required to provide reactor vessel make-up water and decay heat removal capability, including contingency plans for alternate cooling methods, shall be carefully controlled consistent with decay heat generation rate.

- The integrity of fission product containment barriers shall be controlled, consistent with outage activities, to minimize potential for unintentional radioactive releases.

- Fuel handling operations shall be conducted carefully in strict compliance with procedures. Special emphasis will be given to precautions to prevent fuel and other core component mishandling or damage.

PLANNING

BASED ON SYSTEM WINDOW CONCEPT

- Maintain 5 Key Safety Functions

    + Containment Integrity
    + Inventory Control
    + Decay Heat Removal
    + Power Availability
    + Reactivity Control

INDEPENDENT REVIEW

- Nuclear Safety Review Group conducted independent review of schedule. Comments have been addressed in schedule.

- Operations Department also conducted schedule review. Comments have been addressed in schedule.

PLANNING (CONTINUED)

OPERATIONAL EXPERIENCE INPUT FOR RISK MANAGEMENT

- FCS LER'S

89-001	SDC flow was terminated approx. 5 minutes as a result of closure of HCV-347&348 while performing CP-SCMM-A.
90-006	DG-SI pump interlock prevented DG from loading immediately. DG manually locked on within 44 seconds.

- NRC GENERIC LETTERS, BULLETINS AND INFORMATION NOTICES

- NUREG 1410

- INPO SOER'S

- NUMARC 91-06

PROCEDURAL GUIDANCE
FOR SHUTDOWN OPERATIONS

OP-1	Master Checklist for Start-Up or Trip Recovery.
OP-6	Hot Shutdown to Cold Shutdown or Refueling Conditions and Conduct of Shutdown Cooling Operations.
OP-11	Reactor Core Refueling.
OI-SC-1 Through OI-SC-6	Covers Shutdown Cooling Using Normal and Alternate Configurations.
OI-SFP-1 Through OI-SFP-4B	Covers Spent Fuel Cooling Using Normal and Alternate Cooling Configurations.
OI-SFP-5	Alternate Spent Fuel Pool Cooling
OI-SFP-6	SFP Heat-Up Rate

ABNORMAL / EMERGENCY PROCEDURES

AOP-11	Loss of Component Cooling Water
AOP-18	Loss of Raw Water
AOP-19	Loss of Shutdown Cooling
AOP-32	Loss of 4160/480V Bus Power
EOP-7	Station Blackout

ADMINISTRATIVE GUIDANCE FOR SHUTDOWN OPERATIONS

Nuclear Policy 2.03	Safety During Shutdown
S. O. M-104	Outage Planning and Execution
S. O. G-87	Non-Routine Activities Requiring Formalized Plans
S. O. G-92	Conduct of Infrequently Performed Procedures
Switchyard Charter	1992 Refueling Outage Switchyard Activities

OPERATOR TRAINING

LICENSED OPERATORS:

- Shutdown Risk Management
  + Reduced RCS Inventory
  + Containment Closure
  + Station Blackout
  + Loss of Shutdown Cooling
  + Vulnerabilities while Shutdown
- Loss of Shutdown Cooling Simulator Exercise (AOP-19)
- Raw Water Malfunctions (AOP-18)
- Reactivity Control

NON-LICENSED OPERATORS:

- Loss of Shutdown Cooling
- Nozzle Dam Mockup
- Diesel Generator Local Operation
- Reactivity Control
- Containment Integrity

POST MODIFICATION TRAINING

MAINTENANCE TRAINING

CONDUCT OF MAINTENANCE:

- On-site and Off-site Personnel

- Equipment Isolation and Tagouts

- Shutdown Risks
  + Loss of Off-site Power
  + Shutdown Cooling
  + System Interdependency
  + Control Panel Access

- Self Checking and Verification

CEDM COUPLING AND UNCOUPLING

NOZZLE DAM MOCKUP

FREEZE SEAL MOCKUP

CONFIGURATION CONTROL

OTHER TRAINING

EMERGENCY PLAN TRAINING

- January 8, 1992, Quarterly Drill

GENERAL EMPLOYEE TRAINING

- Outage Manager Briefings

ON SHIFT TRAINING

- Supervisor Discussions
- Pre-Shift Briefings
- Pre-Job Briefings

OUTAGE MANAGEMENT

ORGANIZATION
- Organization Chart
- Extra Shift Supervisor

OUTAGE CONTROL CENTER
- Personnel
- Status Boards
  + Outage Control Center
  + Control Room

SWITCHYARD COORDINATION
- Dedicated Switchyard Coordinator

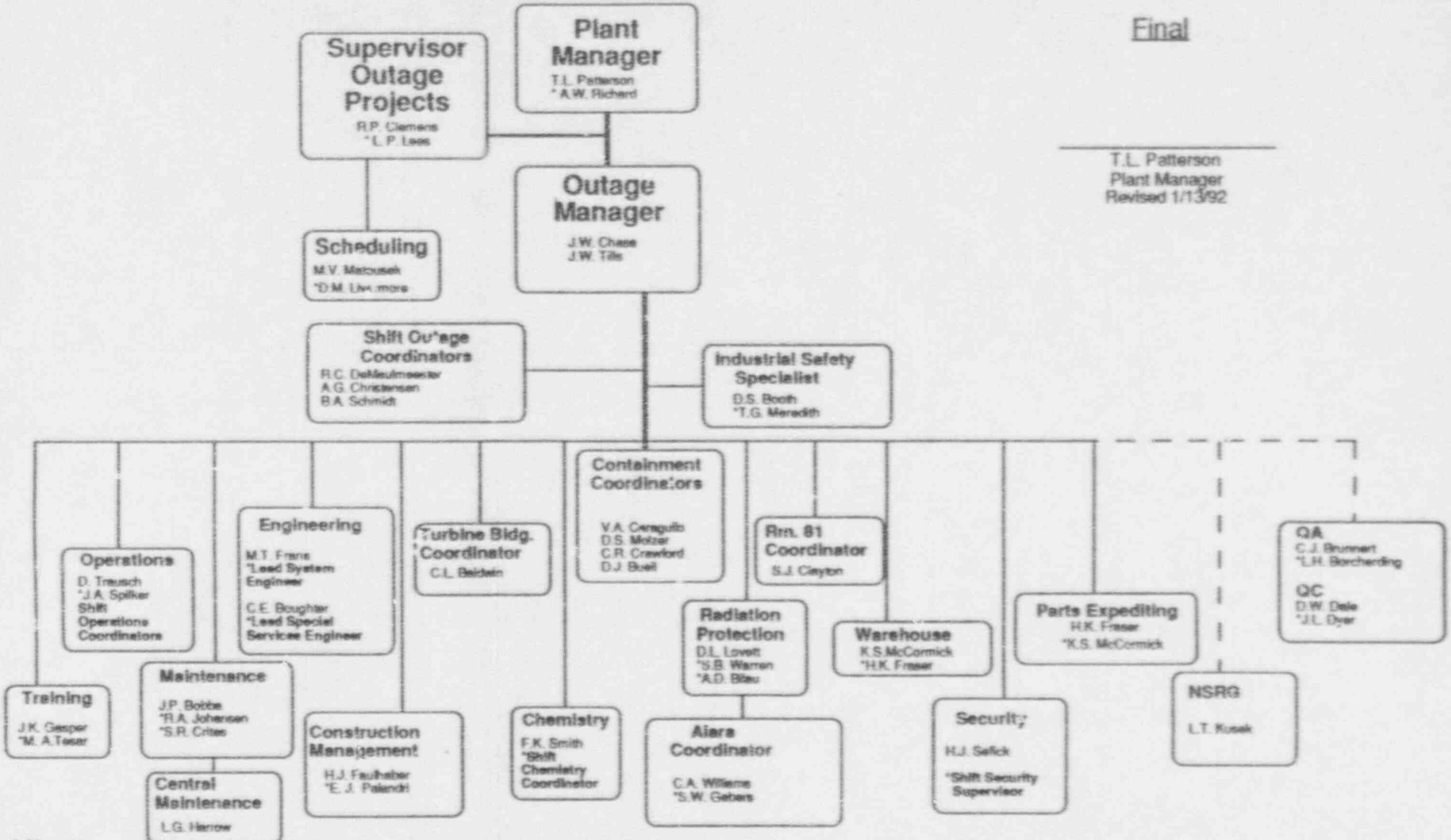
WORK FLOW AND CONTROLS
- Plan of the Day
  + Critical Systems Status
- System Window Concept
  + Scheduled Work
  + Emergent Work
- Basis for Schedule (Scope) Changes
  + Safety
  + Reliability
- Pre-job Briefings
- Pre-shift Briefings

MISCELLANEOUS
- 8 Hours at Mid-Loop
- 86 Day Outage
- Approximately 400 Personnel Augmenting Normal Plant Staff

1992 Ft. Calhoun Refueling Outage Organization

Final

T.L. Patterson
Plant Manager
Revised 1/13/92



* Alternate

Shift Operations Coordinator
D.J. Bannister
J.A. Drahota

Lead System Engineers
K.R. Henry
M.W. Butt
D.C. Gorence
J.L. Connolley

Lead Special Services Engineers
C.N. Bloyd
J.C. Knight
C.W. Norris
S.F. Swearngin

Shift Chemistry Coordinator
R.R. Henning
T.R. Dukarski

Security Shft Supervisors
W.L. Lawson
R. Undajon
G.W. Dempsey
W.L. Groves
S.A. Bremmerkamp

DATE __/__/__ TIME _____

CRITICAL SYSTEM STATUS

POWER SUPPLY

	Inservice Y	Inservice N
345KV	____	____
161KV	____	____

	Operable Y	Operable N
DG-1	___	___
DG-2	___	___

DECAY HEAT REMOVAL

	Inservice	Operable
LPSI Pumps		
SI-1A	______	______
SI-1B	______	______
Containment Spray Pumps		
SI-3A	______	______
SI-3B	______	______
SI-3C	______	______
SDC Heat Exchangers		
AC-4A	______	______
AC-4B	______	______

CORE LOADED __________

CORE OFFLOADED ______

	Inservice	Operable
SFP Cooling Pumps		
AC-5A	_______	_________
AC-5B	_______	_________
CCW Heat Exchangers		
AC-1A	_______	_________
AC-1B	_______	_________
AC-1C	_______	_________
AC-1D	_______	_________
CCW Pumps		
AC-3A	_______	_________
AC-3B	_______	_________
AC-3C	_______	_________

RW Pumps

AC-10A _______

AC-10B _______

AC-10C _______

AC-10D _______

INVENTORY CONTROL

	Operable Y	Operable N
Charging Pumps		
CH-1A	____	____
CH-1B	____	____
CH-1C	____	____
HPSI Pumps		
SI-2A	____	____
SI-2B	____	____
SI-2C	____	____

On-Line

SIRWT _________

BAST A _________

BAST B _________

Containment Integrity Required

Y N

Remarks: ________________________

____

____

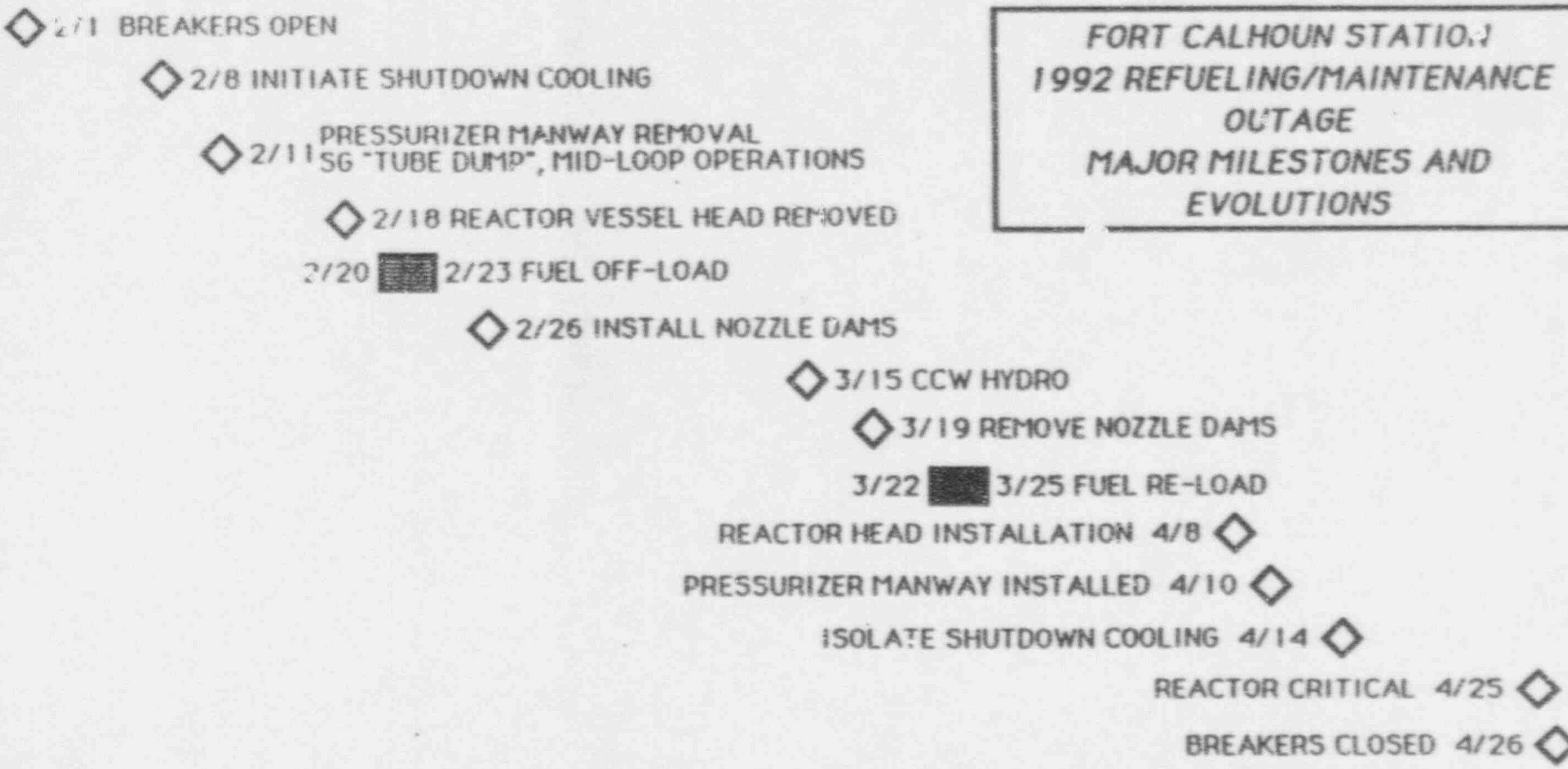
____

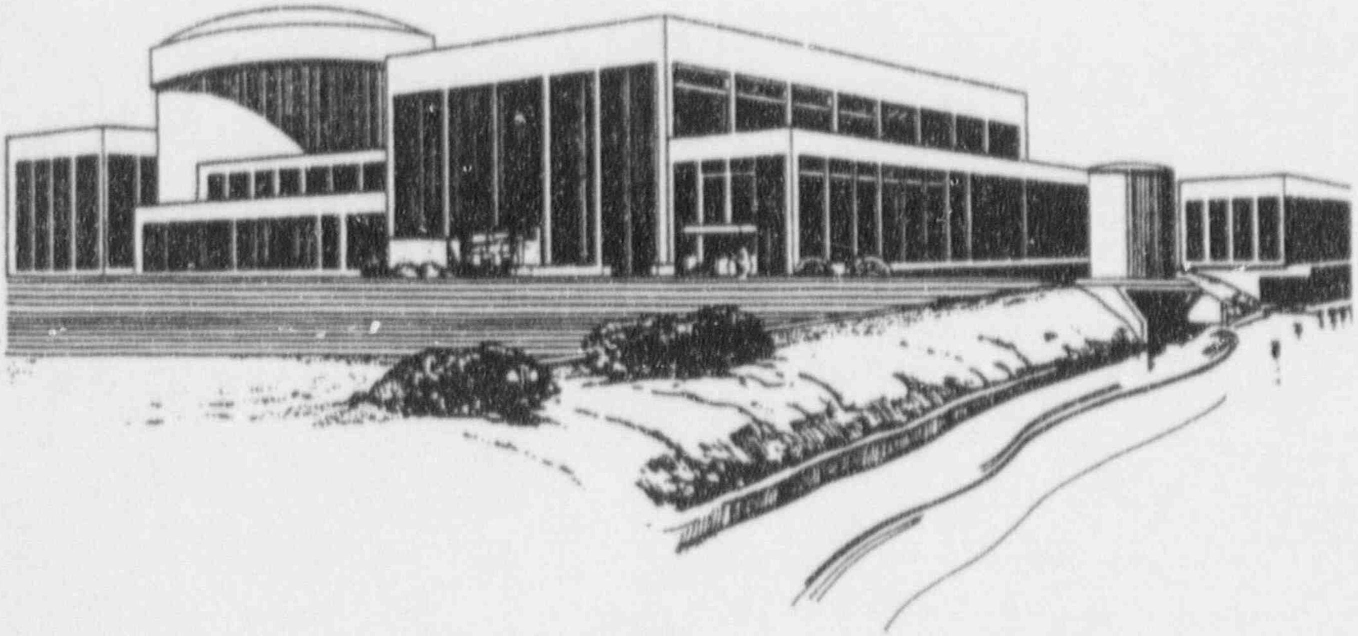
____

____

____

FORT CALHOUN STATION
1992 REFUELING/MAINTENANCE OUTAGE
MAJOR MILESTONES AND EVOLUTIONS





OMAHA PUBLIC POWER DISTRICT
SELECT TOPICS
JANUARY 28, 1992

AGENDA FOR NRC / OPPD SELECT TOPICS

ENERGY PLAZA ATRIUM - JANUARY 28, 1992

OVERVIEW	S. K. Gambhir
- Thermal Shield Inspection/Repair	C. E. Boughter
- Steam Generators:	J. M. Cate
- Reactor Vessel Inspections:	C. N. Bloyd
- System Report Card	R. L. Jaworski
- Pressurized Thermal Shock:	K. C. Holthaus
- PRA Level I:	H. A. Hackerott
- Total Quality Advantage:	M. A Ferdig

FORT CALHOUN STATION
1992 REFUELING/MAINTENANCE OUTAGE
PRODUCTION ENGINEERING PROJECT LIST

Special Services Engineering

1* Steam Generator Services (ECT, Sludge Lancing, Tube Plugging, Secondary Side Visual Inspections)
2* Reactor Vessel Thermal Shield Inspection
3* Reactor Vessel ISI Exams
4 ISI Exams
5 BOP Eddy Current Testing (Low Pressure Feedwater Heaters, High Pressure Feedwater Heaters, Drain Coolers, Condensers)
6 Motor Operated Valve (MOV) Testing
7 Snubber Maintenance and Testing
8 Check Valve Inspections
9 Pressurizer Sludge Inspection
10 Relief Valve ISI Testing
11 Erosion/Corrosion Inspections/Repairs

Hydrostatic Tests

12 SS-ST-CCW-3001

System Engineering

13 Diesel Generators 1 & 2
14 Electrical Distribution System Maintenance
15 Condenser Valve Repairs/Circulating Water Outage
16 LP Turbine Overhaul
17 ESF Testing
18 Main Generator Rotor Balancing

