JUL 28 1982

Docket Nos: 50-369
and 50-370

APPLICANT: Duke Power Company

FACILITY: McGuire Nuclear Station, Units 1 and 2

SUBJECT: SUMMARY OF MEETING HELD ON JULY 14, 1982

A meeting was held with the Duke Power Company on July 14, 1982, in Bethesda, Maryland. The purpose of the meeting was to discuss examinations and evaluations performed by Duke Power Company regarding detached thermal sleeves. A list of attendees is presented in Enclosure No. 1. The agenda for the meeting is shown on Enclosure No. 2. Enclosure No. 3 presents viewgraphs used during the licensee presentation.

The McGuire Nuclear Station, Unit 1, was shutdown on June 23, 1982, for purposes of eddy current testing of all Model D steam generators. Pursuant to recent evidence of the degradation of thermal sleeve components in the reactor coolant system of the Trojan plant, another Westinghouse plant, the licensee promptly initiated an inspection of all thermal sleeves in the Unit 1 reactor coolant system utilizing radiography techniques. The RCS contains the following thermal sleeves: (4) -10" accumulator nozzle cold leg, (2) -3" charging nozzle cold legs and (1) -14" pressurizer surge nozzle hot leg.

The radiograph of the 10" accumulator nozzle thermal sleeve on loop B revealed that it was detached and missing. This was confirmed by the licensee by a visual inspection with a small TV camera going through the upstream check valve on the 10" line. The licensee has inspected all other connections to the RCS having similar thermal sleeves to assure that the remaining sleeves were in place with their welds intact. Only one detached (B loop) thermal sleeve was evidenced by the licensee's inspection. The licensee reported that the loop B 10" line welds located at the top of the sleeve had failed at the interface to the nozzle wall with possibly only a small portion of one weld remaining on the nozzle wall. The nozzle wall, according to the licensee, showed no indication that the sleeve had broken apart prior to being released. As a result of monitoring of the lower reactor vessel area for loose parts during low flow and normal RCS flow conditions, the licensee concludes that the missing 10" thermal sleeve is located in the lower reactor internals and that the small impacts during low flow conditions are of a minor nature. No movement is indicated when full flow is present. The licensee considers this to indicate that the sleeve is parked and remains fixed in place.

The licensee presented the results of detailed stress analysis for the accumulator and other nozzles without thermal sleeves as well as a description of the impact and wedging effect of a loose thermal sleeve on reactor internals. The potential hydraulic effects of the detached thermal sleeve was also discussed. The licensee by letter dated July 13, 1982, documented all of the information provided during the meeting.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

As a result of discussions with the licensee and Westinghouse and review of the information provided in its July 13, 1982, letter, the staff, at the conclusion of the meeting, indicated that continued operation of McGuire Unit 1 until the next refueling outage or outage of sufficient duration, with the thermal sleeves as presently positioned is acceptable without undue risk to the health and safety of the public.

/S/

Ralph A. Birkel, Project Manager
Licensing Branch No. 4
Division of Licensing

Enclosures:
As stated

cc: See next page

OFFICE	DL:LB #4 RAB	DL:LB #4	DL:LB#				
SURNAME	RBirkel/hmc	EAdensam	MDUNCAN				
DATE	7/22/82	7/22/82	7/23/82				

ATTENDANCE LIST
McGUIRE NUCLEAR STATION, UNITS 1 AND 2
July 14, 1982

Duke Power Company

W. M. Sample
G. S. Lanady
W. O. Parker, Jr.
Hal B. Tucker
N. A. Rutherford
R. E. Harris

Westinghouse

M. J. Zegar
W. C. Gangloff
J. c. Hoelsel
D. G. Maire
D. W. Alexandra
R. W. Beer
Don White
W. R. Spezialetti
F. J. Twogood
K. C. Chang

NRC Staff

R. A. Birkel
R. A. Purple
W. S. Hazelton
Joe Holonich
R. L. Tedesco
S. Hou
R. Bosnak
C. D. Sellers
H. F. Conrad
L. Frank
L. Engle
E. Murphy
C. Y. Cheng
C. Trammell
E. Adensam
F. Orr
Wm. J. Collins (IE-HQ)
P. R. Bemis (RII)
A. R. Herdt (RII)
J. A. Olshinski (RII)

Other

Jack McEwen (Tech. Services Internationai)
D. W. Lippard (VEPCo)
E. R. Smith, Jr. (VEPCo)

McGuire

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Mr. Paul Bemis
Senior Resident Inspector
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Cornelius, North Carolina 28013

James P. O'Reilly, Regional Administrator
U.S. Nuclear Regulatory Commission,
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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McGUIRE NUCLEAR STATION, UNITS 1 AND 2
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P. R. Bemis (RII)
A. R. Herdt (RII)
J. A. Olshinski (RII)

Other

Jack McEwen (Tech. Services International)
D. W. Lippard (VEPCo)
E. R. Smith, Jr. (VEPCo)