Nuclear Construction Management

19 Radiography

* indicates presentation topic

Fort Calhoun Station
1992 Refueling/Maintenance Outage
Modifications List

1 MR-FC-84-176
Letdown and Backpressure Controls

2 MR-FC-87-008
Annunciator Upgrade

3 MR-FC-87-014
Replacement of HCV-249 and HCV-2988

4 MR-FC-88-017
Addition of a Third Auxiliary Feedwater Pump

5 MR-FC-88-064
Install Fans to Inverters A/B/C/D and 1/2

6 MR-FC-88-076
Relay 94/1045 Contacts 7-8, 9-10 Configuration Change

7 MR-FC-89-013
Replacement of 480V Breaker Trip Devices

8 MR-FC-89-019
Shutdown Cooling Low Flow Alarm

9 MR-FC-89-048
Instrumentation for CH-4A and CH-4B

10 MR-FC-89-055
Auxiliary Feedwater Pump Instrumentation

11 MR-FC-89-074
Electrical Changes to Charging Pumps

12 MR-FC-89-076
Boric Acid Concentration Reduction

13 MR-FC-89-081
FW-10 Steam Supply Line Break Protection

14 MR-FC-90-003
TE-601 Containment Sump Penetrations

15 MR-FC-90-005
DG Instrumentation Upgrade

16 MR-FC-90-023
161KV System Modifications

17 MR-FC-90-024
LPSI Pump Low Voltage Trip Interlock

18 MR-FC-90-026
Raw Water Discharge Valve Replacement

19 MR-FC-90-038
Main Steam and Main Feedwater Supports in Room 81

20 MR-FC-90-047
Pipe Restraints RCH-32 and RCH-33

21 MR-FC-90-060
SI Relief Valves, Flanged Connections

22 MR-FC-90-061
On-Line CECOR

23 MR-FC-90-062
Thermal Shield Locking Collar Replacement

24 MR-FC-90-063
Diesel Generator Room HVAC Control

25 MR-FC-90-067
FW-8C Loadshed Following OPLS

26 MR-FC-90-071
(90% complete during September, 1991 battery outage)
CEA Change Machine Removal

27 MR-FC-91-008
Undervoltage Protection for 480V Safety Related Motors

28 MR-FC-91-013
RPS Delta-T Power Fluctuations

29 MR-FC-91-0"5
Pressurizer Flange Leak

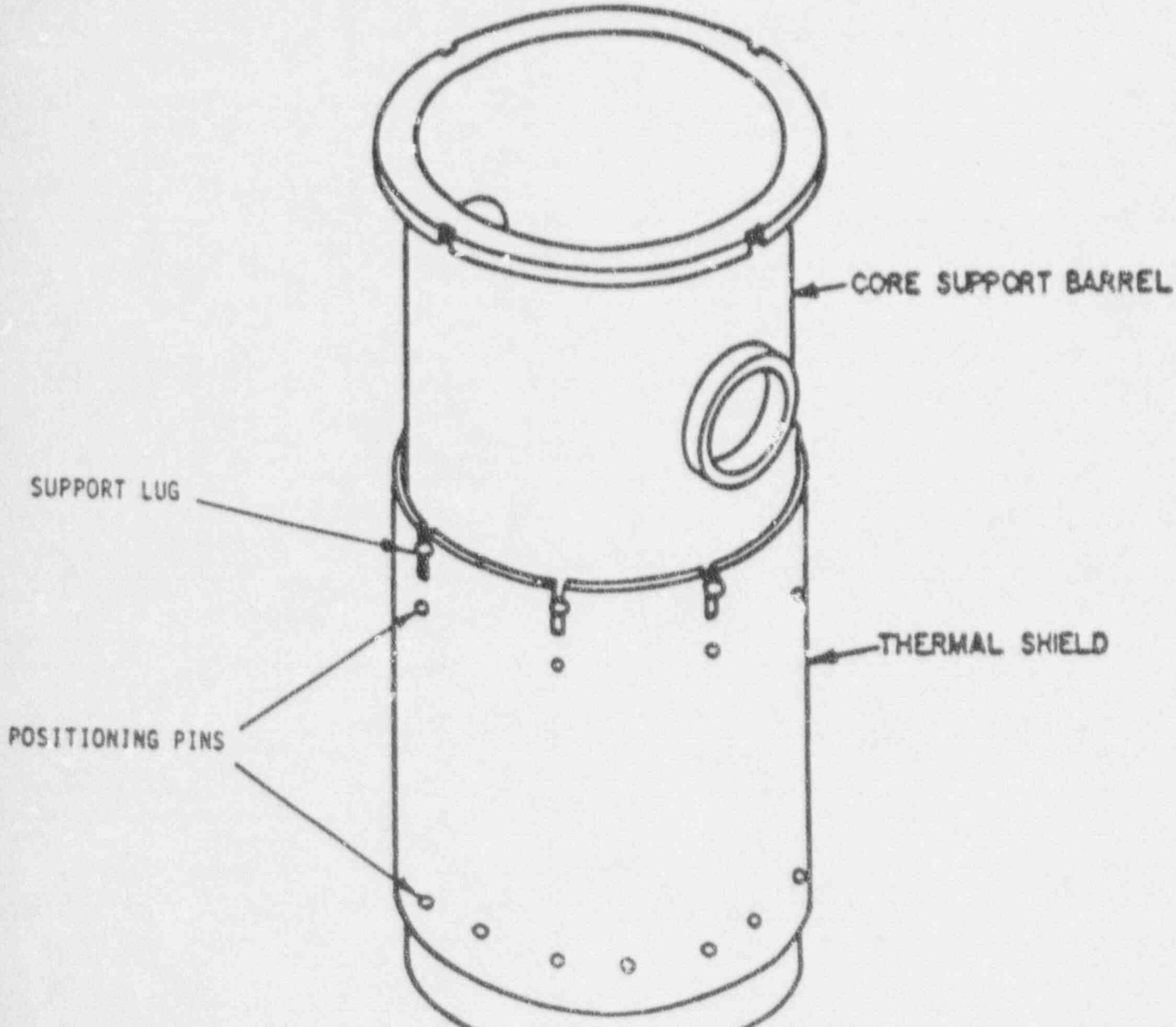
30 MR-FC-91-025
Replacement of LCV-383-1 and LCV-383-2

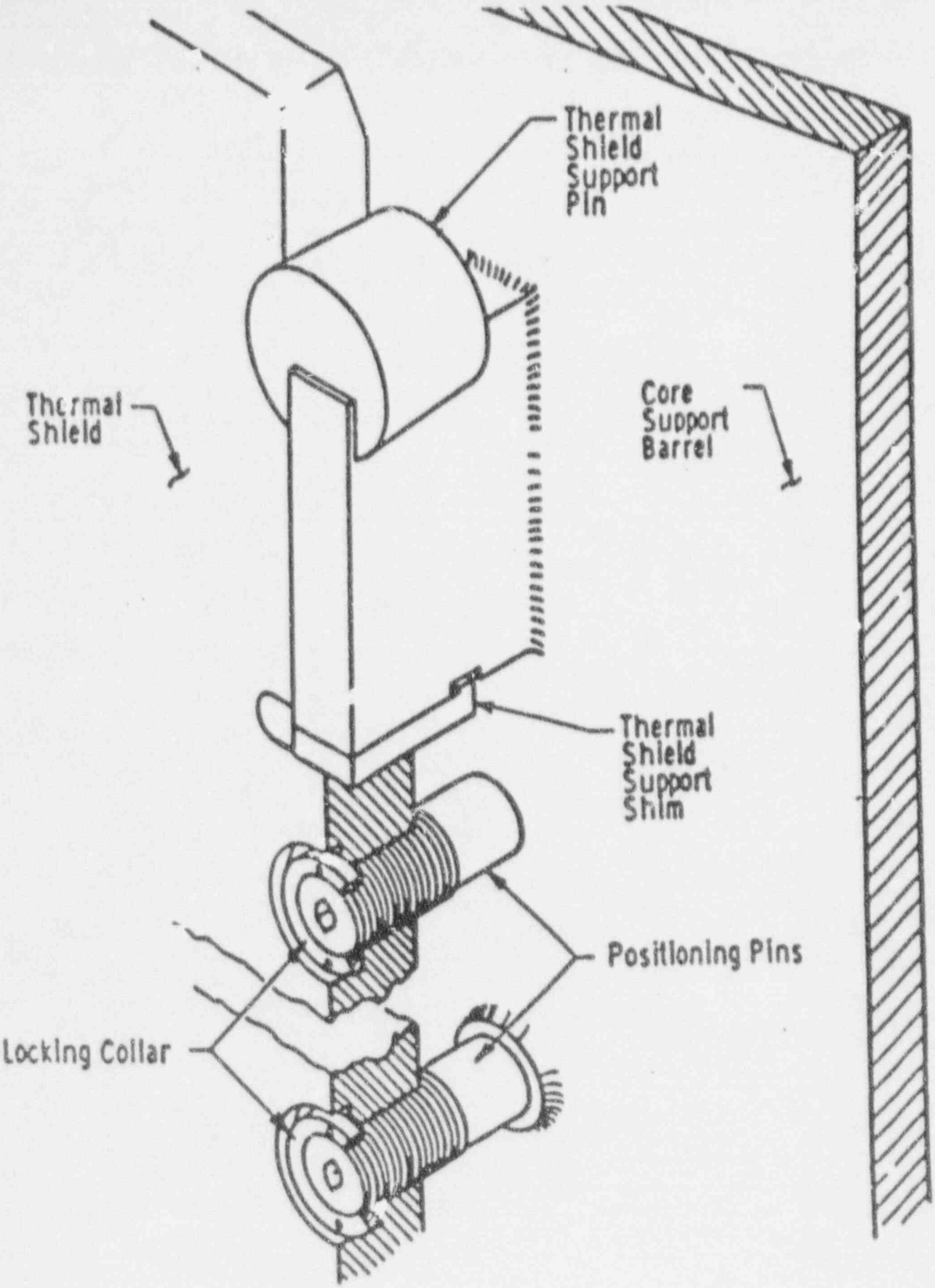
31 MR-FC-91-026
Station Battery Replacement

32 MR-FC-91-028
Steam Generator Insulation Support Ring

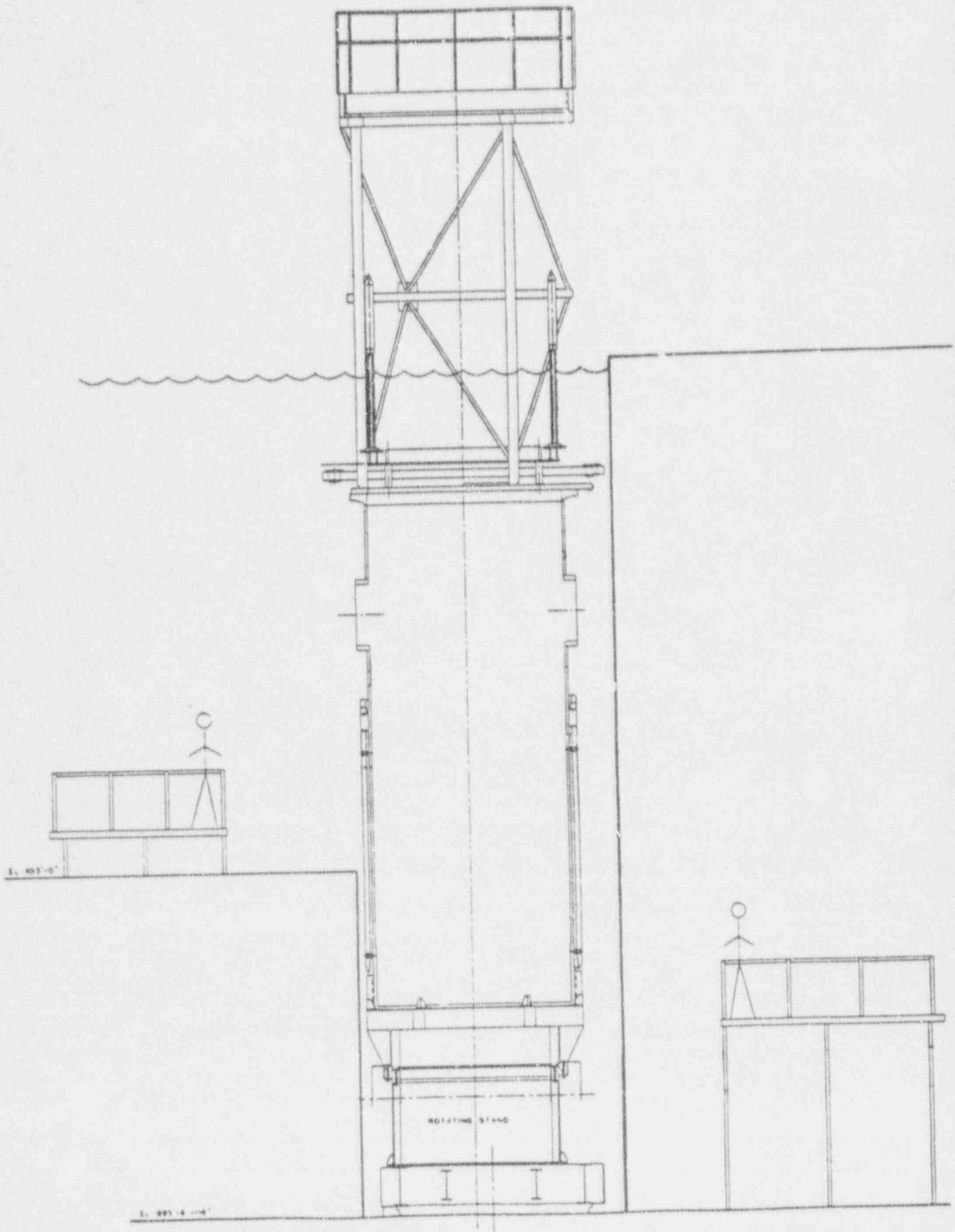
THERMAL SHIELD INSPECTION/REPAIR

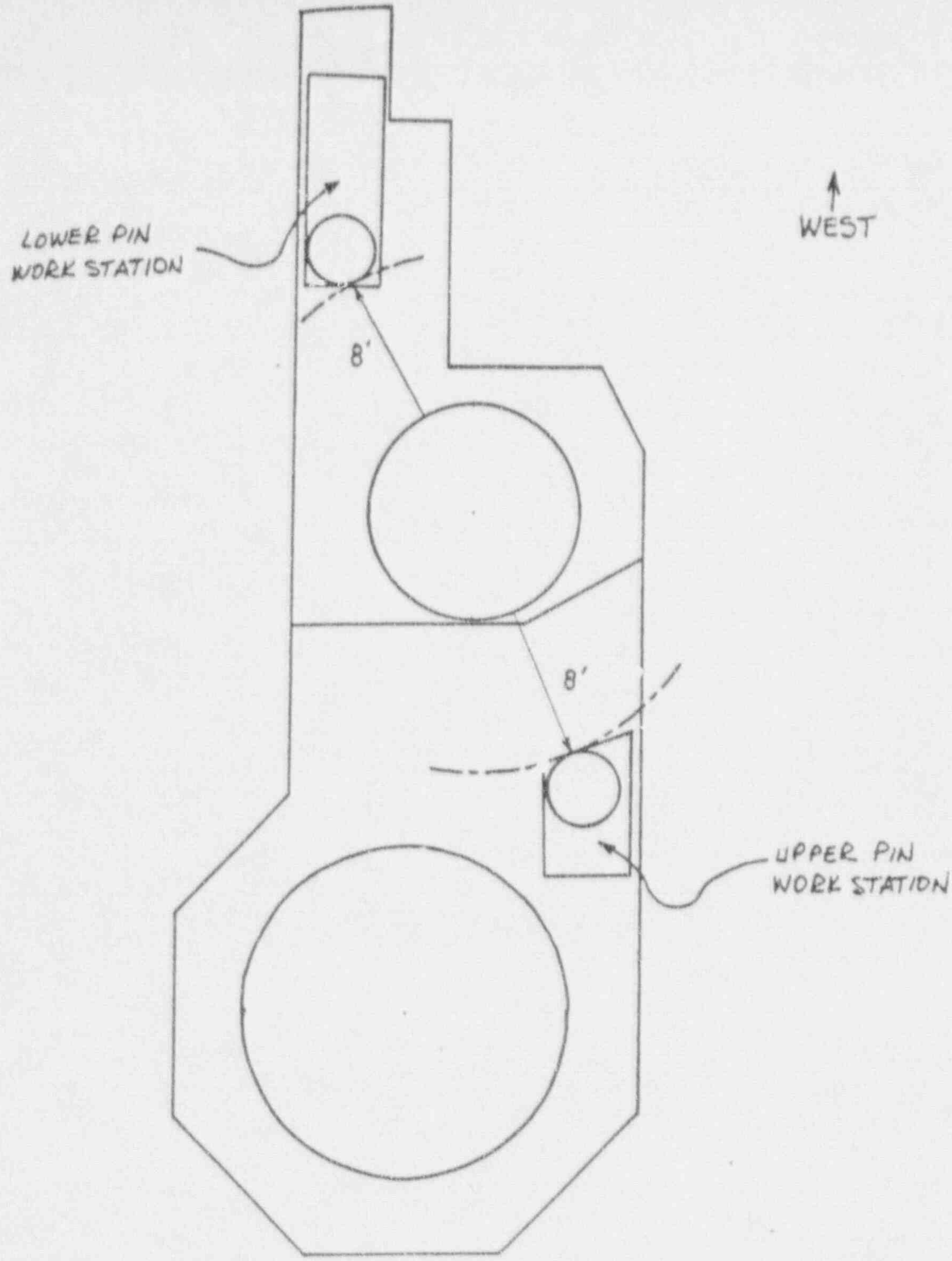
CORE SUPPORT BARREL
WITH
THERMAL SHIELD



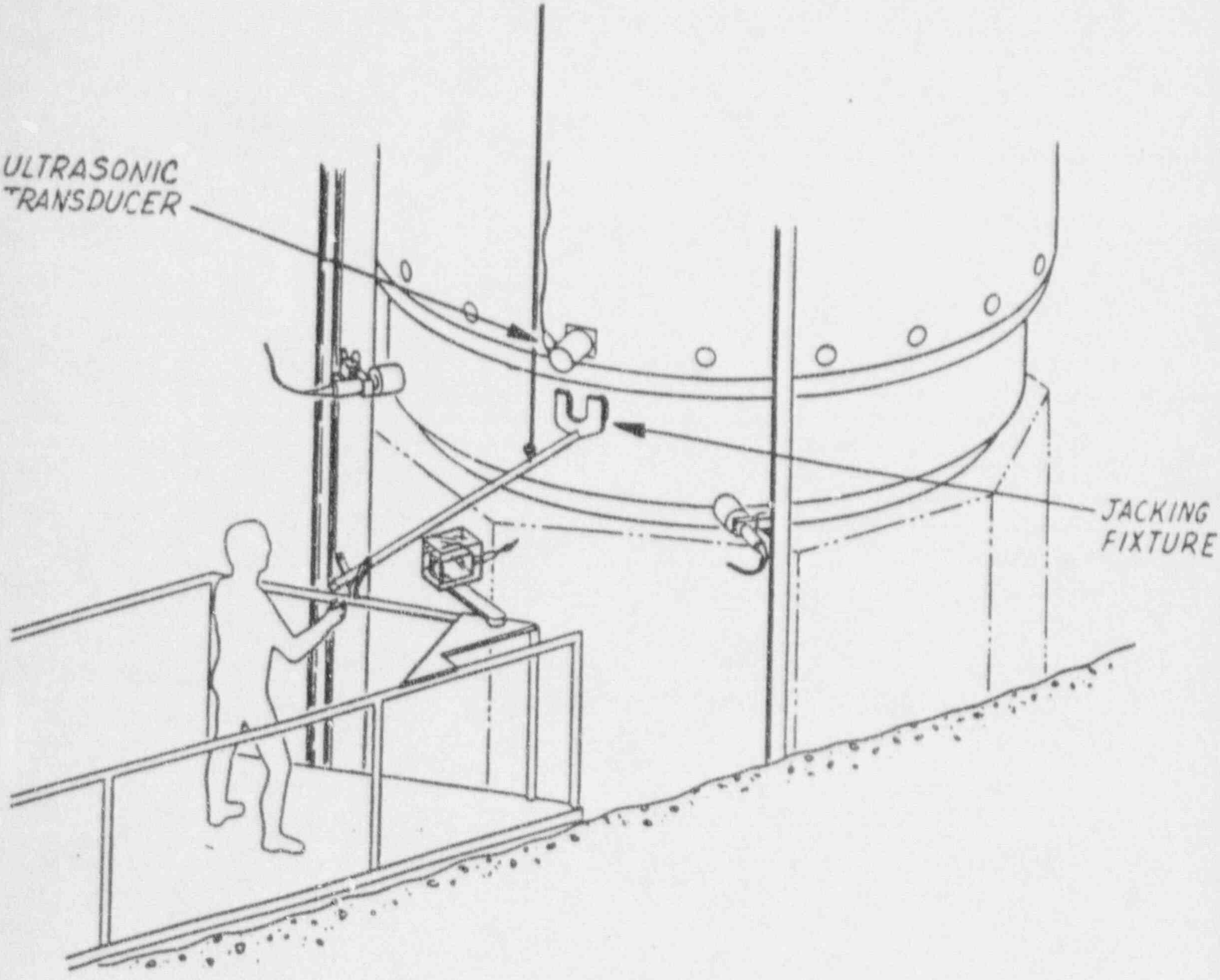


Thermal Shield Connection

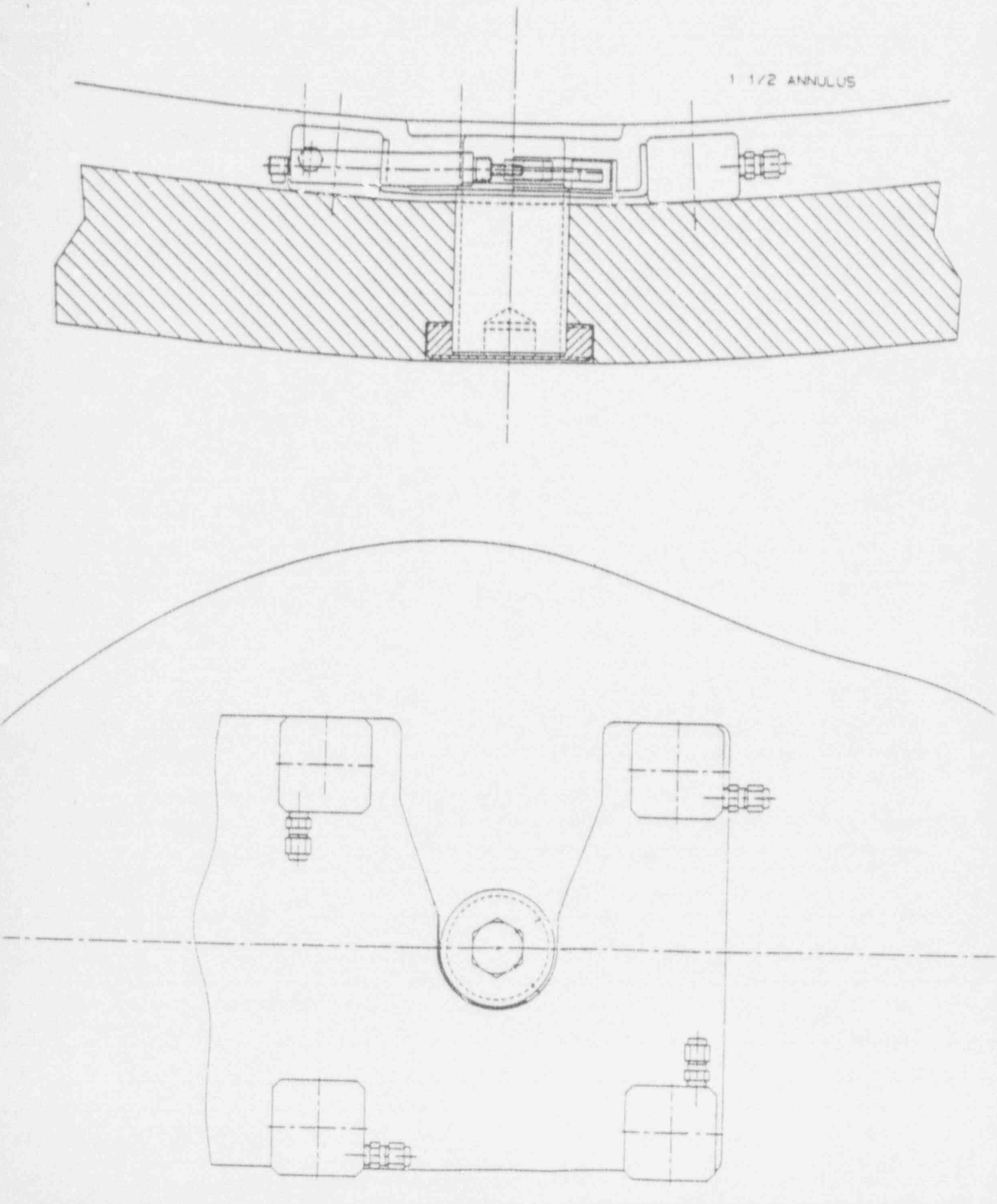

ROTATING STAND

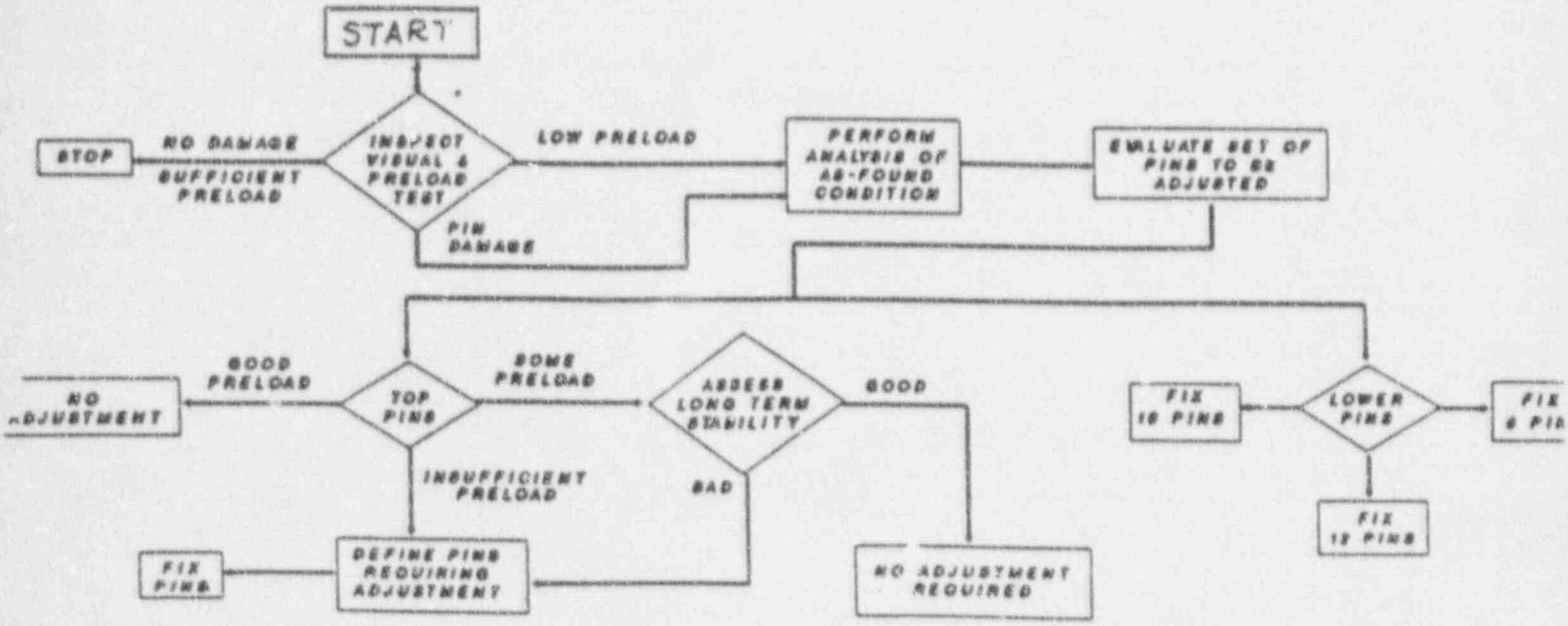


DIVER WORK STATIONS

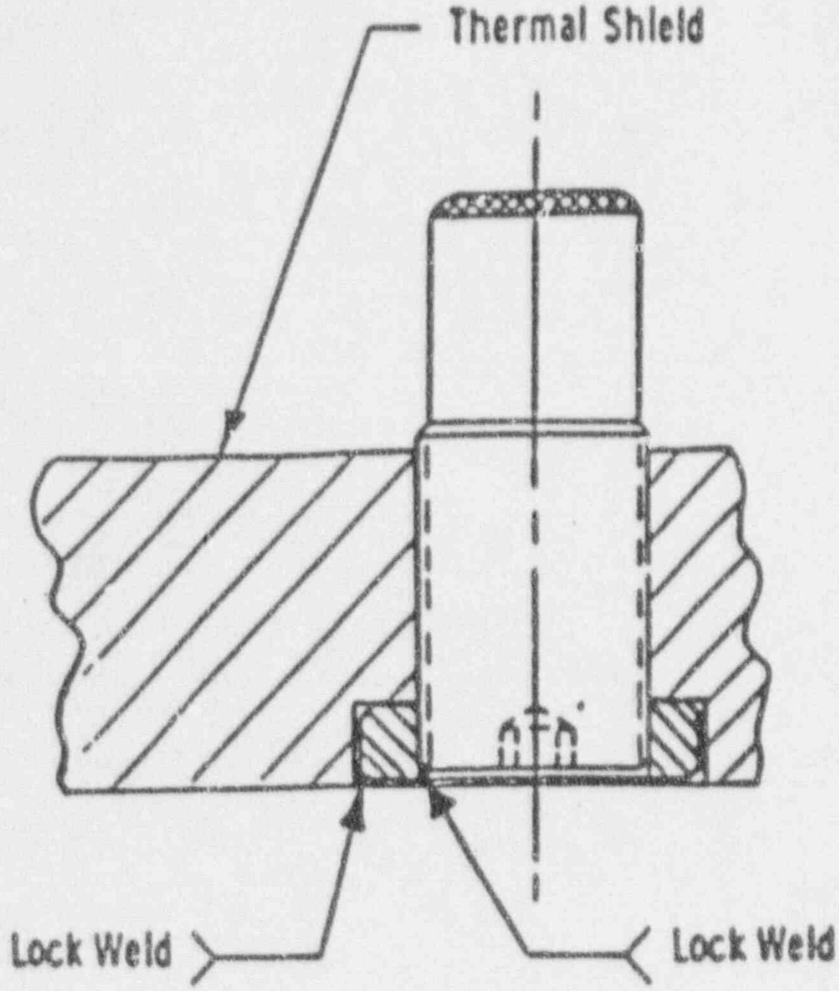


POSITION PIN PREOAD TESTING

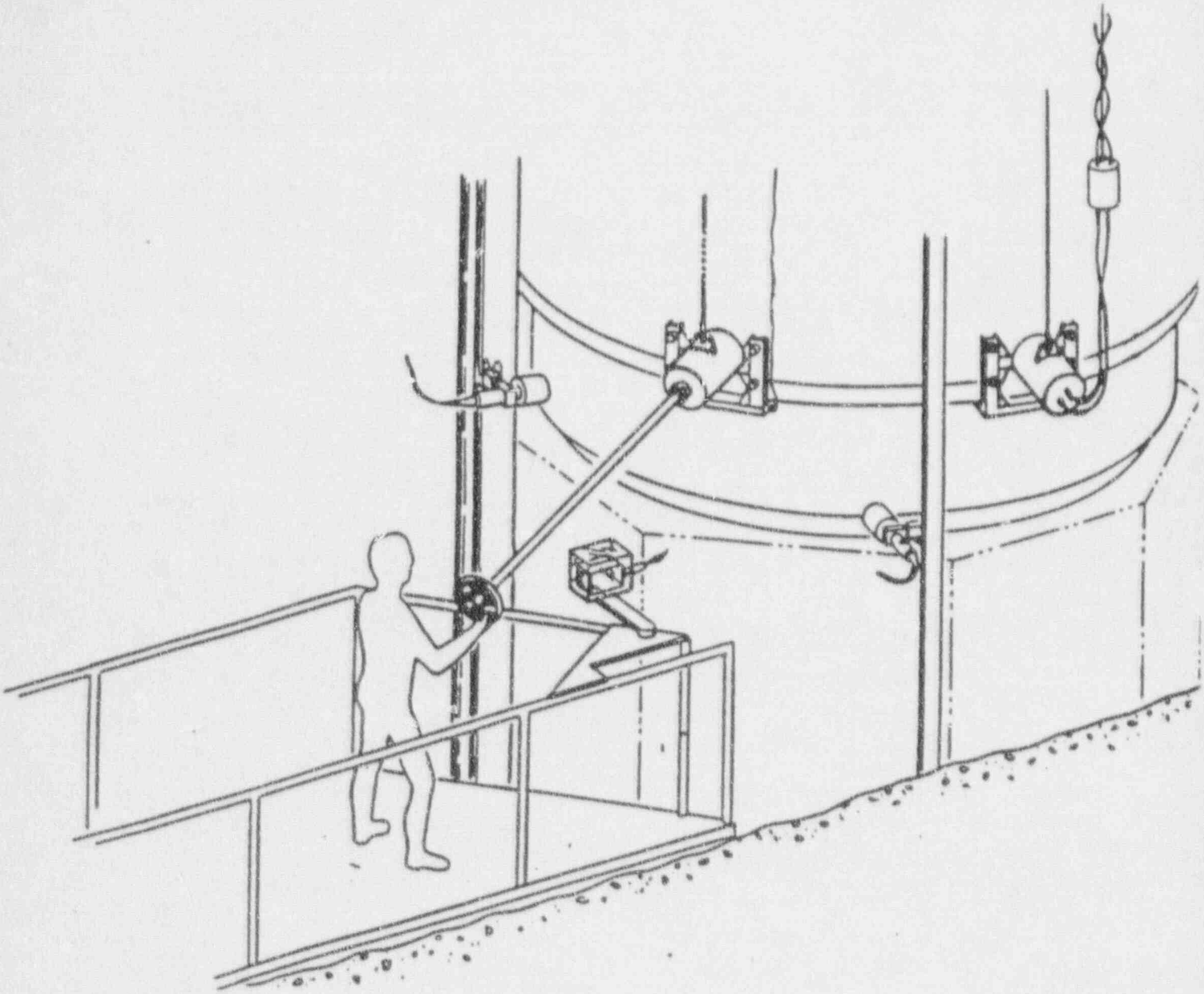

1 1/2 ANNULUS



LOGIC DIAGRAM FOR
PARTIAL RESTORATION OF PIN RELOAD

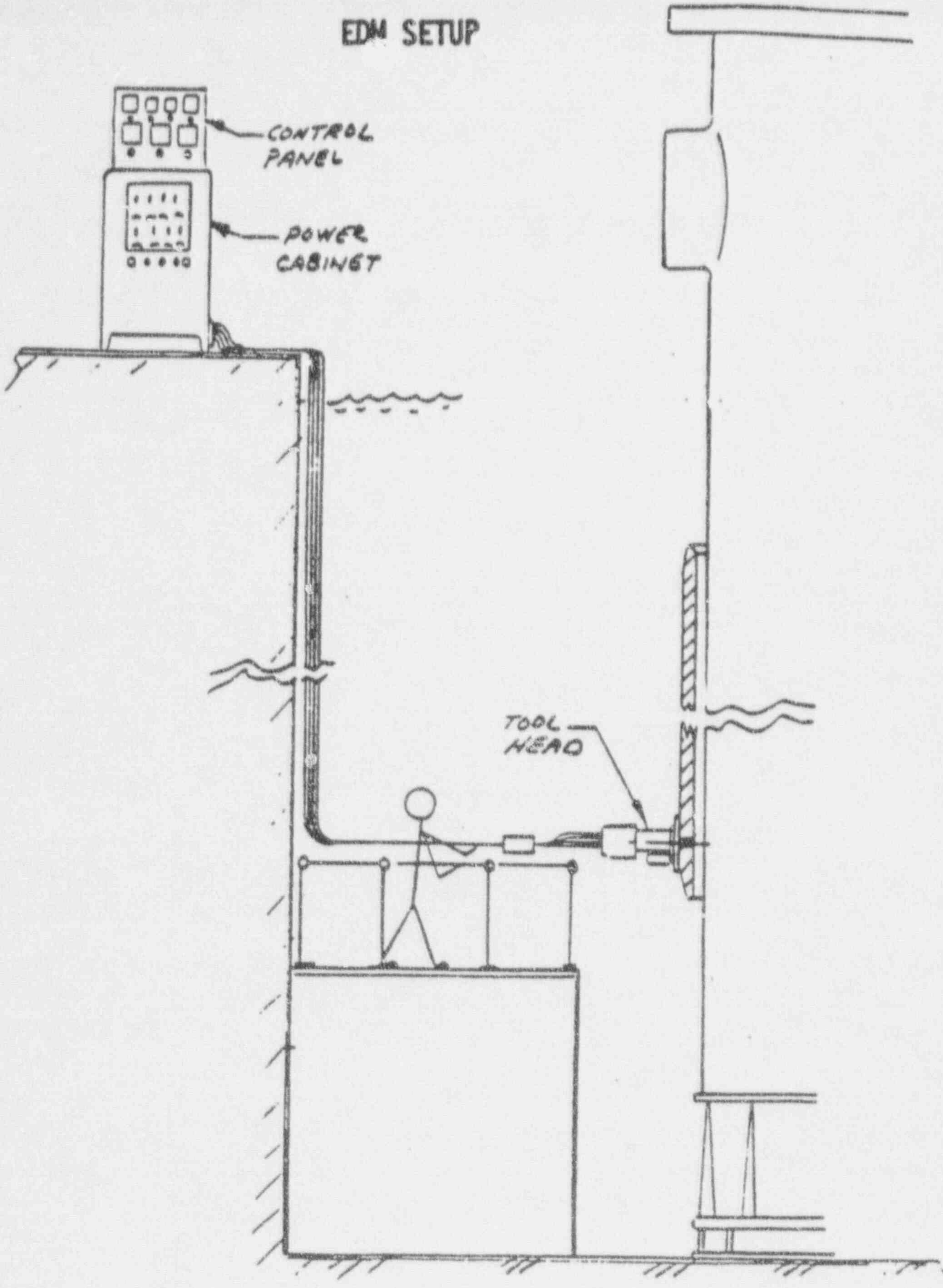


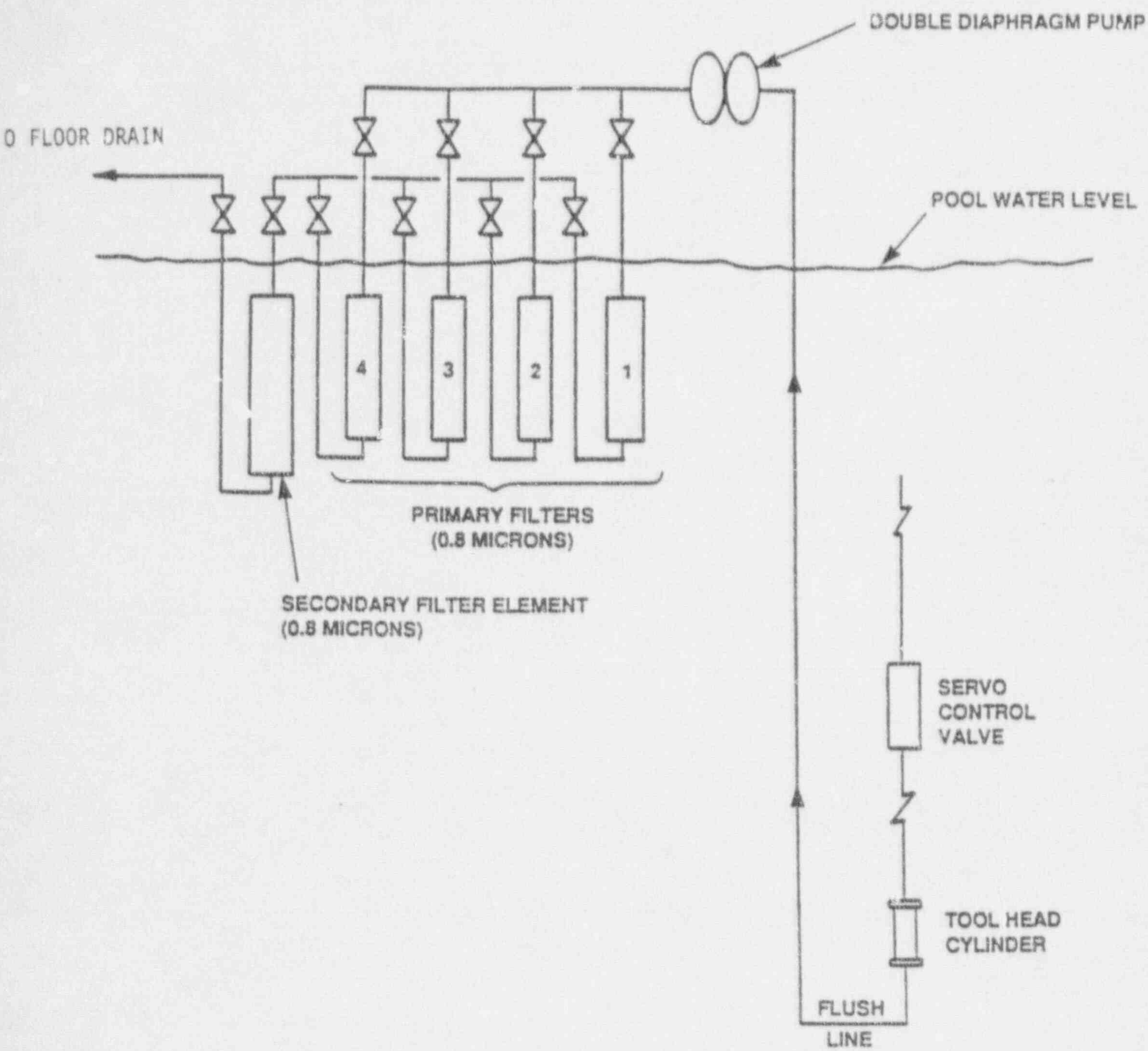
Thermal Shield Positioning Pin



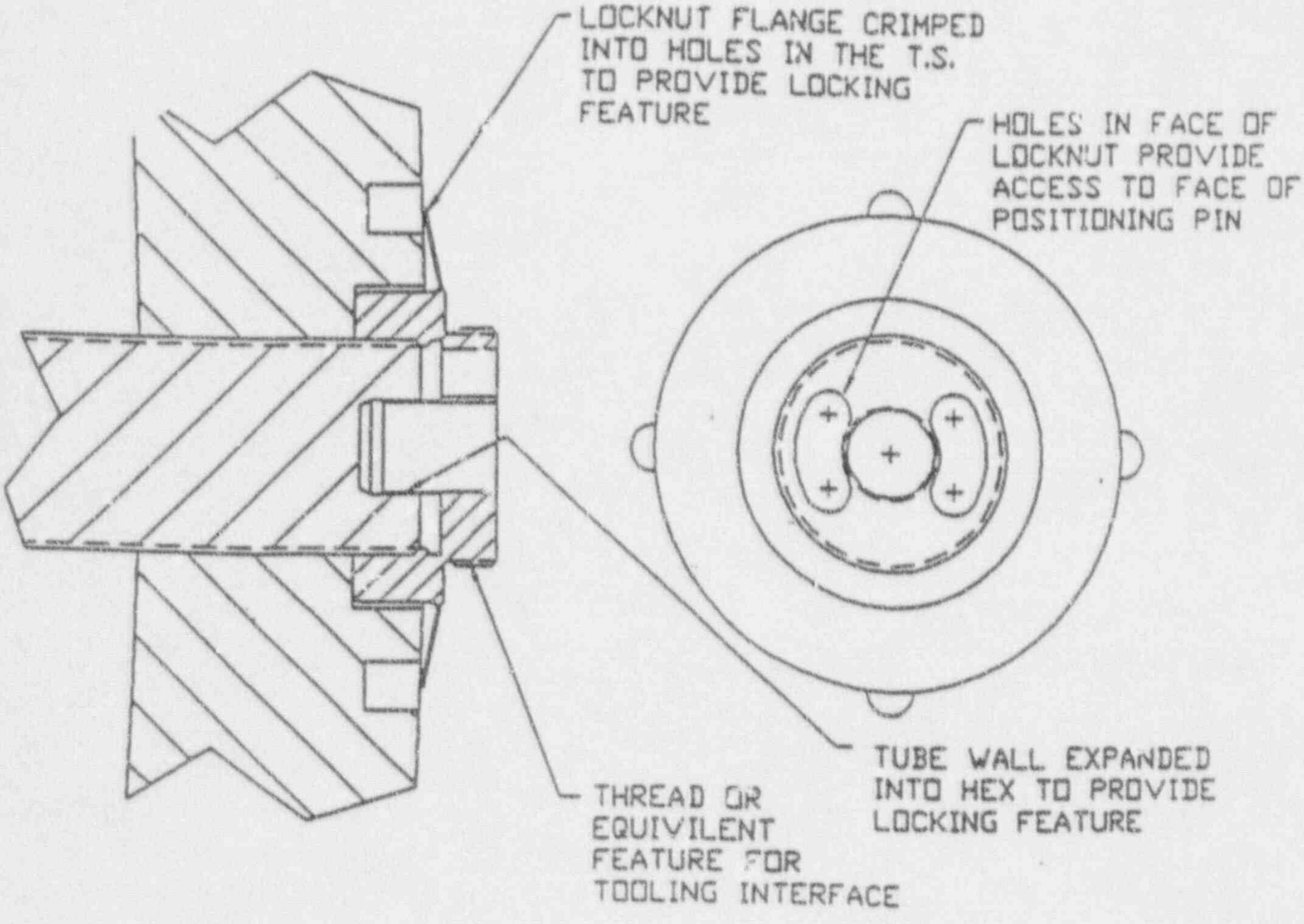
EDM TOOL DELIVERY AND OPERATION

EDM SETUP





EDM FILTRATION SCHEMATIC



ONE PIECE LOCKNUT DESIGN

STEAM GENERATORS

FORT CALHOUN STATION
STEAM GENERATOR EDDY CURRENT INSPECTION PROGRAM
1992 REFUELING OUTAGE

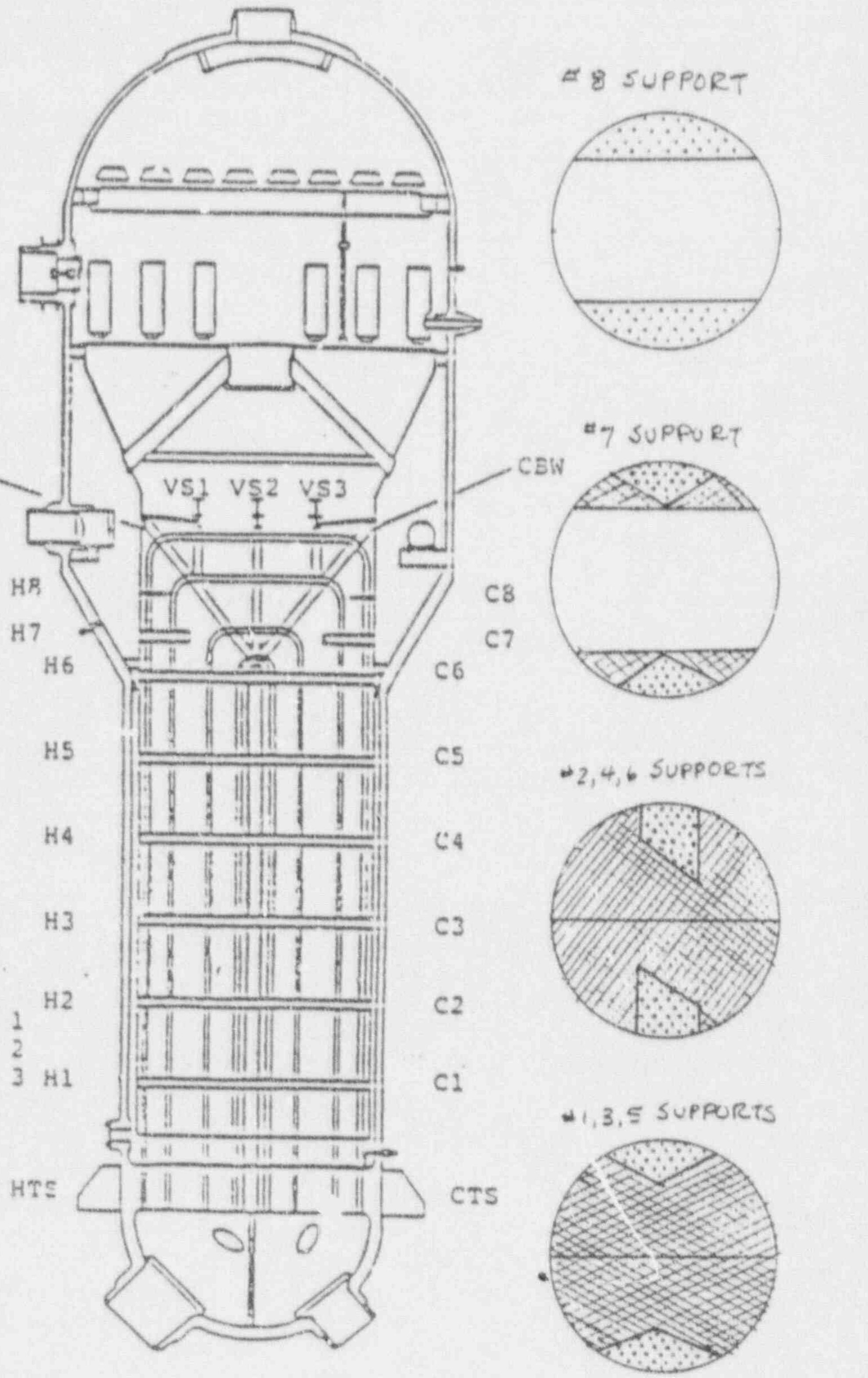
BACKGROUND

- Fort Calhoun Station has two Combustion Engineering Steam Generators with 5005 Inconel-600 tubes each

- Steam Generators have been in service for 18 years

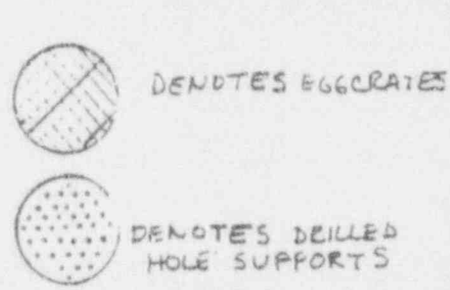
- Steam Generator-A has 55 tubes plugged (1.10%), 48 of which were plugged pre-operational

- Steam Generator-B has 54 tubes plugged (1.08%), 49 of which were plugged pre-operational



HTS - Hot Tube Sheet
HBW - Hot Batwing
VS1 - Vertical Support 1