Enclosure 2

AGENDA FOR MEETING WITH NRC

McGuire 1

Reactor Coolant System Thermal Sleeves

July 14, 1982

I. Introduction - W. O. Parker, Duke Power Co.

II. Statement of Problem

A. Description of Thermal Sleeves - Marty Zegar, W

B. Thermal Sleeves - Inspection and Loose Part Monitoring Results - Morris Sample, Duke Power

III. Safety Evaluation

A. Nozzle Integrity without Thermal Sleeve - Dr. Ken Chang, W

B. Loose Thermal Sleeves - Dave Maire, W

1. Impact with RCS Components

2. Flow Maldistribution

IV. Proposed Action by Licensee - Morris Sample, Duke Power

A. Augmented Monitoring for Loose Parts

B. Additional Operator Training

C. Augmented Inspections

D. Schedule for Plant Operation - Hal Tucker, Duke Power

1. Removal of Loose Thermal Sleeve

2. Removal of Installed Thermal Sleeves

V. Conclusions

DUKE THERMAL SLEEVE

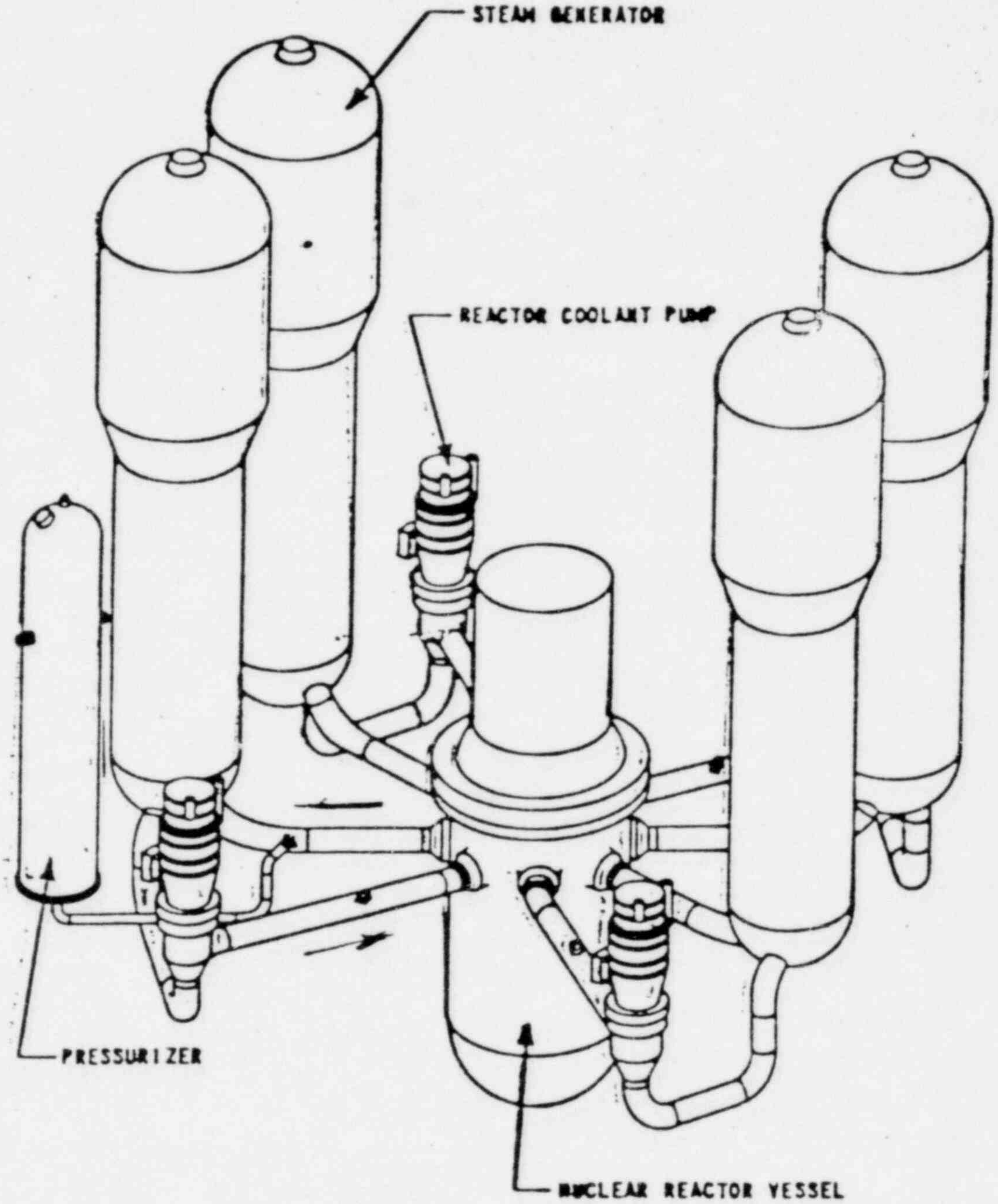
INVENTORY

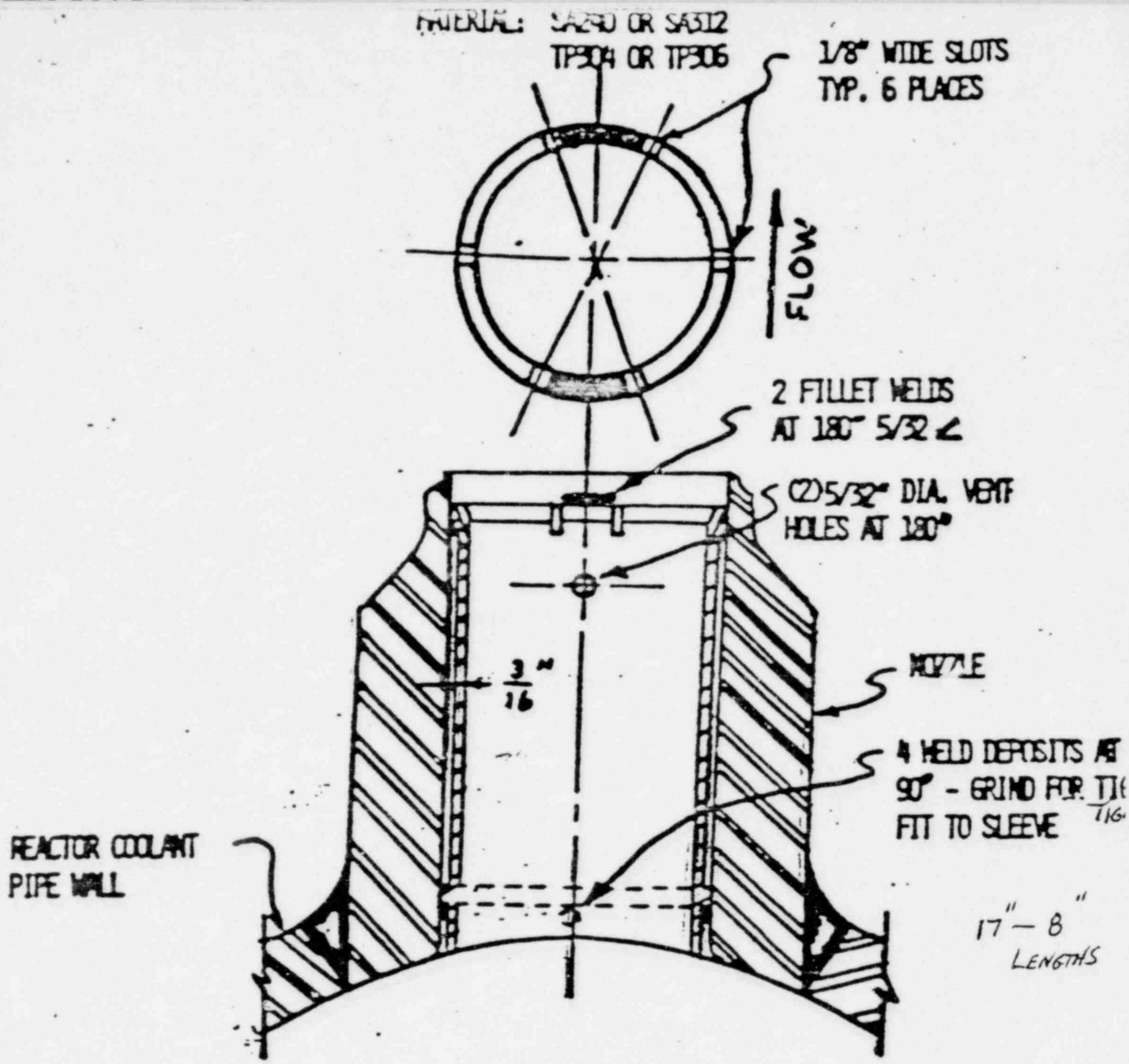
- REACTOR COOLANT PIPING

    (4) 10" ACCUMULATOR NOZZLES COLD LEGS

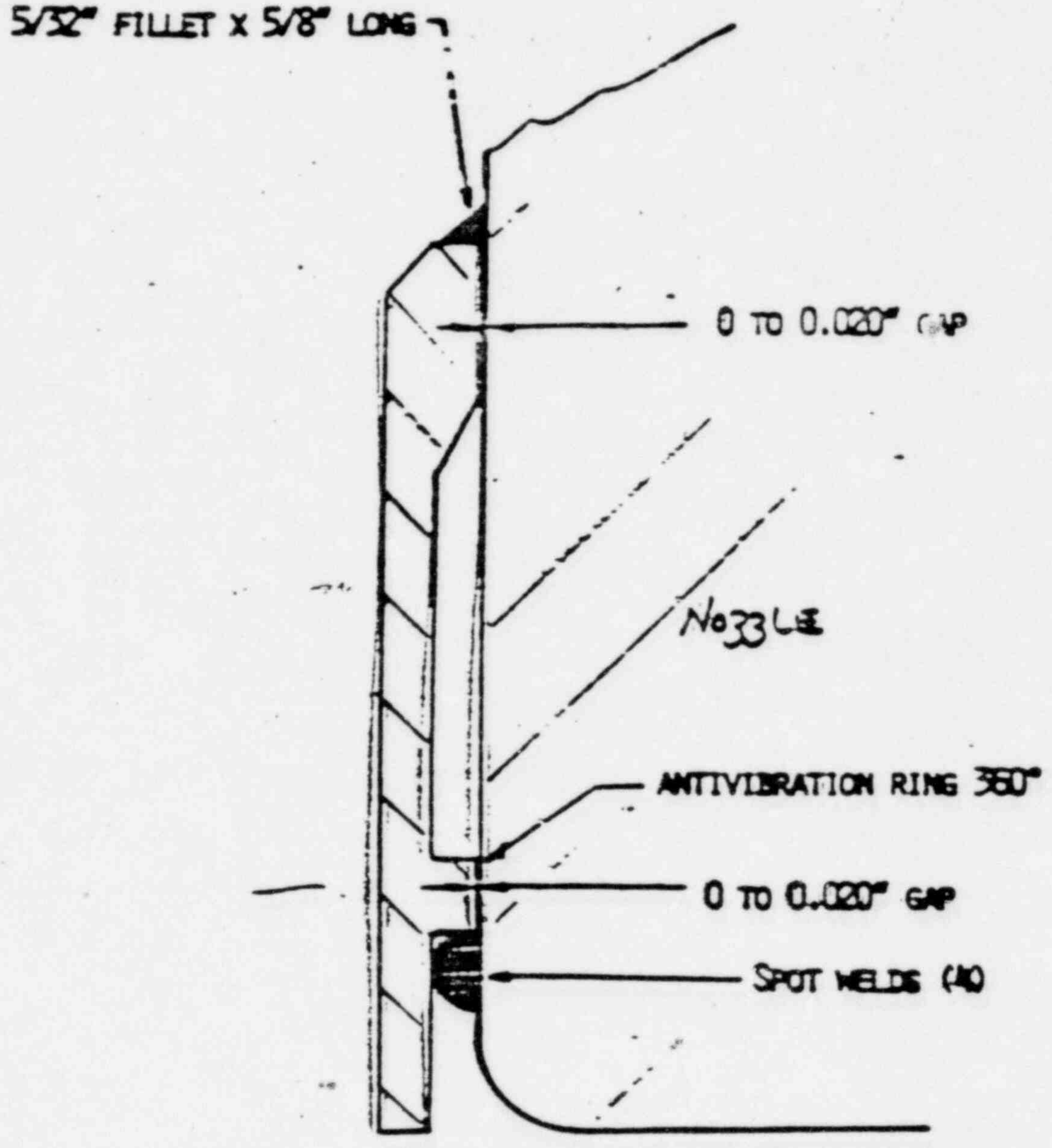
    (2) 3" CHARGING NOZZLES COLD LEGS

    (1) 14" PRESSURIZE SURGE NOZZLE HOT LEG


STEAM GENERATOR
REACTOR COOLANT PUMP
PRESSURIZER
NUCLEAR REACTOR VESSEL


MATERIAL: SA240 OR SA312
TP304 OR TP306
1/8" WIDE SLOTS
TYP. 6 PLACES
FLOW
2 FILLET WELDS
AT 180° 5/32
(2) 5/32" DIA. VENT
HOLES AT 180°
3/16"
NOZZLE
4 WELD DEPOSITS AT
90° - GRIND FOR TI
TIG
FIT TO SLEEVE
REACTOR COOLANT
PIPE WALL
17" — 8"
LENGTHS

THERMAL SLEEVE DESIGN

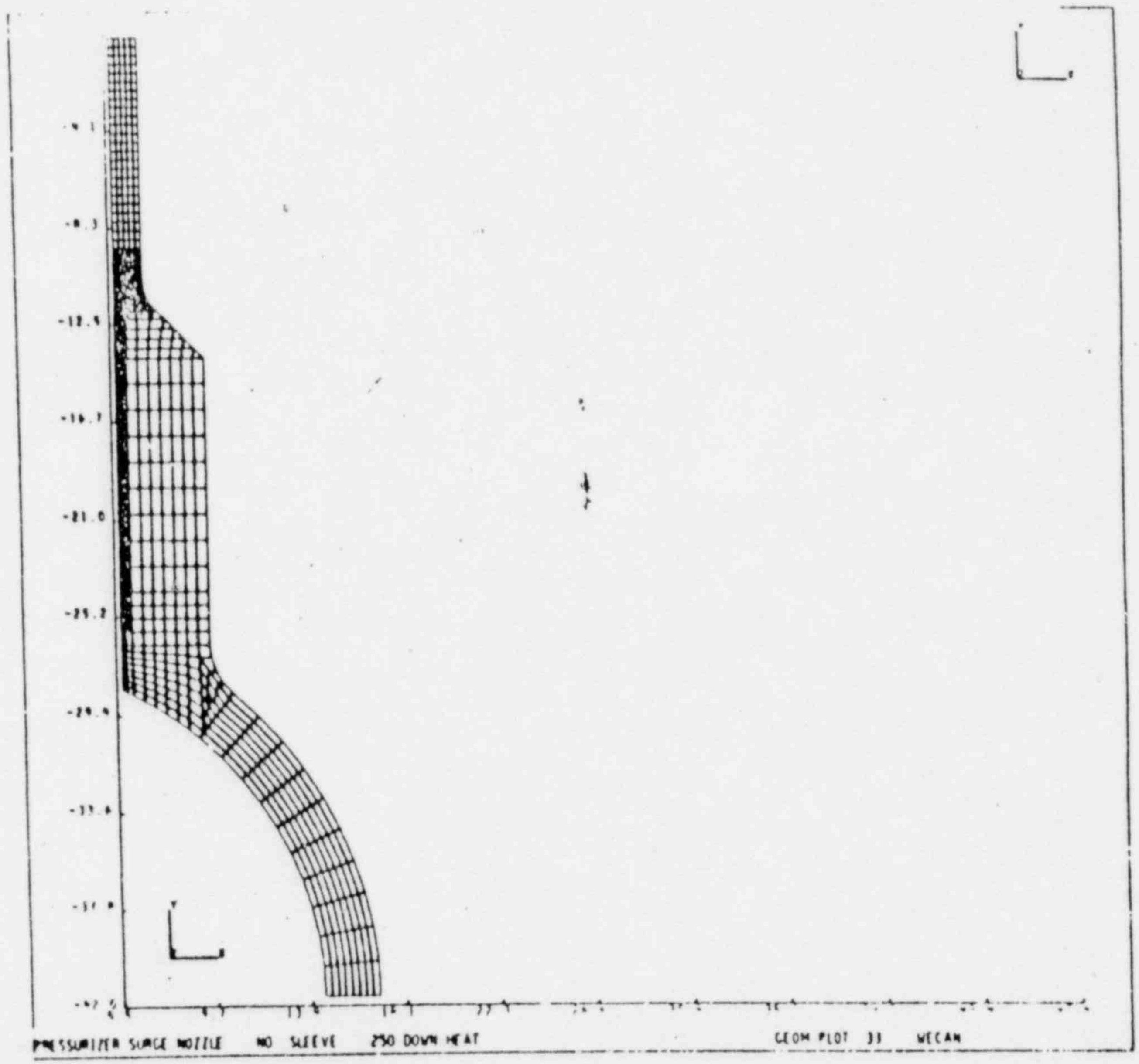


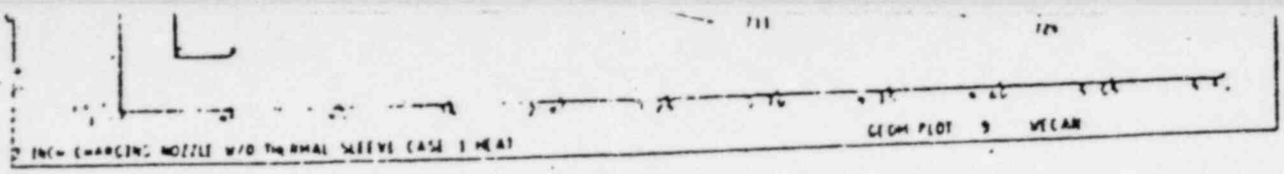
INSPECTION PROGRAM SUMMARY

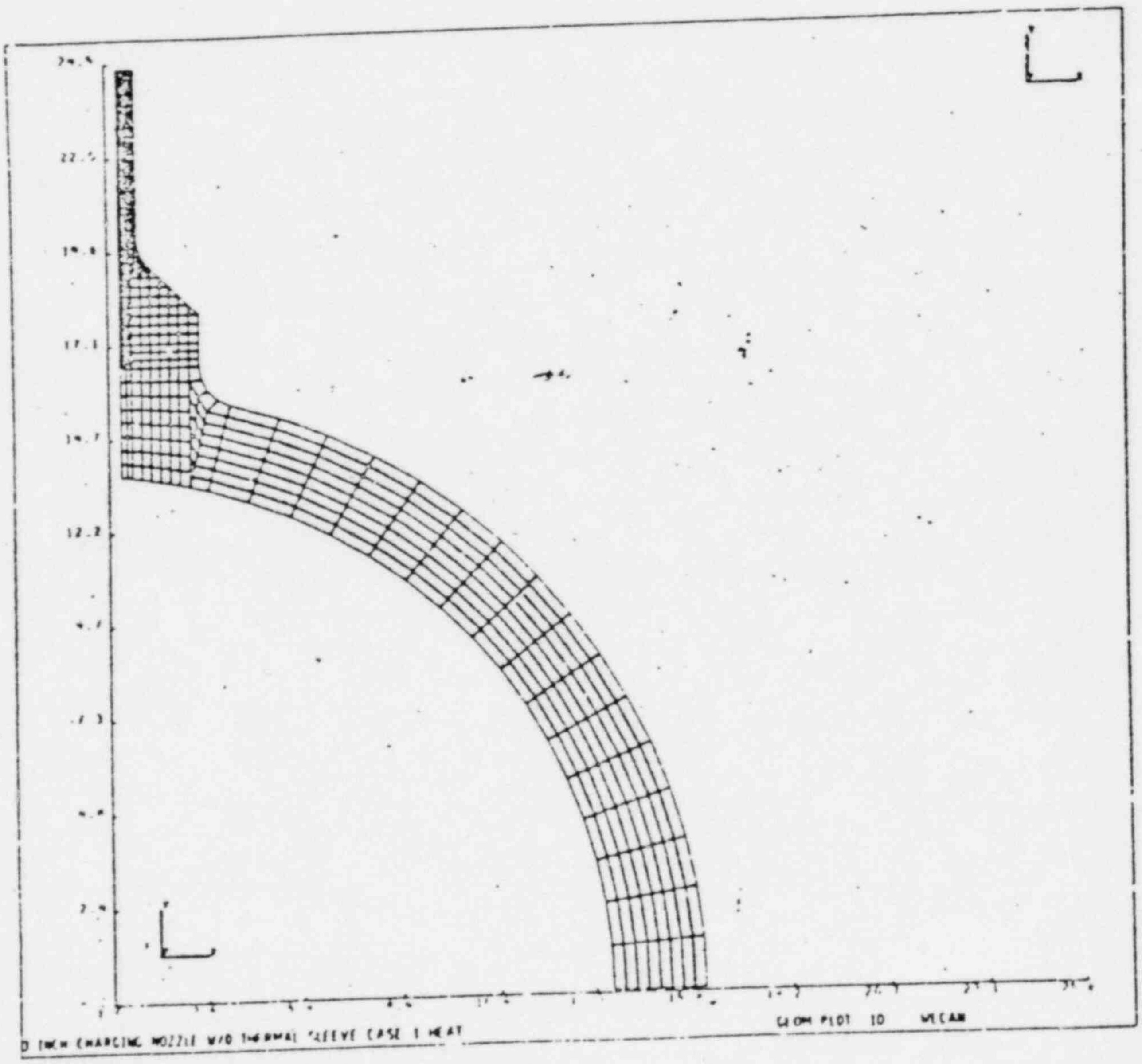
July 1	-	Radiographic examination of 10" lines began.
July 1-4	-	RT results indicated thermal sleeves in place on Loops A, C, and D.
July 5	-	RT results on Loop B 10" line indicated thermal sleeve missing. Confirmed by visual inspection early on July 6.
July 6	-	Radiographic examination on 3" lines on Loop A and D showed sleeves in place and welds intact.
July 7	-	Additional visual inspection of Loop B 10" line showed no sleeve pieces remaining in nozzle.
July 7	-	Radiographic examination of 14" line on Loop B showed sleeve in place and welds intact.
July 8	-	Additional RT on remaining sleeves in 10" lines showed welds intact.
July 10	-	During operation of decay heat removal system minor impacts in lower reactor vessel internals.
July 11	-	During reactor coolant pump runs minor impacts with one pump running, no movement with all 4 pumps running.