VS2 - Vertical Support 2
VS3 - Vertical Support 3
CBW - Cold Batwing
CTS - Cold Tube Sheet

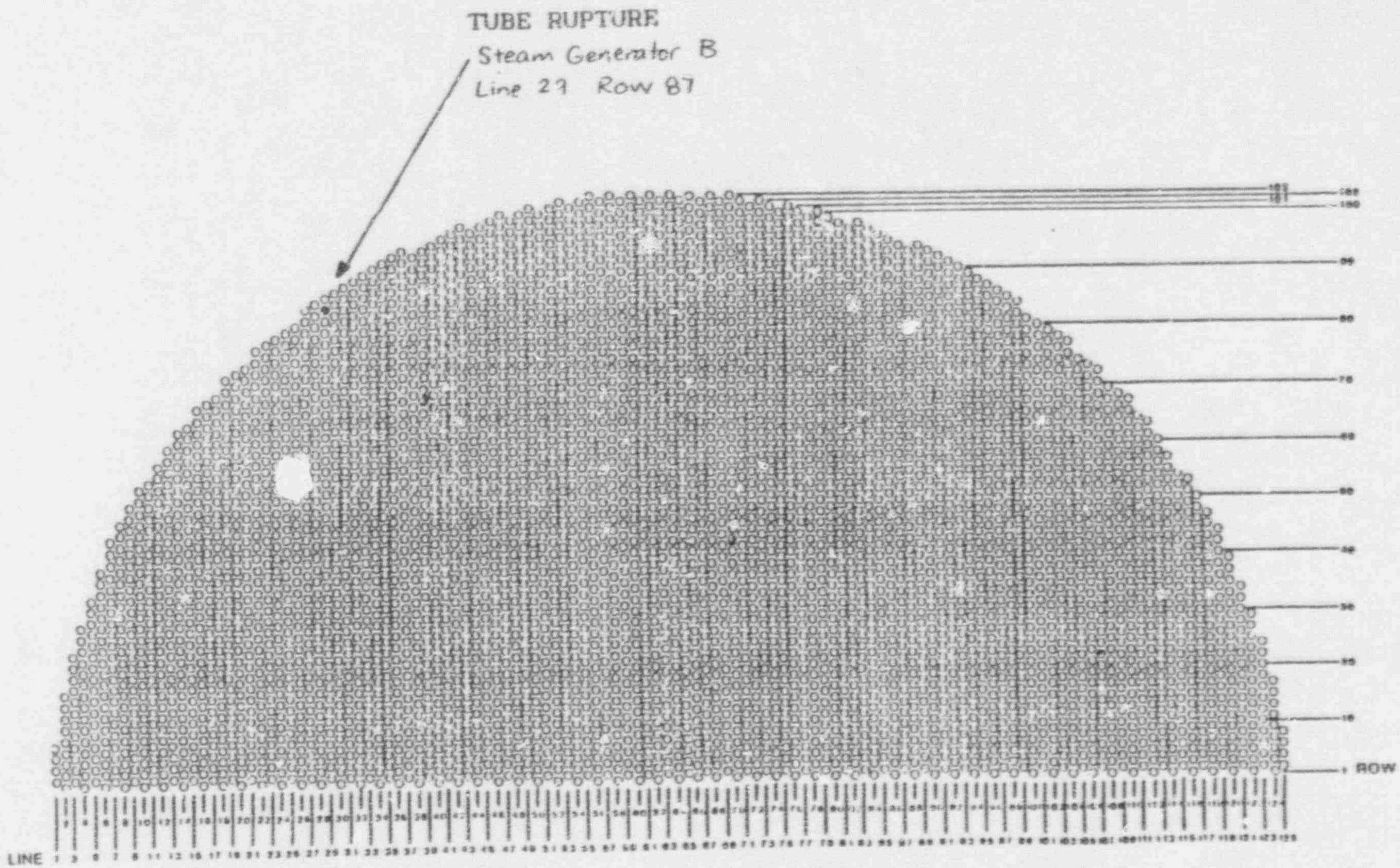
OPPD STEAM GENERATOR
SUPPORT NOTATION

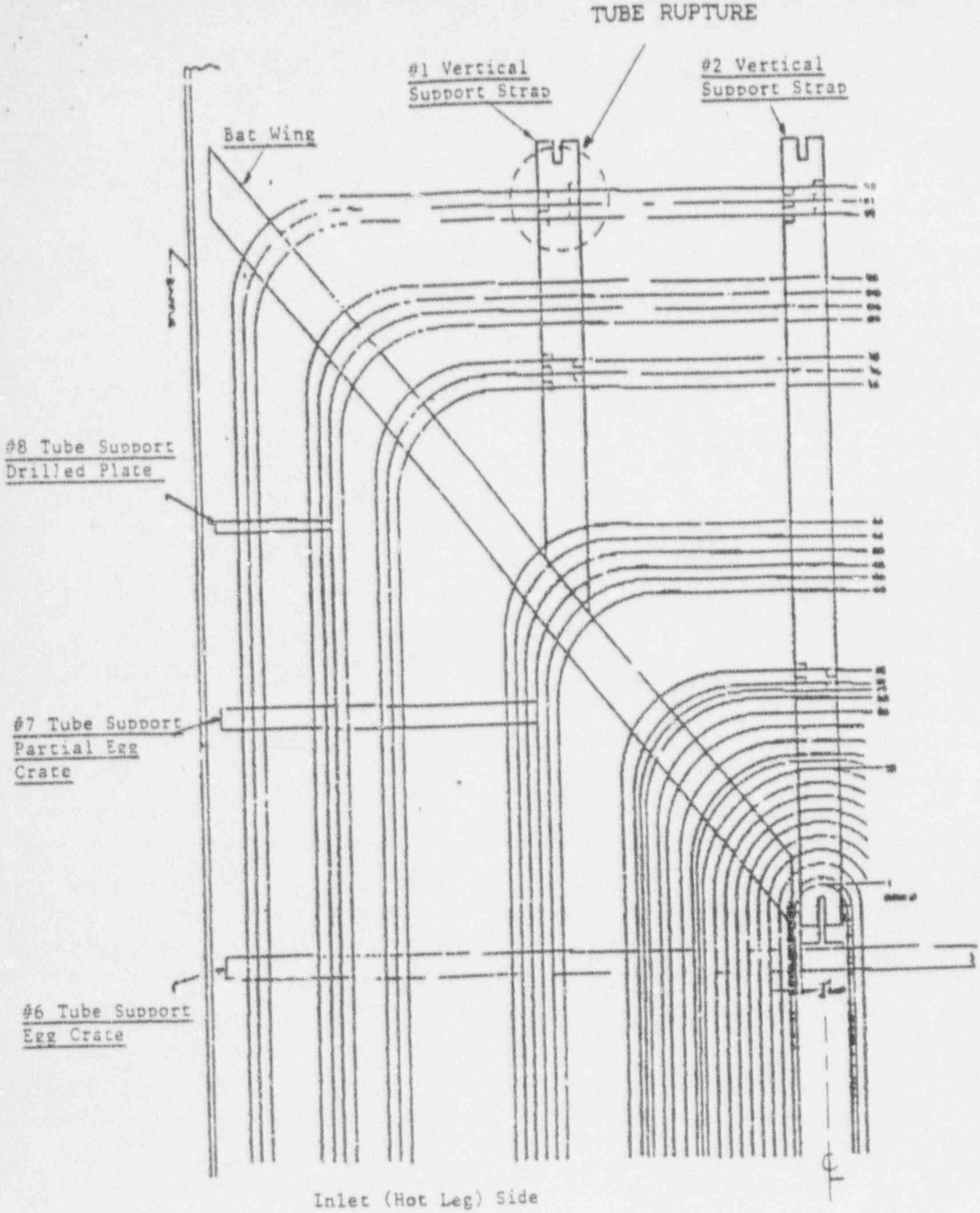


HISTORICAL INFORMATION

- Prior to 1984, scope of Eddy Current Testing met minimum requirements of Technical Specifications
- Results of each inspection prior to 1984 - Category C-1
- 1978 - Two tubes preventively plugged with < 40% indications
- 1981 - First indications of magnetite denting reported for Steam Generator-B. No tubes required plugging
- 1982 - Moderate dent like indications seen in both Steam Generators. No tubes required plugging
- 1984 - Tube rupture occurred in Steam Generator-B during "hot hydro" leak test of the Reactor Coolant System
  - Cause - Caustic induced IGSCC in a highly stressed portion of the tube - associated with a large dent

May 16, 1984





Steam Generator Support
'U' Bend & Vertical Support

HISTORICAL INFORMATION

- 1984 - As a result of the tube rupture, Eddy Current Testing exams were performed on nearly 100% of the tubes in each Steam Generator

  - Baseline profilometry exams were performed on approximately 350 tubes to better quantify denting

  - Total of 25 tubes plugged

    - 9 due to denting
    - 8 due to indications in hot leg VS region
    - 8 other

- In an effort to promote Steam Generator integrity, the following actions were taken in 1984:

  - Steam Generator Integrity Committee formed.