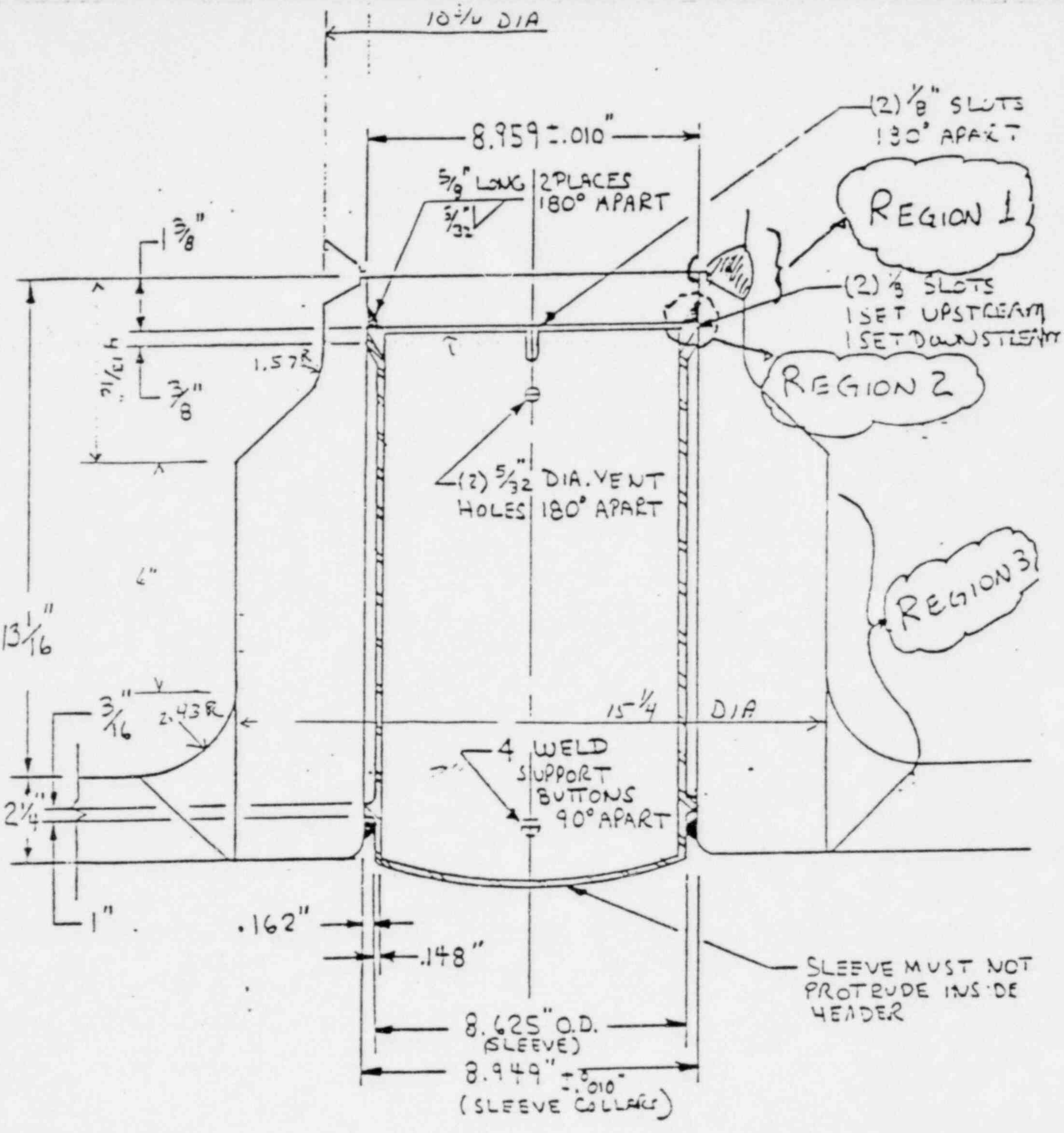
EVALUATION OF RCL NOZZLE WITHOUT THERMAL SLEEVE

- NOZZLES
- GEOMETRY AND MATERIAL
- MODELING
  - 1-D FINITE DIFFERENCE
  - 2-D FINITE ELEMENT
- TRANSIENT DEFINITION
- HEAT TRANSFER ANALYSIS
- STRESS ANALYSIS


-12.5
-16.7
-21.0
-25.2
-29.4
PRESSURIZER SURGE NOZZLE NO SLEEVE 250 DOWN HEAT
GEOM PLOT 33 WECAN


GEOM PLOT 9 WECAN
2 INCH CHARGING NOZZLE W/O THERMAL SLEEVE CASE 1 HEAT


24.5
22.0
19.6
17.1
14.7
12.2
GEOM PLOT 10 WECAN
2 INCH CHARGING NOZZLE W/O THERMAL SLEEVE CASE 1 HEAT



10" THERMAL SLEEVE ON 27 1/2" HEADER

THERMAL SLEEVE APPROX. LENGTH = 13 15/16"