    - Committee comprised of Fort Calhoun Station staff and support personnel

    - Initiated and implemented suggestions concerning programmatic, operational and mechanical improvements to promote Steam Generator integrity

    - Committee continues to meet periodically to discuss Steam Generator integrity issues and potential improvements

  - Sludge lancing first performed. Performed each outage since 1984

  - Chemistry Limits and Guidelines revised to EPRI Standards

  - Condenser inleakage more actively monitored and aggressively combatted. This practice has continued to present

HISTORICAL INFORMATION

- 1985 - Eddy Current Testing of approximately 20% of tubes in both Steam Generators

  - Inspection included all tubes in Row 74 and above - most severely dented region

  - Profilometry sample reinspected

  - Total of 33 tubes plugged

    - 28 due to denting
    - 2 due to indications in hot leg VS region
    - 3 other

  - Corrosion mechanism has obviously not been arrested

- In an effort to promote Steam Generator integrity, the following actions were taken:

  - In 1985, copper feedwater heaters replaced with stainless steel to reduce copper ingress into Steam Generators

  - In 1986, boric acid injection was begun in an attempt to arrest denting process. Has continued to present

  - In 1987, condensate demineralization was performed during the Refueling Outage. Has continued each refueling outage to present

HISTORICAL INFORMATION

- 1987 and 1988

  - Eddy Current Testing sample - approximately 20%

  - Included inspection of all tubes in Row 74 and above

  - Profilometry sample reinspected

  - Results

    - No significant dent growth
    - No tubes required plugging

- 1990 - Began following EPRI guidelines for Eddy Current Testing Inspection Plan

  - Approximately 34% of total tubes inspected including:

    - All tubes in Row 74 and above
    - 20% of remaining tubes

  - Profilometry sample reinspected

  - Results

    - No significant dent growth
    - No tubes required plugging

- Steam Generator Program Basis Document completed in 19[illegible].

  - Defines departmental responsib[illegible]ies

  - Outlines reg[illegible]latory requirements

  - Provides historical information

STEAM GENERATOR PROGRAM

PROGRAM BASIS DOCUMENT

TABLE OF CONTENTS

- OPPD is a member of EPRI Steam Generator Reliability Project

  - Have members on the Senior Representativies Steering Committee and SGRP Technical Advisory Group

  - Use EPRI information to help keep abreast of Steam Generator issues in the industry.

1992 FORT CALHOUN STATION
BOBBIN COIL INSPECTION PLAN

EACH STEAM GENERATOR

- Restricted Sample: All tubes previously restricted to a .560" diameter probe above Row 74

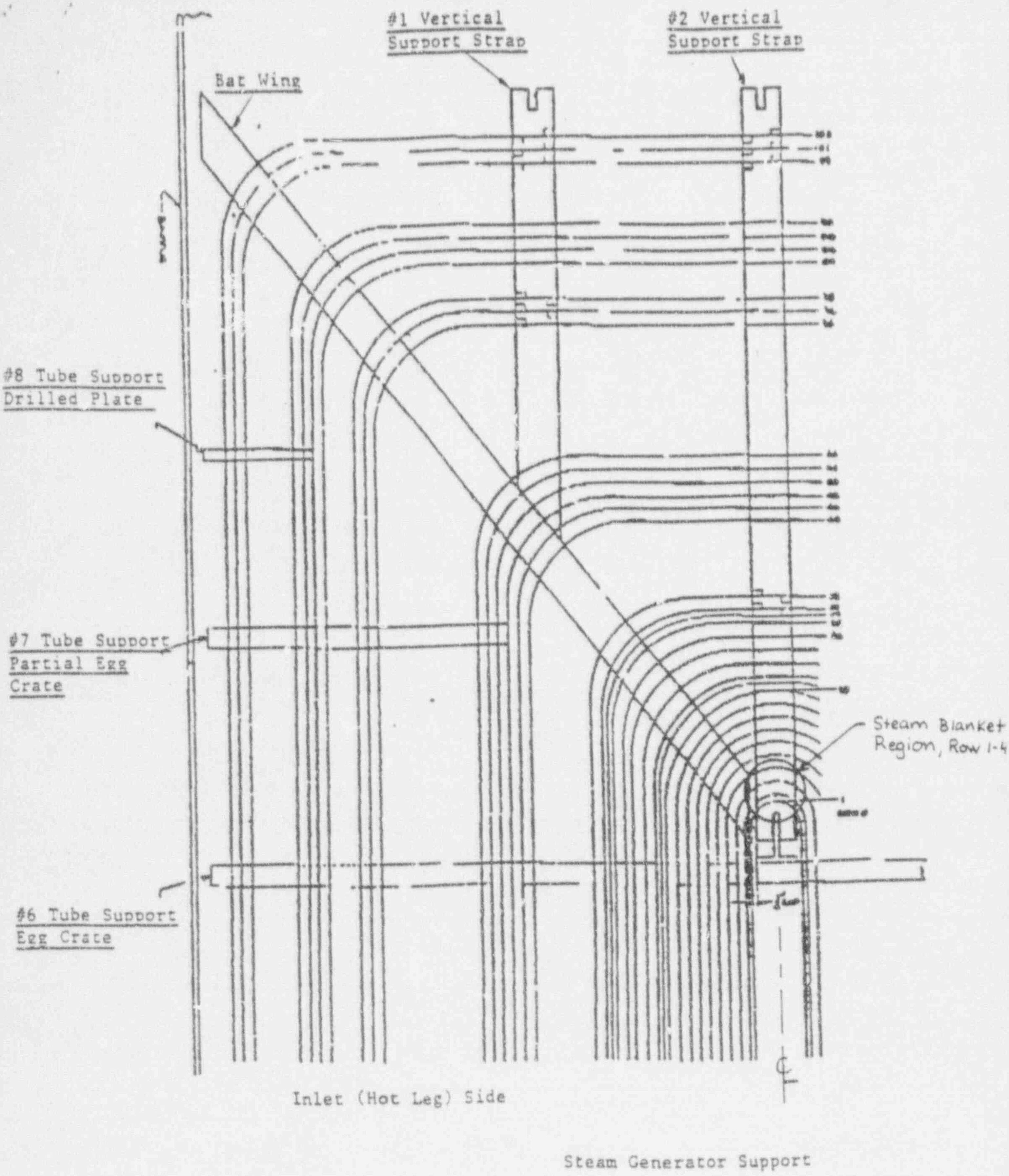
- Evaluate for dent progression and flaw initiation

- If no growth detected in restricted sample, test:
  - All tubes adjacent to stay rod locations
  - All previously degraded tubes and one adjacent
  - All tubes in steam blanket region
  - 20% of remaining tubes

- No growth sample results in approximately 26% sample

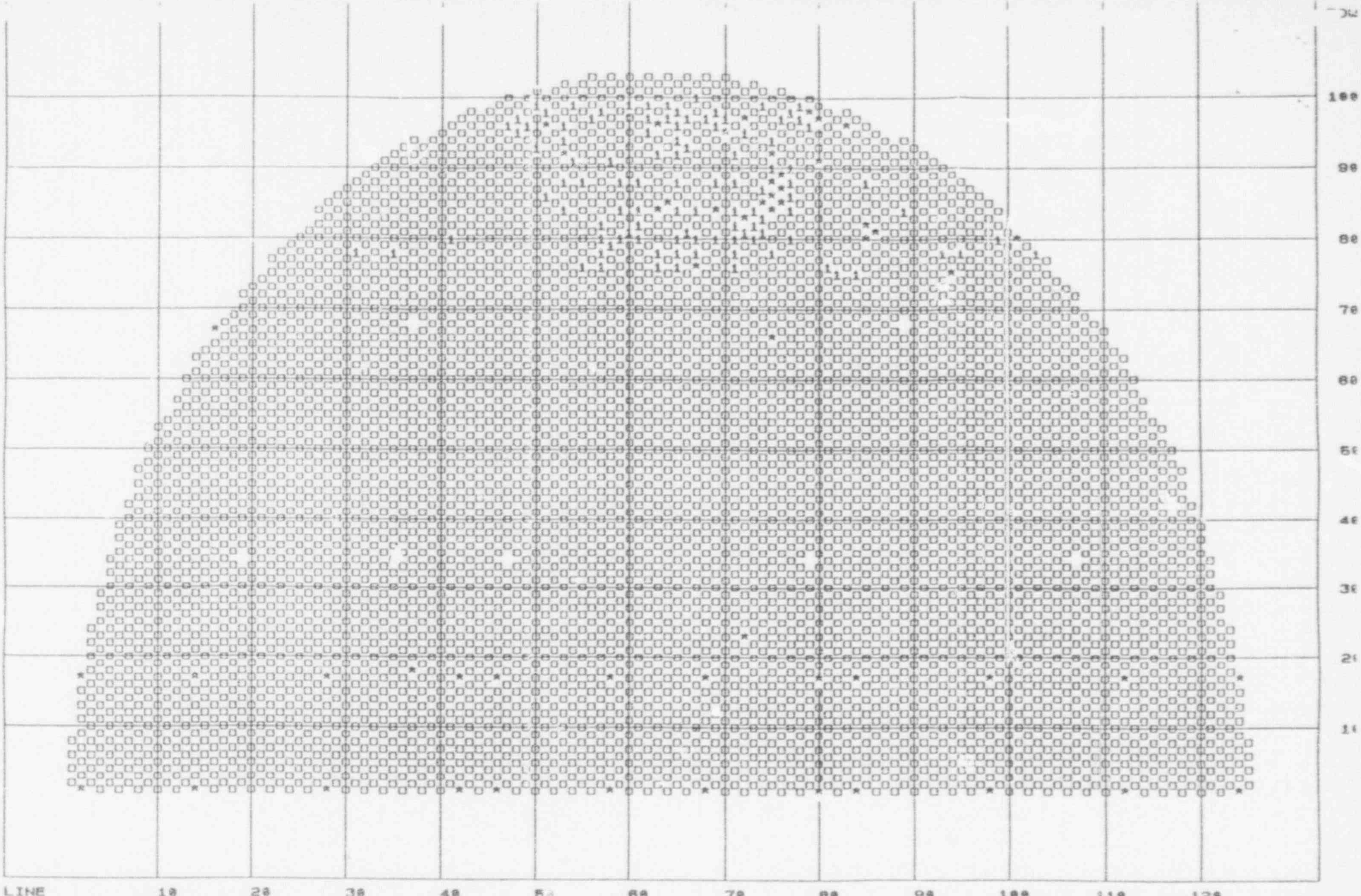
- If growth detected in restricted sample, test:
  - All tubes specified above
  - All tubes in Row 74 and above
  - 20% of remaining tubes

- Growth sample results in approximately 39% sample



Steam Generator Support

'U' Bend & Vertical Support

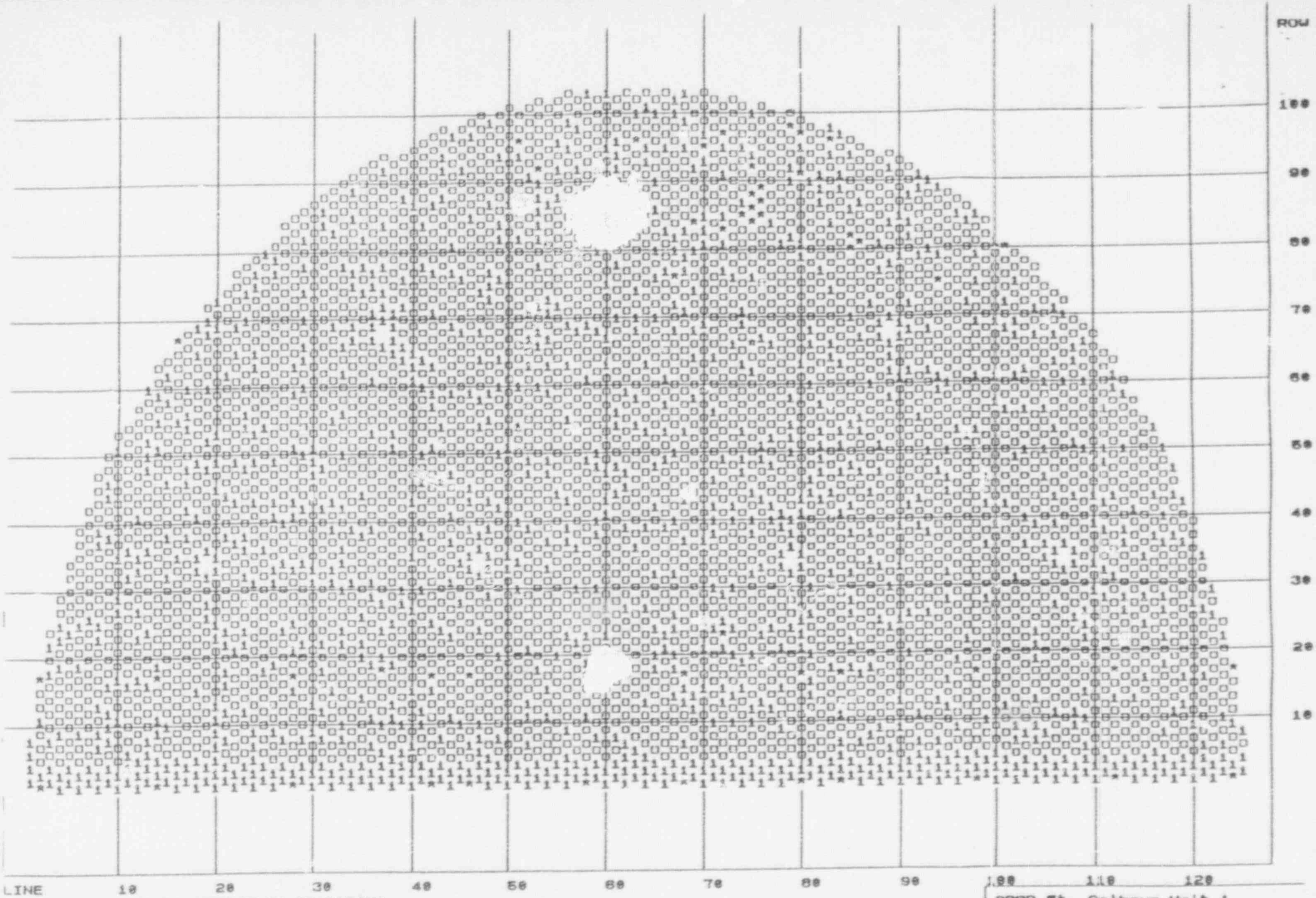


1: 104: INITIAL SAMPLE AC-RES92
*: 66: PLUGGED TUBES

OPPD Ft. Calhoun Unit 1
Steam Generator A
CE Data Mgmt 04/29/91

INFORMATION ONLY

ACRI 1515 Tubes

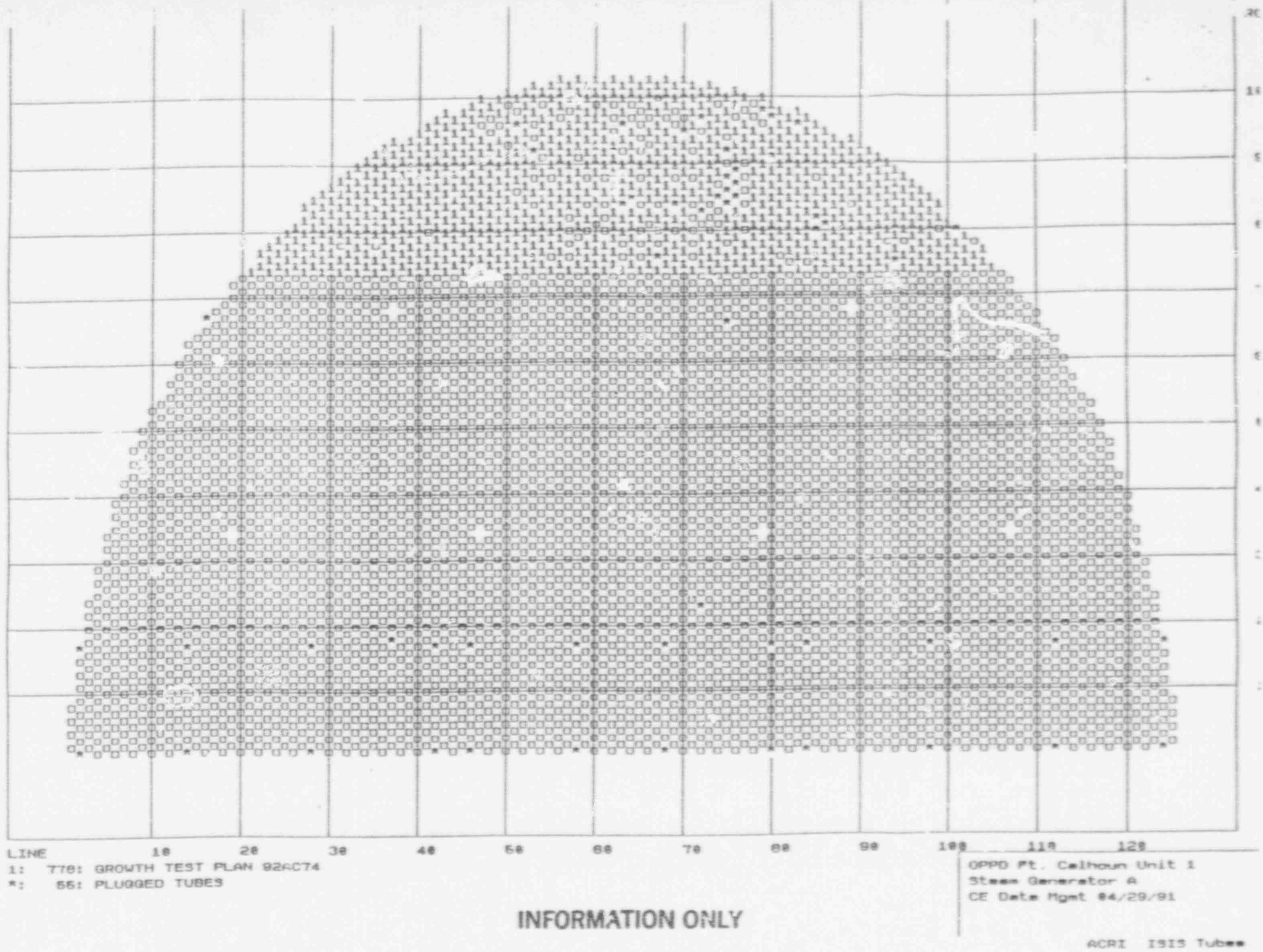


1: 1212: NO GROWTH TEST PLAN AC-20RAN1
*: 66: PLUGGED TUBES

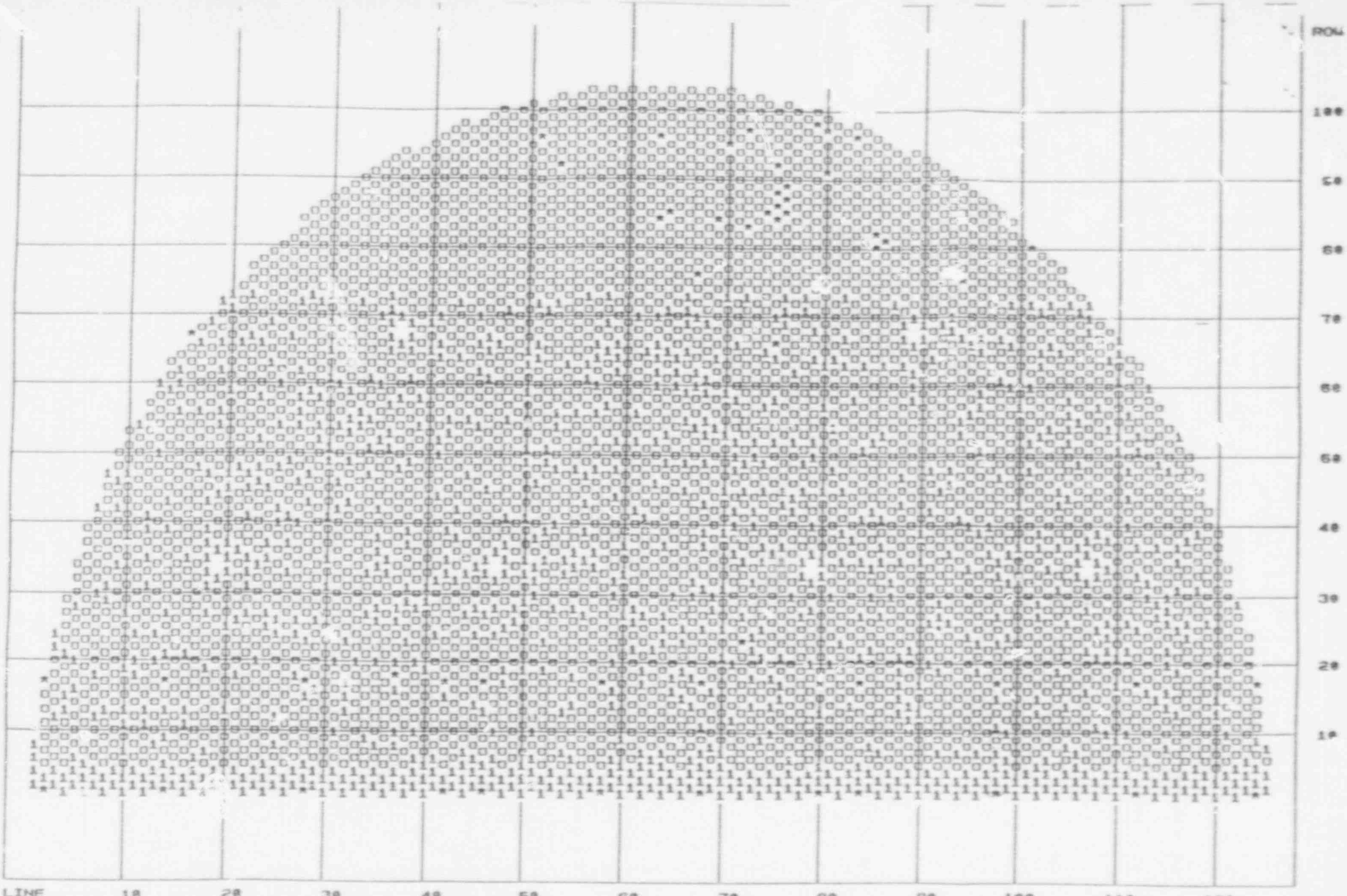
OPPD Ft. Calhoun Unit 1
Steam Generator A
CE Data Mgmt 08/12/91

INFORMATION ONLY

ACRI ( ) Tubes



INFORMATION ONLY



1: 1057: GROWTH TEST PLAN AC-20RAN2
*: 55: PLUGGED TUBES

OPPD Ft. Calhoun Unit 1
Steam Generator A
CE Data Mgmt 09/11/91

INFORMATION ONLY

ACRI .3 Tubes

1992 FORT CALHOUN STATION
MOTORIZED ROTATING PANCAKE COIL EXAMINATIONS

EACH STEAM GENERATOR

- 20% sample of hot leg expansion transitions

- Due to circumferential cracking concern as seen at other Combustion Engineering units

PLUGGING CRITERIA

- Plug all defects equal to or greater than 40% through wall
- Plug all circumferential crack-like indications
- Plug all tubes that will not pass a .540 inch diameter probe

NOTE: Standard Combustion Engineering mechanical expanded plugs of Inconel 690 material will be used for tube plugging.

IMPLICATIONS OF TROJAN EDDY CURRENT EXPERIENCE

- Fort Calhoun Station is not planning additional MRPC inspections in 1992 as a result of recent problems encountered at Trojan Nuclear Plant.

- Bases for this decision:

  - Trojan has had active, widespread IGA/IGSCC for several cycles

  - Trojan has had many undefined signals for several cycles, many of which have been confirmed to be IGA/IGSCC

  - All data at Fort Calhoun Station since 1985 indicate corrosion has been arrested in the Fort Calhoun Station Steam Generators

  - There have been no undefined signals identified at Fort Calhoun Station similar to those at Trojan

  - .f cracking were present which could not be detected with the bobbin coil, it is expected that there would also be cracking present which could. This was the case at Trojan. No such indications have been identified at Fort Calhoun Station

  - Based on these facts, there is no reason to expect that Fort Calhoun Station has active IGA/IGSCC similar to that found at Trojan

REACTOR VESSEL INSPECTIONS

REACTOR VESSEL INSERVICE INSPECTION

1. **Status of ISI Program**

   a. Applicable Code ASME XI 1980 Edition Winter 1980 Addendum

   b. Second Ten Year Interval September 1983 - September 1993

   c. Vessel exam originally scheduled for 1993 Refueling Outage

   d. Rescheduled to be performed concurrent with Thermal Shield work

2. **Description of Exam**

   a. Original Reactor Pressure Vessel exam scope required by Program

   b. Description of exam areas

   c. Scope added with decision to do 100% vessel shell welds

3. **Schedule for Test Performance**

   a. Seven to ten days exam time on vessel for original scope

   b. 10½ to 13½ days on vessel for 100% of shell welds

   c. Total setup, exam and removal - 19 days

4. **Description of PaR and FAST-PaR**

   a. Capabilities of PaR

   b. Concept of FAST-PaR

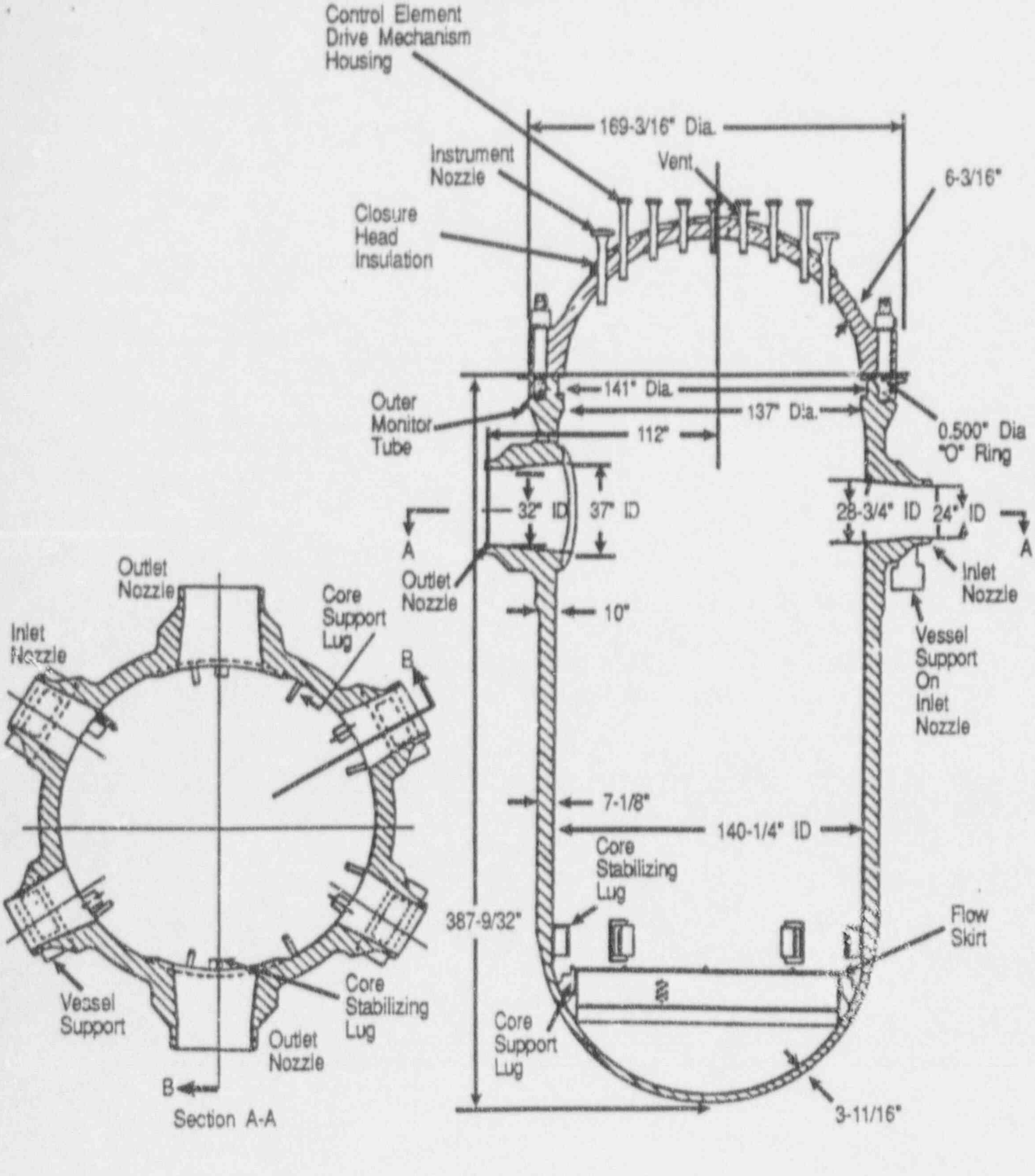
5. **Benefits of Additional Shell Weld Exams**

   a. Satisfies augmented examination of Reactor Vessel in January 31, 1991, Proposed Rule

   b. Data will be taken to satisfy Regulatory Guide 1.154 so that it may be used for a Pressurized Thermal Sh Analysis, if needed

   c. Help alleviate long standing discr between 1974 and 1980 ASME XI

REACTOR VESSEL



#1

RVDIAGRM

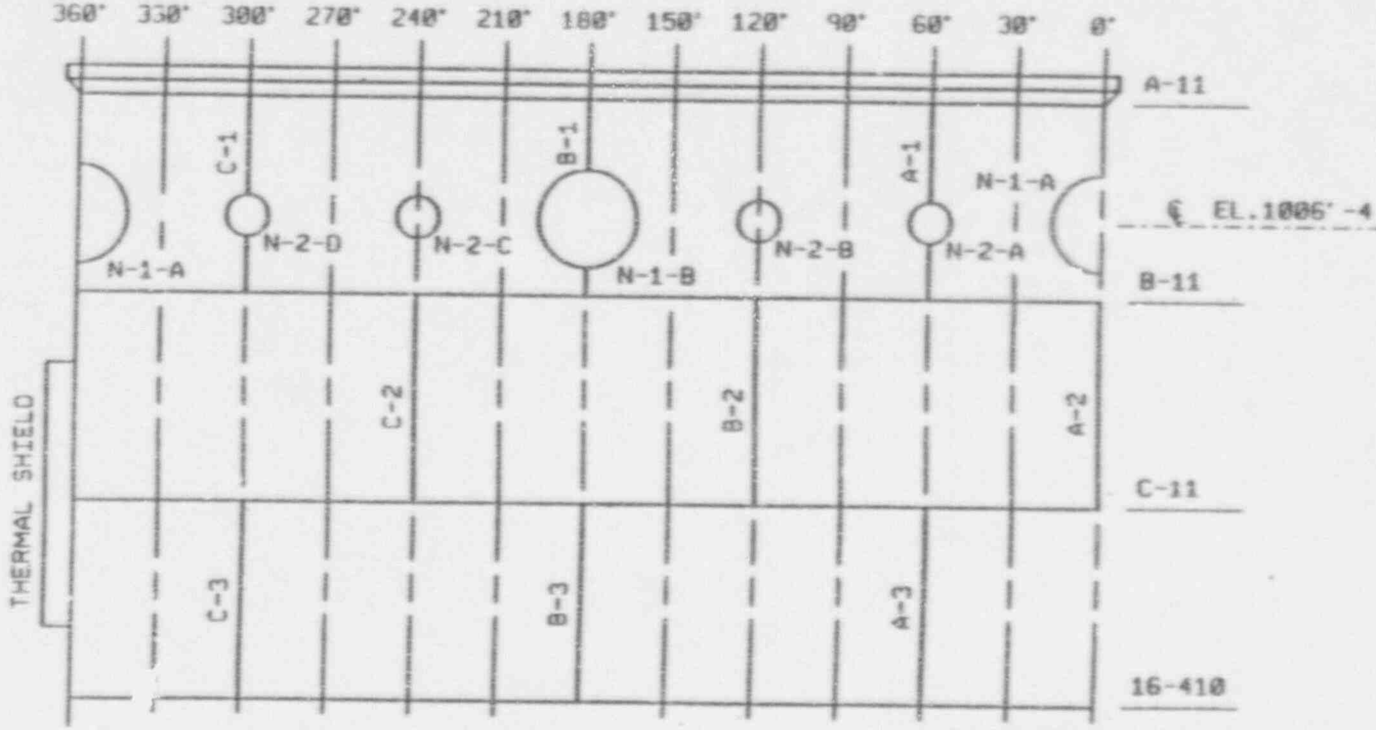
RPV and INTERNALS ISI TO BE PERFORMED WITH CORE BARREL REMOVED

MECHANIZED EXAMS

- RPV BELTLINE REGION CIRCUMFERENTIAL WELD
- RPV BELTLINE REGION LONGITUDINAL WELD
- BASE METAL WELD REPAIR AREAS IN THE RPV BELTLINE REGION WHICH EXCEED 10% OF NORMAL VESSEL WALL
- 4 RPV NOZZLES ON COLD LEGS
- 4 SAFE-ENDS ON THE COLD LEGS
- 4 ELBOW WELDS ON THE COLD LEGS
- 2 SAFE-ENDS ON THE HOT LEGS
- RPV LOWER HEAD CIRCUMFERENTIAL WELD
- RPV LOWER HEAD MERIDIONAL WELDS
- RPV SHELL TO FLANGE WELD

REMOTE VISUAL EXAMS

- VESSEL INTERNALS EXAM
- CORE BARREL EXAM
- CORE SUPPORT STRUCTURE EXAM



REACTOR PRESSURE VESSEL WELDS

CONTROLLED COPY

CALIBRATION BLOCKS:

2 - FCL
5 - FCL
7 - FCL
8 - FCL

RPV-N1-1 (VESSEL INTERIOR)
RPV-N3-CSS-1 (CORE SUPPORT STRUCTURE)