FIGURE 3.1

FATIGUE EVALUATION

- EXTERNAL LOADS
- CRITICAL STRESS LOCATION
- STRESS CONCENTRATION FACTORS
- MATERIALS
- ASME FATIGUE EVALUATION

CONCLUSIONS

3" CHARGING NOZZLE - 2-D MODEL

- BUTT WELD
- TACK WELD
- CROTCH REGION

$\Sigma U < 1.0$

14" SURGE NOZZLE - 2-D MODEL

- BUTT WELD
- CROTCH REGION

$\Sigma U < 1.0$

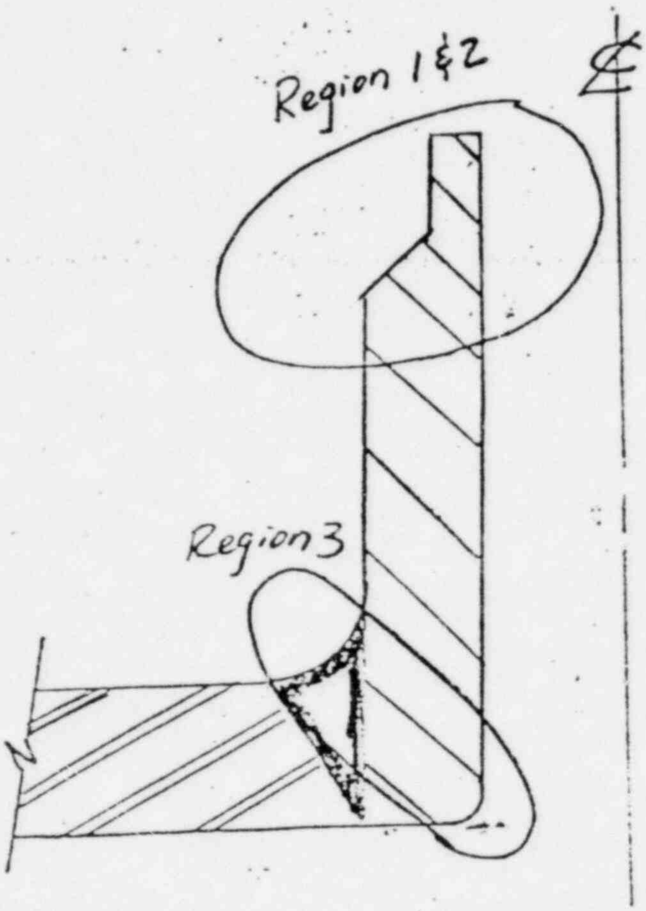
- TACK WELD ----

10" ACCUMULATOR NOZZLE - 2-D & 3-D MODELS

- BUTT WELD - 2-D & 3-D MODELS
- CROTCH REGION
  - 3-D MODEL - 10" 45° NOZZLE
  - 2-D MODEL - 6" 90° SIS NOZZLE

$\Sigma U < 1.0$

- TACK WELD


Region 1&2
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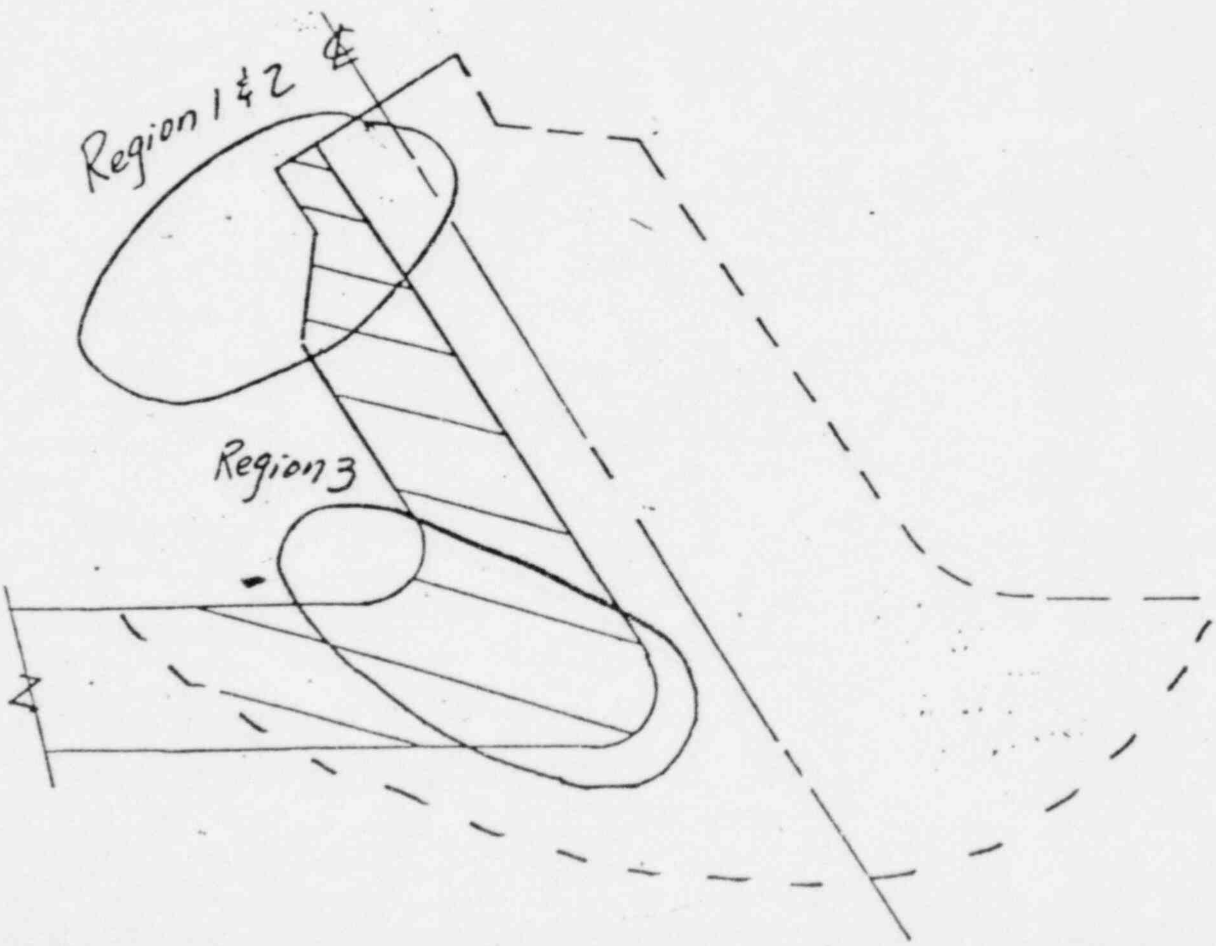
Region 3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Region 1&2
Region 3

SAFETY EVALUATION OF LOOSE THERMAL SLEEVES

I. Nozzle Integrity Without Sleeve

II. Mechanical Effects of Loose Object

- a) Reactor coolant pipe
- b) Steam generator
- c) Reactor coolant pump
- d) Reactor internals
- e) Reactor vessel
- f) Other RCS components
- g) Auxiliary systems
- h) Materials

III. Flow Blockage Effects

- Mechanical Effects on Reactor Coolant Pipe
  - Impact forces acceptable
  - Thermowells and RTD bypass line scoops located upstream of thermal sleeves

- MECHANICAL EFFECTS ON STEAM GENERATOR
    - IMPACT FORCES WILL NOT VIOLATE REACTOR COOLANT PRESSURE BOUNDARY
    - IMPACT FORCES WILL NOT VIOLATE DIVIDER PLATE PRESSURE BOUNDARY

- MECHANICAL EFFECTS ON REACTOR COOLANT PUMP
  - PUMP ISOLATED FROM HOT LEG SLEEVE BY STEAM GENERATOR TUBES
  - COLD LEG SLEEVES ARE DOWNSTREAM OF PUMP
  - REVERSE FLOW DURING STARTUP COULD MOVE OBJECTS TO PUMP. IMPACT FORCES UPON RESTART OF PUMP WOULD NOT VIOLATE REACTOR COOLANT PRESSURE BOUNDARY.

- MECHANICAL EFFECTS ON REACTOR INTERNALS (CONTINUED)

  - LOWER INTERNALS

    - IMPACT LOADS ON CORE BARREL ARE ACCEPTABLE

    - LOADS DUE TO OBJECT WEDGED IN RADIAL KEY GAPS ARE ACCEPTABLE

    - LOADS DUE TO OBJECT WEDGED UNDER SECONDARY CORE SUPPORT ARE ACCEPTABLE

    - IMPACT LOADS FROM LOOSE OBJECT IN LOWER INTERNALS PACKAGE ARE ACCEPTABLE

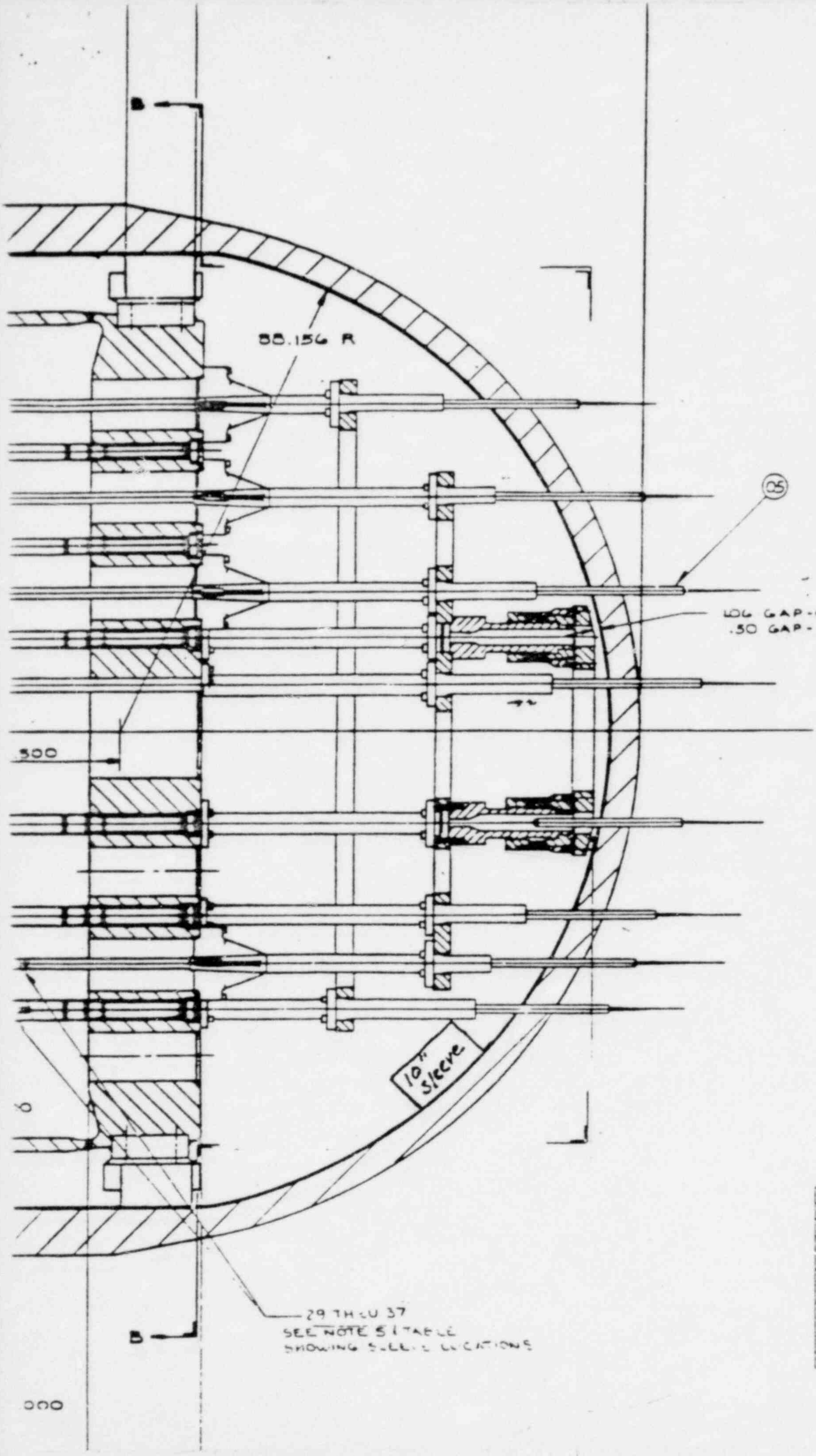
- MECHANICAL EFFECTS ON REACTOR INTERNALS

  - UPPER INTERNALS

    - REVERSE FLOW DURING STARTUP COULD MOVE OBJECTS TO UPPER INTERNALS. IMPACT FORCES ACCEPTABLE.

    - LARGE SIZE OF PIECES AND FULL FLOW PRIOR TO CRITICALITY WOULD PREVENT OBJECTS FROM ENTERING GUIDE TUBES

    - ROD STEP TESTS CONFIRM OPERABILITY OF CRDM DRIVE LINES. EXISTING ANALYSES CONSIDER ONE STUCK CRDM



INSTRUMENTATION GUIDE SLEE	
IT	LOCATION
29	A11, B3, B13, L15, N2, N14, R11
30	A9, D14, F1, J1, N13, P4, R6
31	H15, R8
32	B6, D3, F14, K2, N4
33	B8, C5, D12, H2, J14, L13, P9
34	C7, C8, F3, H3, H13, N6, N8
35	D8, D10, E5, E11, G12, H4, K12, L5
36	E9, G5, H11, K6, L8, L10
37	F7, F8, G9, H6, J7, J8, J10

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- Mechanical Effects on Reactor Vessel

    - Impact loads on pressure boundary components of reactor vessel are acceptable

    - Impact loads on bottom instrumentation nozzle penetrations could damage incore instrumentation thimble tubes. Low probability of thimble tube failure. Leakage would be detectable by operator via seal table area radiation detectors. Failure of three thimbles is within normal high pressure make-up capability.

- Mechanical Effects on Other RCS Components

    - No concern identified due to physical isolation from loose sleeves (e.g., safety valves, relief valves, etc.)

- Mechanical Effects on Auxiliary Systems

    - Low probability of objects entering auxiliary systems (CVCS and RHR)