DATE	TASK/MR #

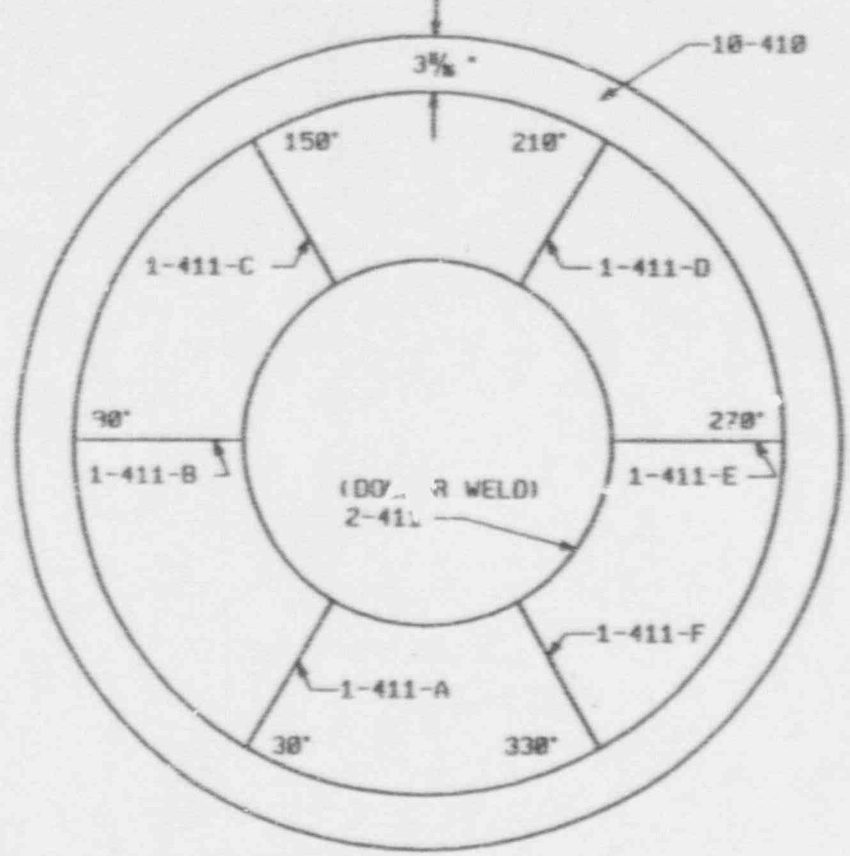
#3

REF.DWGS.
E-232-408-5

CONTAINMENT

FORT CALHOUN STATION

I.S.I. ISOMETRIC
A-1



REACTOR PRESSURE VESSEL LOWER HEAD

CONTROLLED COPY
DO NOT REMOVE COPIES FROM THIS AREA

DATE	TASK/MR #

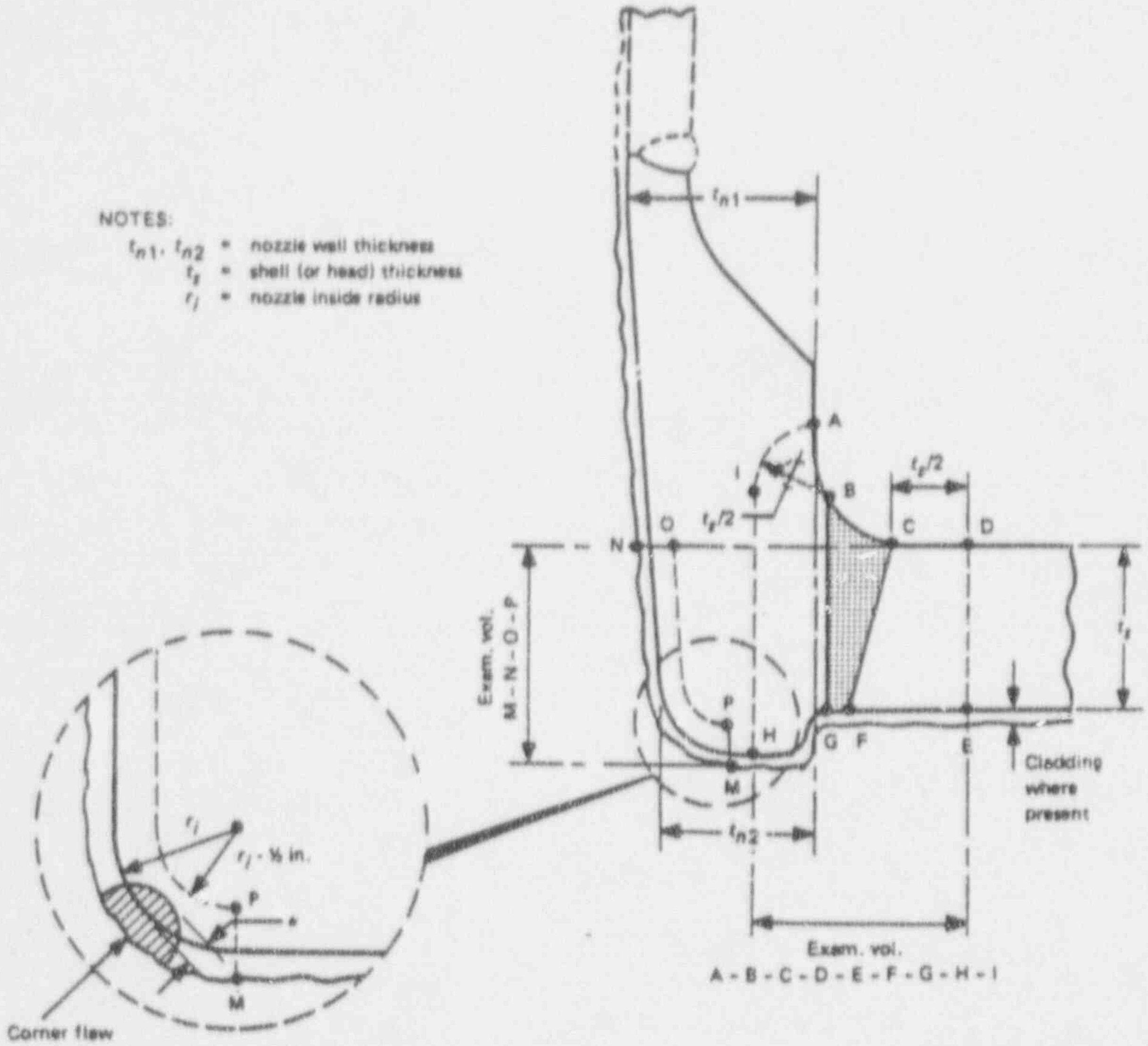
4

CONTAINMENT

REF. DWGS.
E-232-439

FORT CALHOUN STATION
I.S.I. ISOMETRIC
A-2

NOTES:
t_{n1}, t_{n2} = nozzle wall thickness
t_s = shell (or head) thickness
r_i = nozzle inside radius



EXAMINATION REGION[1]	EXAMINATION VOLUME[2]
Shell (or head) adjoining region	C-D-E-F
Attachment weld region	B-C-F-G
Nozzle cylinder region	A-B-G-H-I
Nozzle inside corner region	M-N-O-P

NOTES:
(1) Examination regions are identified for the purpose of differentiating the acceptance standards in IWB-3512.
(2) Examination volumes may be determined either by direct measurements on the component or by measurements based on design drawings.

W80

FIG. IWB-2500-7(a) NOZZLE IN SHELL OR HEAD
(Examination Zones in Barrel Type Nozzles Joined by Full Penetration Corner Welds)

5

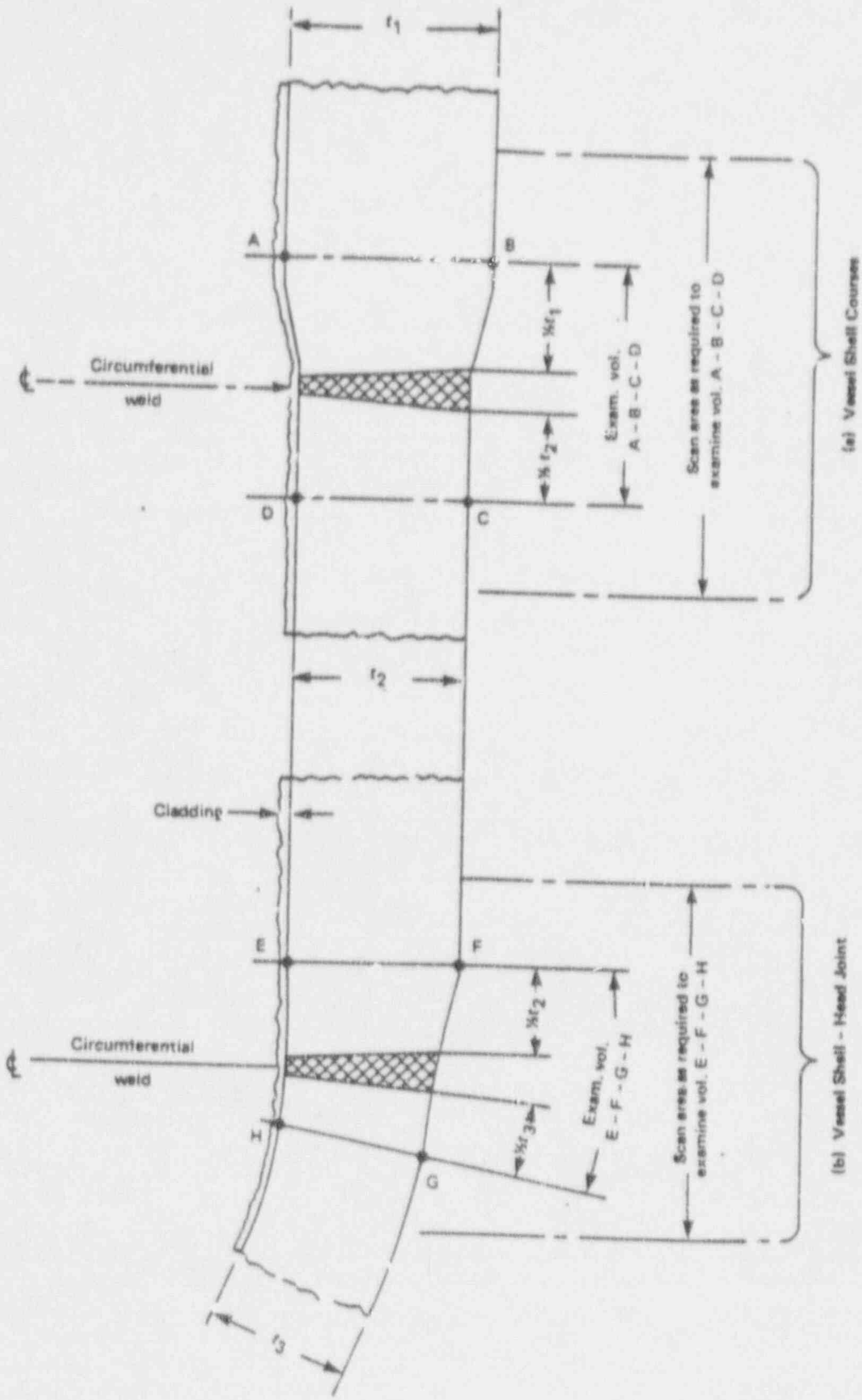


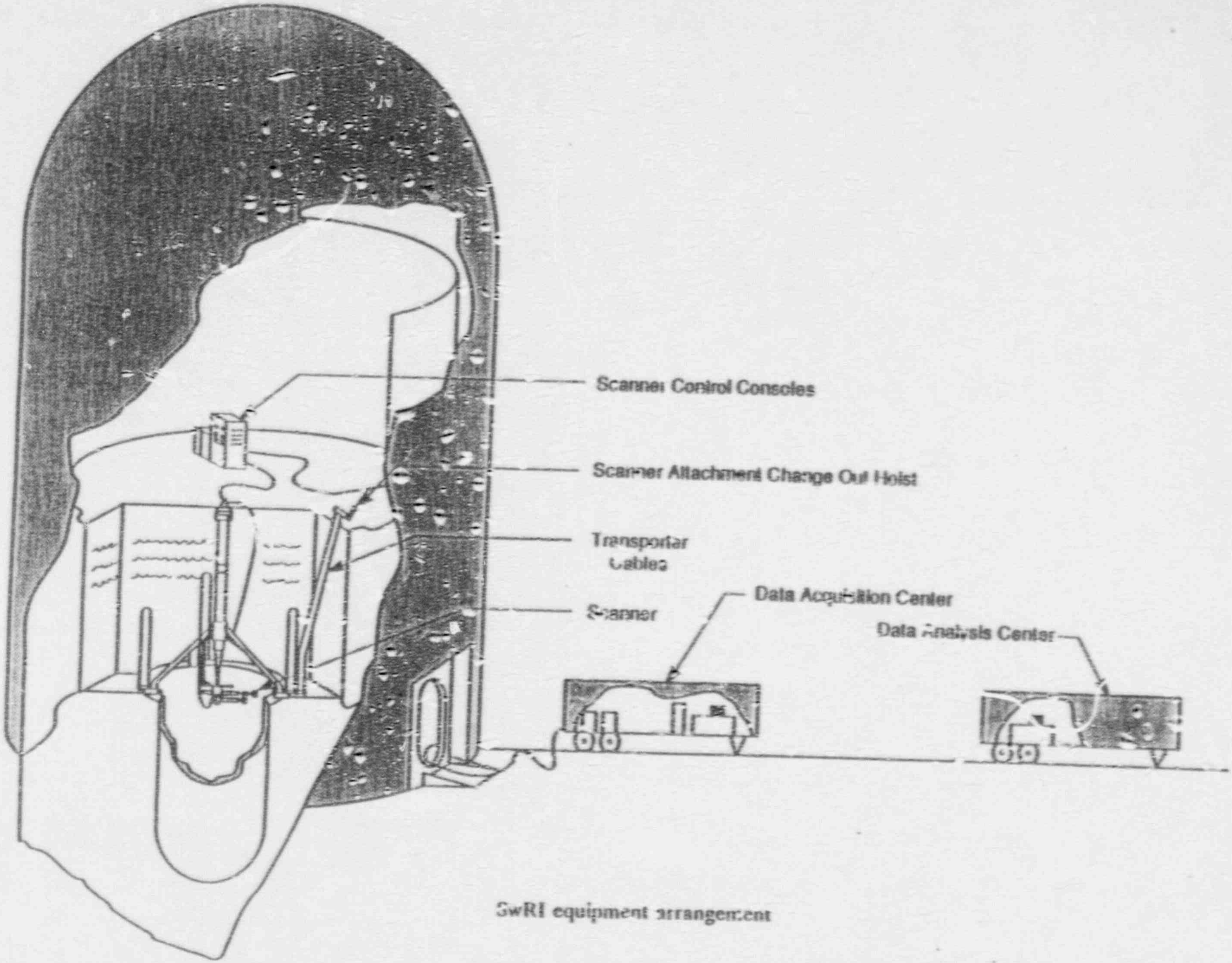
FIG. IWB-2500-1 VESSEL SHELL CIRCUMFERENTIAL WELD JOINTS

#6

REACTOR VESSEL EXAMS PERFORMED BY SOUTHWEST RESEARCH INSTITUTE (SwRI)

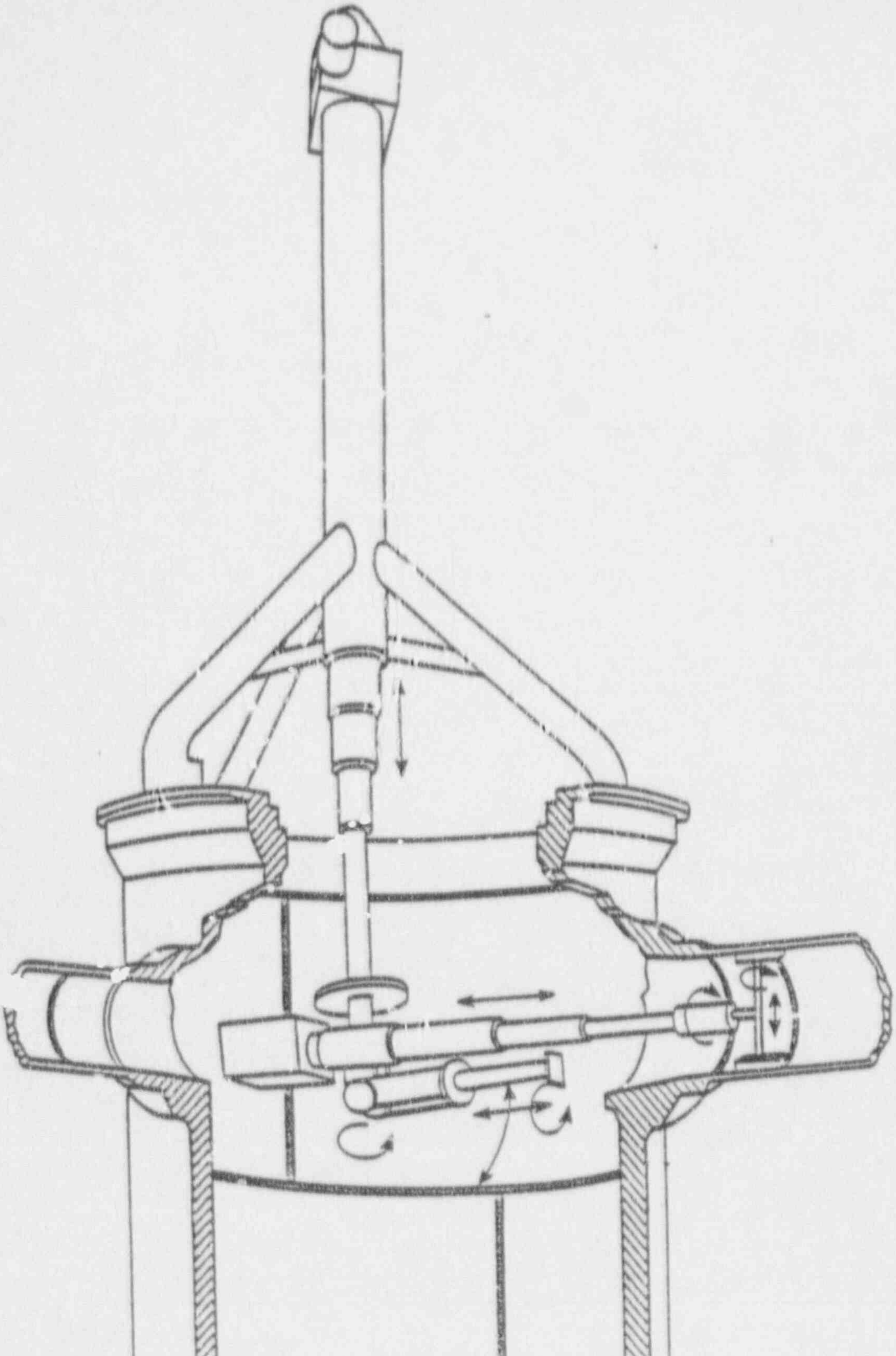
(KEY FOR TERMS AT END OF LIST; 1/17/92)

ISO #	Component ID	MWR #	MWO #	ITEM #	Cat. #	Meth.	ROOM	INSUL.	TIMES	FLOOR	ELEV.	COMMENTS
A-01	RPV-A-11	9101168		B1.30	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV SHELL-FLANGE WELD (FCL-5, FCL-35)
A-01	RPV-N1-1	9101168		B13.10	B-N-1	VT3	C	N/A	SwRI*	1045'	1045'	RPV VESSEL INTERIOR
A-01	RPV-N-1-A	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV OUTLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV-N-1-A-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV OUTLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-N-1-B	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV OUTLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV- -1-B-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV OUTLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-N-2-A	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV-N-2-A-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-N-2-B	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV-N-2-B-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-N-2-C	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV-N-2-C-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-N-2-D	9101168		B3.90	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE WELD (FCL-5, FCL-32)
A-01	RPV-N-2-D-IR	9101168		B3.100	B-D	UT	C	N/A	SwRI*	1045'	1045'	RPV INLET NOZZLE INNER RADIUS (FCL-31)
A-01	RPV-SC-B-11	9101168		B1.11	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV CIRCUMFERENTIAL WELD (7-FCL)
A-01	RPV-SC-C-11	9101168		B1.11	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE CIRCUMFERENTIAL WELD (7-FCL)
A-01	RPV-SC-16-410	9101168		B1.11	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV CIRCUMFERENTIAL WELD (7-FCL)
A-01	RPV-SL-A-1	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-A-2	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-A-3	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-B-1	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-B-2	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-B-3	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-C-1	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-C-2	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01	RPV-SL-C-3	9101168		B1.12	B-A	UT	C	N/A	SwRI*	1045'	1045'	RPV BELTLINE LONGITUDINAL WELD (7-FCL)
A-01A	RPV-G1-FL-01/48*	9101168		B6.40	B-G-1	UT	C	N/A	SwRI*	1045'	1045'	RPV FLANGE LIGAMENT AREAS (5-FCL)
A-02	RPVLH-1-411-B	9101168		B1.22	B-A	UT	C	N/A	SwRI*	1045'	1045'	LOWER HEAD MERIDIONAL WELD (6-FCL)
A-02	RPVLH-2-411	9101168		B1.21	B-A	UT	C	N/A	SwRI*	1045'	1045'	LOWER HEAD DOLLAR WELD (6-FCL)
A-08	MRC-1/01	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV [illegible]LET NOZZLE-SAFE END WELD (FCL-9, FCL-34)
A-09	MRC-2/01	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV OUTLET NOZZLE-SAFE END WELD (FCL-9, FCL-34)
A-08	MRC-1/02	9101168		B9.11	B-J	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV OUTLET SAFE END-PIPE WELD (FCL-9, FCL-64)
A-09	MRC-2/02	9101168		B9.11	B-J	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV OUTLET SAFE END-PIPE WELD (FCL-9, FCL-64)
A-08	MRC-1/18	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET SAFE END-NOZZLE WELD (FCL-10, FCL-33)
A-08	MRC-1/30	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET SAFE END-NOZZLE WELD (FCL-10, FCL-33)
A-09	MRC-2/18	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET SAFE END-NOZZLE WELD (FCL-10, FCL-33)
A-09	MRC-2/30	9101168		B5.10	B-F	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET SAFE END-NOZZLE WELD (FCL-10, FCL-33)
A-08	MRC-1/17	9101168		B9.11	B-J	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET PIPE-SAFE END WELD (FCL-10, FCL-66)
A-08	MRC-1/29	9101168		B9.11	B-J	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET PIPE-SAFE END WELD (FCL-10, FCL-66)
A-09	MRC-2/17	9101168		B9.11	B-J	UT-S/V	C	N/A	SwRI*	1045'	1045'	RPV INLET PIPE-SAFE END WELD (FCL-10, FCL-66)
A-09	MRC-2/29	9101168		B9.11	B-J	UT-S/V	[illegible]	N/A	SwRI*	1045'	1045'	RPV INLET PIPE-SAFE END WELD (FCL-10, FCL-66)