    - Failure of single pump, valve, orifice, etc. would not create a safety concern

SAFETY EVALUATION CONCLUSION

- ADEQUATE ASSURANCE EXISTS THAT SAFE PLANT OPERATION IS NOT COMPROMISED BY FAILED THERMAL SLEEVES

- MATERIALS

  - NO UNACCEPTABLE MATERIALS INVOLVED

  - MINOR STAINLESS STEEL CLADDING DAMAGE ON RV AND SG PRESENTS NO SAFETY CONCERN

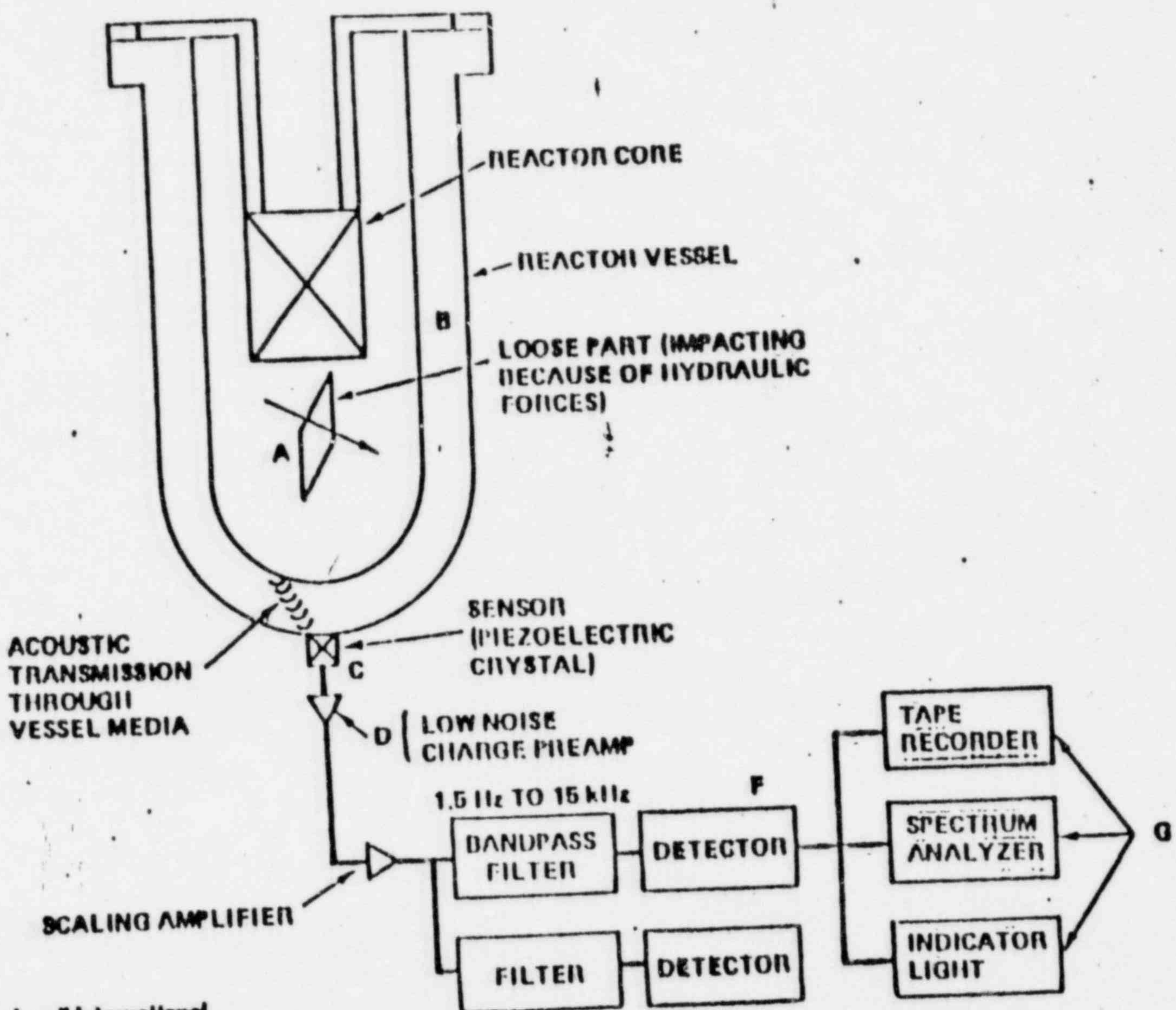
- FLOW BLOCKAGE EFFECTS

  - CONSIDERED BLOCKAGE AT FUEL, LOWER INTERNALS, STEAM GENERATOR TUBES, LOOP PIPING AND AUXILIARY SYSTEMS

  - CONSIDERED NORMAL OPERATION AND TRANSIENT CONDITIONS

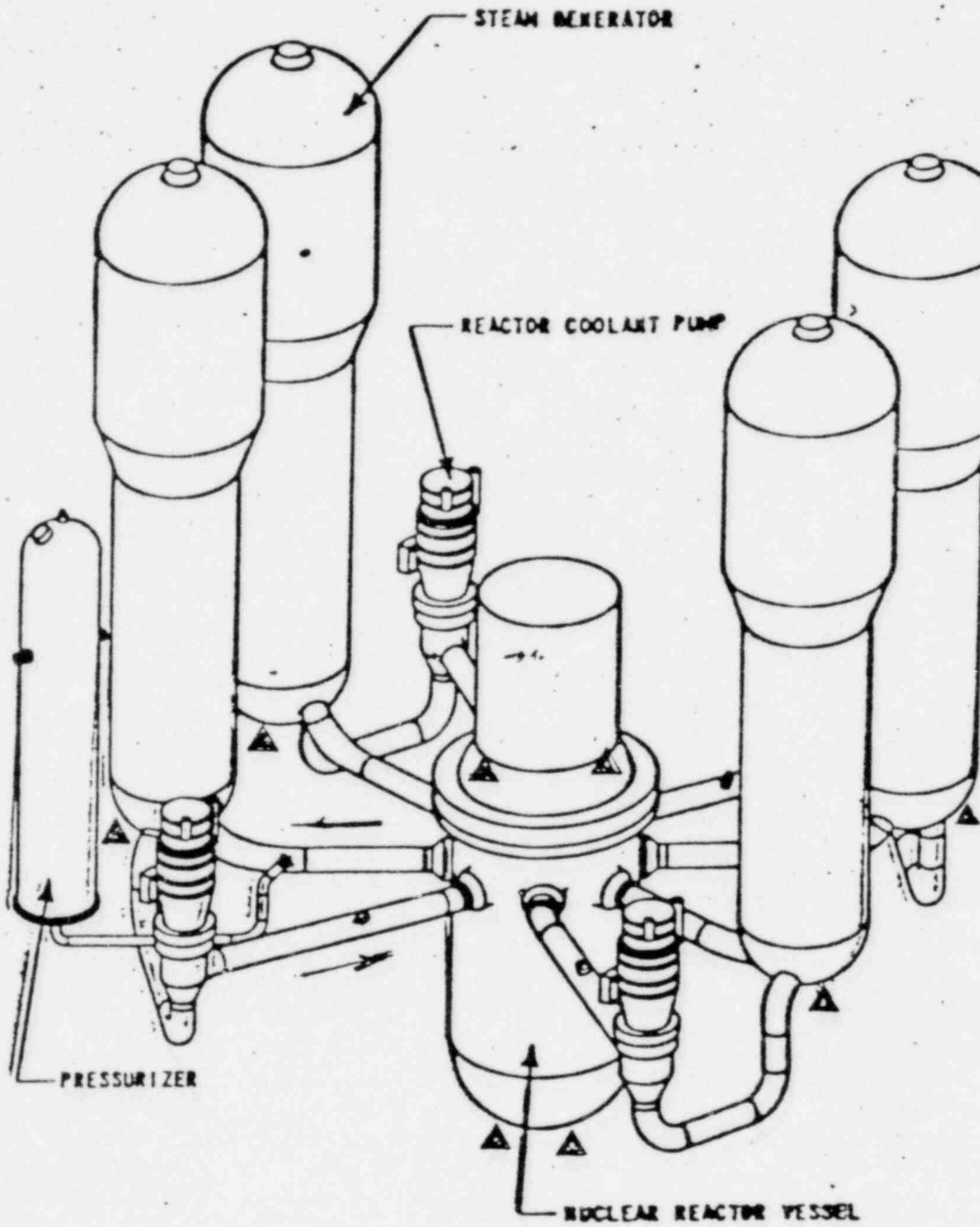
  - SIZE AND CURVED GEOMETRY OF PIECES RESULT IN NO SIGNIFICANT FLOW BLOCKAGES

V&LP MONITOR VESSEL OR STEAM GENERATOR



Rockwell International
Atomics International Division

78-S14-59-1



▲ Sensor Locations

ADDITIONAL ACTIONS

- VERIFICATION OF LPM CALIBRATION
- AUGMENTED LPM DURING RESIDUAL HEAT REMOVAL OPERATION AND REACTOR COOLANT PUMP RUNS
- INCREASE SURVELLIENCE FREQUENCIES FOR CONTROL ROD OPERABILITY INCORE FLUX MAP AND REACTOR COOLANT ACTIVITY
- SPECIAL PROCEDURE FOR LOOSE PART ALARM RESPONSE

MEETING SUMMARY DISTRIBUTION

Docket No(s): 50-369/370

JUL 2 8 1982

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bcc: Applicant & Service List