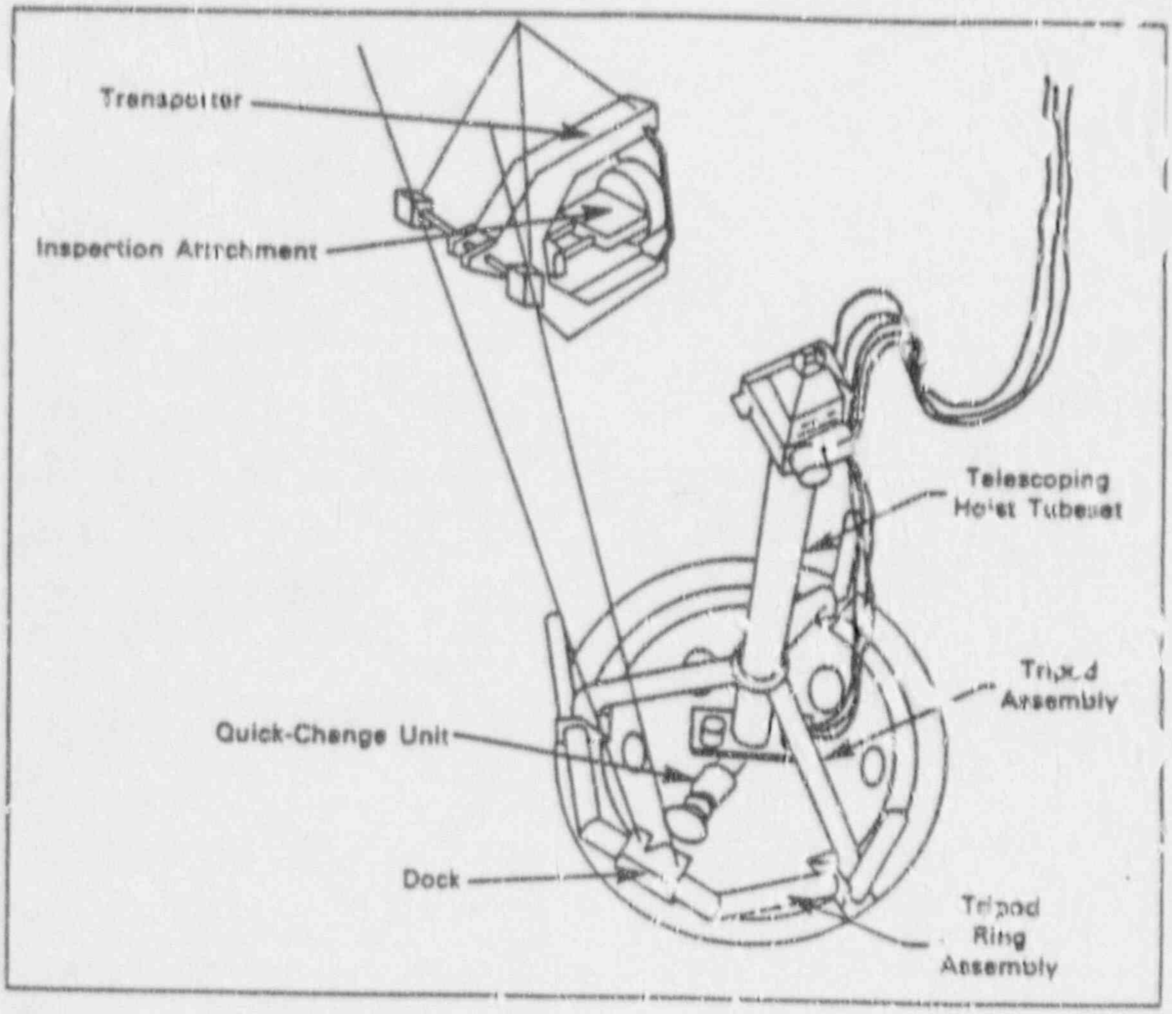
SwRI equipment arrangement

#8



Conceptual drawing of the PaR ISI-2 device

#19



Fast PaR ISI-2 Device

#10

SYSTEM REPORT CARDS

SYSTEM REPORT CARDS

- Purpose/Frequency
- Content
- Review Process
- Follow-Up Actions
- Status and Long Term Benefits

SYSTEM REPORT CARD CONTENT

I. Start and End Dates of Report Cards

II. Items or Events - Past

III. Items or Events - Future

IV. Management Attention Items

V. List and Status:

- LERs
- Special Reports
- Temporary Modifications
- Modifications
- Engineering Change Notices
- Engineering Assistance Requests
- Maintenance Work Orders (Numbers)
- Commitments
- Corrective Action Reports
- Facility License Changes
- Nonconformance Reports
- Safety Analysis for Operability
- NSRG Recommendations

VI. Attachments

PRESSURIZED THERMAL SHOCK

PRESSURIZED THERMAL SHOCK

DEFINITION:

- A PTS event is a low probability event which results from the introduction of cold water into a hot pressurized reactor vessel. If the reactor vessel is sufficiently embrittled due to neutron irradiation the sudden thermal stress induced could result in cracking and failure of the vessel.

APPLICABLE REGULATIONS AND REG GUIDES:

- 10 CFR 50.61 (PTS Rule)
- 10 CFR 50 Appendix G (Fracture Toughness Requirements)
- Reg. Guide 1.99, Rev. 02
- Reg Guide 1.154 (Plant Specific PTS Safety Analysis and Acceptance Criteria)

MITIGATION STRATEGIES:

- Fuel Management
- Reg. Guide 1.154 Analysis
- Reactor Vessel Annealing

FORT CALHOUN STATION (FCS) HISTORY:

1981 FCS identified as plant with "High" Embrittlement Rate (and undesirable weld chemistries).

1983 Cycle 8 implemented low radial leakage fuel management.

1984 Shadow Shield Test unsuccessful.

1985 Reactor Vessel weld 2-410 sample obtained and analyzed.

1986
a. Cycle 10 implemented extreme low radial leakage fuel management with part length poison rods in 16 fuel assemblies.

b. PTS Rule submittal - screening criterion (RT_{PTS}) would not be reached until 54 EFPY.

1987
a. 5 year license extension request.

b. Cycle 11 Low Radial Leakage Fuel Management.

1988
a. Reg Guide 1.99, Rev. 2, issued.

b. CEOG Task for Reg. Guide 1.154 analysis.

1991
a. Cycle 14 optimized extreme low radial leakage core designed (see figure). Aggressive flux reductions:

- Hafnium Rods (12 assemblies).
- 4 natural uranium assemblies.
- IFBA Fuel Rods.

b. Submittal for revised PTS Rule - Reach screening criterion in 2009.

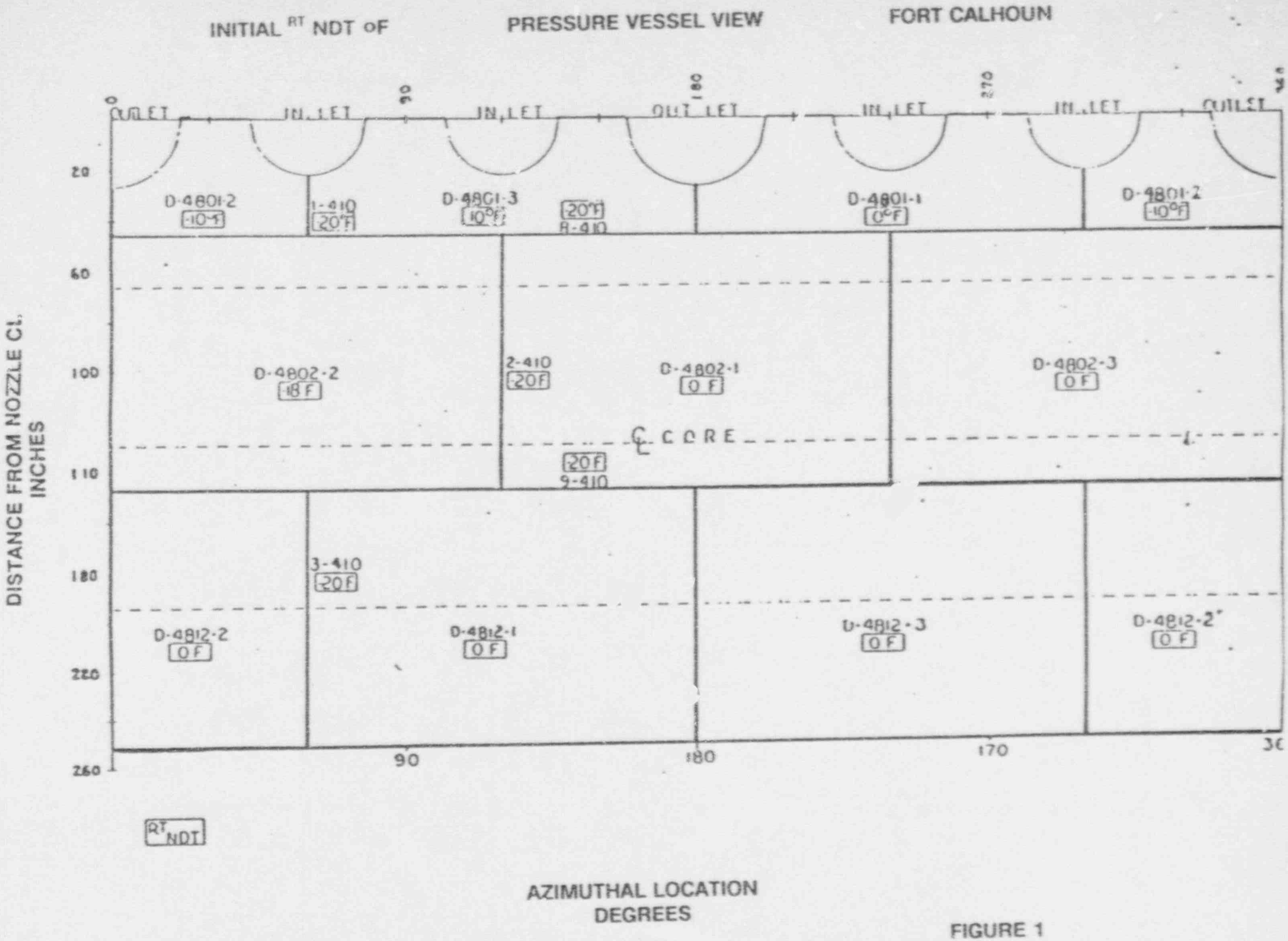
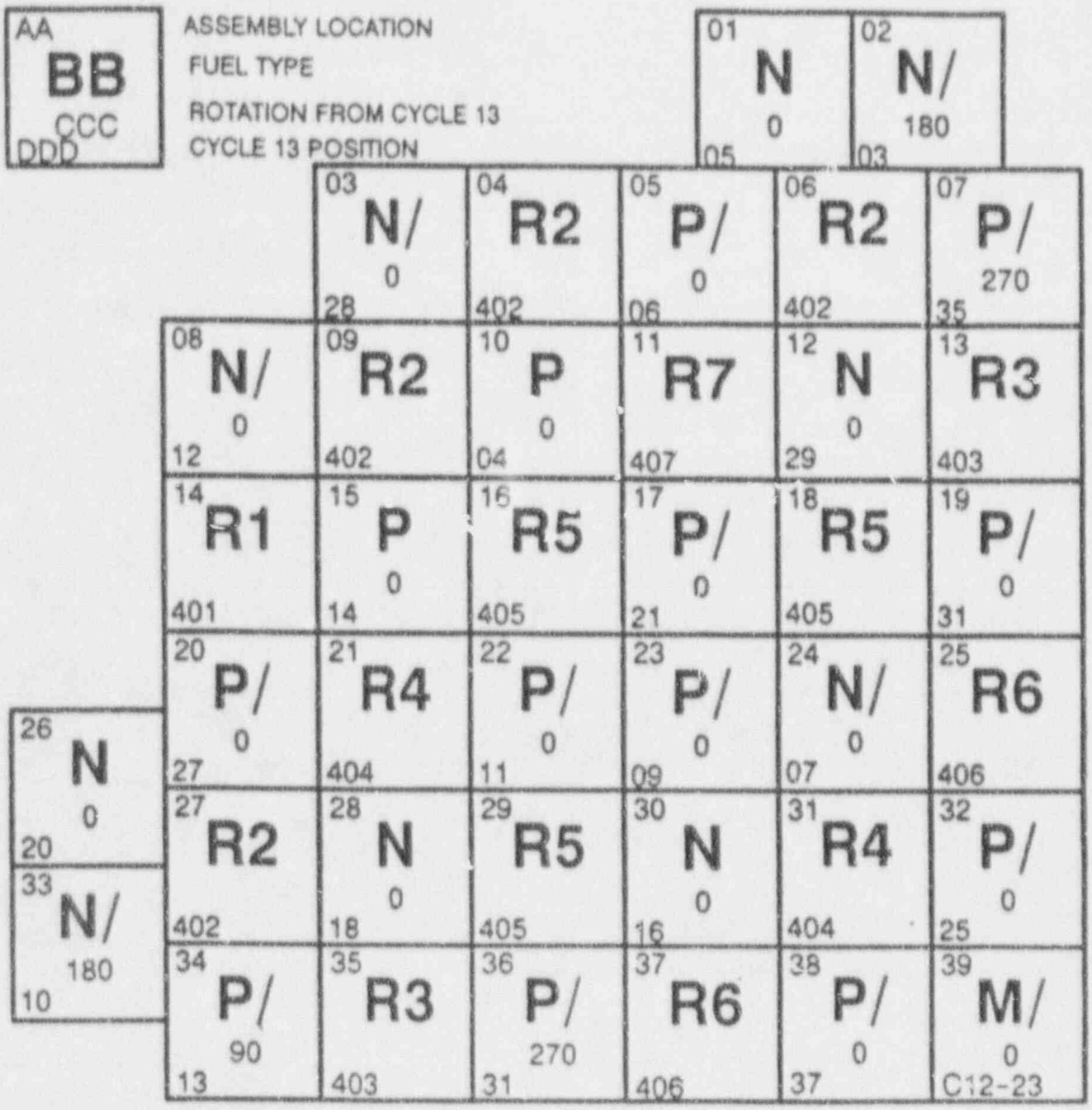

INITIAL RT NDT oF
PRESSURE VESSEL VIEW
FORT CALHOUN
OUTLET
INLET
INLET
OUTLET
INLET
INLET
OUTLET
D-4801-2
D-4801-3
D-4801-1
1-410
20°F
8-410
D-4802-2
18 F
2-410
20F
D-4802-1
0 F
D-4802-3
0 F
CORE
9-410
3-410
D-4812-2
D-4812-1
D-4812-3
DISTANCE FROM NOZZLE CL. INCHES
AZIMUTHAL LOCATION DEGREES
RT NDT

FIGURE 1

CYCLE 14

FIGURE 2



R1 – 0.74% w/ 0 IFBA, 4 Assys

R2 – 3.85% w/ 28 IFBA, 16 Assys

R3 – 3.85% w/ 48 IFBA, 4 Assys

R4 – 3.85% w/ 64 IFBA, 8 Assys

R5 – 3.85% w/ 84 IFBA, 12 Assys

R6 – 3.60% w/ 84 IFBA, 4 Assys

R7 – 3.60% w/ 64 IFBA, 4 Assys

Full Length Hf Poison Rods (QC Assy 1,2,8)

52-Batch, Reflective Symmetry, EOC Burnup – 14,000 MWD/MTU

CURRENT STATUS:

- Cycle 14 type fuel management expected to allow operation to at least 2013.

- DOT 4.3 analysis underway to quantify above.

- CEOG Task 632 (Reg Guide 1.154 analysis) reports on fracture mechanics, T/H methodology, and event sequences, under review.

FUTURE:

- Continued use and further optimization of extreme low radial leakage fuel management.

- FCS specific Reg. Guide 1.154 analysis completion.

- Reactor Vessel Annealing - continue to track and evaluate.

- Potential Rule Changes (SECY 91-333).

SUMMARY:

- OPPD has taken aggressive actions to minimize the rate of reactor vessel embrittlement.

- The 10 CFR 50.61 screening criterion of 270°F will not be reached before expiration of operating license in 2008, due to the implementation of PTS mitigating fuel management.

- Expect to be able to operate to at least 2013.

- Evaluating other options for lifetime extension.

PROBABILISTIC RISK ASSESSMENT
(PRA)

BACKGROUND

In response to Generic Letter 88-20 Supplement 1, OPPD is performing a full Level III PRA with external events.

- Comprehensive Program.
- Provide a full risk picture/profile.
- Experienced OPPD PRA Group with extensive plant knowledge.
- Provide Basis for "Living PRA" (ISAP).

IPEEE

In response to Generic Letter 88-20 Supplement 4, OPPD committed to provide an IPEEE milestone schedule when the NRC Supplemented Safety Evaluation Report (SSER) on Evaluation of USI A-46, Seismic Qualification Guidelines, is issued. OPPD plans to perform:

- Fire PRA with EPRI
- Enhanced margins approach for seismic analysis to be coordinated with the resolution of USI A-46
- Full risk treatment of remaining events

SCHEDULE

From OPPD, response to Generic Letter 88-20, Supplement 1:

- Initial Level I models developed - August 1, 1991 COMPLETE.
- Limited Level II models developed and interfaced with Level I - April 1, 1992.
- Final Level I/II/III models - May 5 [illegible]
- Submittal to NRC - December 1, 1993.

STATUS

- 58% complete with total project (not counting reports and update required for NRC submittal).
- Second revision of Level I system models and notebooks (which includes fault trees) is complete and under 3rd revision.
- First revision of Level II documents and models is underway.

SELECTED APPLICATIONS

- Design Discrepancy Resolution/Support
  + SIRWT Discharge Isolation Valves
  + LOCA / CCW Failure

- Licensed Operator Training
  + Internal Flooding Analysis
  + Shutdown Risk

- Outage Operations Support
  + Shutdown Risk Scenario

- Engineering Modifications/Design
  + Third AFW Pump
  + Internal Flooding Analysis

SUMMARY

- Fully documented, state-of-the-art living PRA.
- Will provide a powerful tool for enhancing plant operation, training, maintenance and design.
- Aggressive, pro-active approach that exceeds requirements of Generic Letter 88-20.