C. Cheng
copy to [illegible]

APR 0 5 1982

MEMORANDUM FOR: Brian K. Grimes, Director
Division of Emergency Preparedness

Hugh L. Thompson, Acting Director
Division of Human Factors Safety

James P. Knight, Assistant Director
for Components & Structures Engineering
Division of Engineering

William V. Johnston, Assistant Director
for Materials & Qualification
Division of Engineering

Themis P. Speis, Assistant Director
for Reactor Safety
Division of Systems Integration

FROM: G.C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

SUBJECT: REVIEW OF INFORMATION PRIOR TO GINNA RESTART

We will need your assistance in review of information provided by Rochester Gas & Electric Corporation with regard to the restart of the R.E. Ginna Nuclear Power Plant. Enclosure (1) is an outline of the Ginna Restart SER. We have indicated the areas you are requested to review and provide input for. Enclosure (2) is a 2.206 petition that has been submitted by the Sierra Club. We have agreed to address items 1, 2, 3, 4, 5, 6, 9, 13, 14, 15, and 16 in our Restart SER and to review the remaining items as to their pertinence to the Restart of the Ginna Plant. Our initial review of the remaining items (7, 8, 10, 11, and 12) are attached as Enclosure (3). We would like your input on the remaining items of the 2.206 petition to be included with the rest of your SER submittal since we must address all of the issues prior to restart.

~~PDR~~ 8204220150 XA
4pp

D/11

OFFICE ▶							
SURNAME ▶							
DATE ▶							

The licensee's submittal should be completed by April 16, 1982. Your input is requested to be submitted to the Project Manager, Jim Lyons (X24362), by April 23, 1982, to support the licensee's planned return to power, May 1, 1982. If the licensee's schedule should slip, the Project Manager will keep you informed of their projected startup date. Work performed during the review should be charged to TAC #47911.

Original signed by

G.C. Lainas, Assistant Director
for Safety Assessment
Division of Licensing

Enclosures
As stated

cc w/enclosures:
D. Eisenhut
R. Vollmer
R. Mattson
R. Bosnak
C. Cheng
V. Benaroya
D. Ziemann
T. Ippolito
B. Sheron
D. Crutchfield
J. Lyons
G. Holahan
K. Wichman

OFFICE ▶	DL:ORB#5/PM	DL:ORB#5/BC	DL:ORAB/BC	DL:AD/SA			
SURNAME ▶	JLyons:san	DCrutchfield	TIppolito	GLainas			
DATE ▶	3/31/82	4/1/82	4/1/82	4/5/82			

ENCLOSURE (1)

OUTLINE OF GINNA RESTART SER

I.	Introduction and Background	DL/ORB#5
	A. Synopsis of Event of 1/25/82	
	B. Other Staff Evaluations	
II.	Summary and Conclusions ← who?	
III.	Steam Generator Analysis	DL/ORAB Lead - Support From MTEB, CMEB, MEB
	A. Previous History of S.G. Performance	
	1. ECT Adequacy Inspection Techniques & Results	MTEB
	2. Types of Degradation Experienced	MTEB
	3. Plugging History	ORB#5
	4. Sleeving Efforts	ORB#5
	B. Cause of Failure	
	1. Ruptured Tube	MTEB
	2. Other Damaged Tubes	MTEB
	3. Source of Foreign Material	RI
	4. QA Aspects	RI
	C. ~~Extent of Damage~~ Tube Inspections following [illegible]	MTEB/CMEB
	1. S.G. Tubes	
	2. Reactor Coolant System	
	D. Inspection Results	MTEB
	1. Primary Side	
	2. Secondary Side	
	E. Metallurgical Results	MTEB
	1. RG&E/Westinghouse	
	2. NRR Independent Analysis	
	F. Repairs and Modifications	MTEB/CMEB/MEB
	G. Future Inspections and Actions	
	1. Secondary Side Video Inspections	MTEB
	2. Loose Parts Monitor	CPB
	3. Coolant Activity L.C.O.	ORB#5/RAB
	4. Review Primary to Secondary Tech Specs Leakage Limit	ORB#5/ RAB

IV.	Pressurizer Power Operated Relief Valve Performance	DL/ORB#5/MEB
	A. System Description	
	B. Failure Mechanism	
	C. Modification and Repair	
V.	Steam Generator Safety Valve Performance	MEB
	A. Description	
	B. Use and Failure	
	C. Inspection Results	
VI.	Adequacy of Accident Response	
	A. Emergency Procedures	DSI/RSB Lead - support from PTRB
	B. Instrumentation to follow the course of an accident	DHFS
	C. Emergency Preparedness	EPLB
VII.	Conclusions and Recommendations	DL/ORB#5

EO-244

April 23, 1982

MEMORANDUM FOR: R. J. Mattson, Director
Division of Systems Integration

R. H. Vollmer, Director
Division of Engineering

S. H. Hanauer, Director
Division of Safety Technology

H. L. Thompson, Acting Director
Division of Human Factors Safety

B. K. Grimes, Director
Division of Emergency Preparedness

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: INPUT FOR GINNA RESTART SER

A memo dated April 5, 1982 from G. C. Lainas set out a program for the review of information prior to the restart of Ginna. The restart SER outline proposed in that memo has been modified based on our latest understanding of the event and is enclosed here as Enclosure 1.

Since the release of the April 5, 1982 memo, the NRC Task Force that was investigating the steam generator tube rupture incident at Ginna has documented its findings in NUREG-0909. These findings were presented to the Commissioners on April 15, 1982. Enclosure 2 is a list of the Task Force findings. Enclosure 3 represents questions that the Commissioners raised during the April 15, 1982 briefing. Enclosure 4 is a list of questions from Commissioner Ahearne regarding NUREG-0909 and the questions raised by the Commissioners need to be addressed in your SER inputs. We call to your particular attention several issues flowing from the Commission meeting that must be addressed prior to restart:

a. whether 15 minutes for identifying a SGTR accident is acceptable (RSB),
b. the need for anindependent staff computer code analysis of the thermal gradients that the RV experienced (RSB), and
c. whether the licensee has adequate capability (hardware and operator training) to recognize a large SGTR event in a timely manner (RSB).

8205060626 820423
~~PDR ADOCK 05000244~~
~~PDR~~

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Copies of NUREG-0909 and Rochester Gas & Electric's evaluation of the incident are available and have been distributed. The RG&E submittal covers everything except the steam generator inspection, evaluation, and repair program which is due to be submitted April 23, 1982 and will be hand-carried to appropriate division representatives.

One copy of the transcript of the Commission briefing is being provided to each Division Director's office by separate cover.

In order to ensure that this major effort is completed in a timely manner, we have estimated the following schedule:

Licensee submittals - received
TR input to DL - May 7, 1982
Draft SER to TR Management for review - May 14, 1982
Issue SER - May 19, 1982

We will need your SER input by c.o.b. May 7, 1982.

A meeting has been scheduled for April 23, 1982 at 10:00 a.m. in Room 542A with Gus Lainas and your Assistant Directors to discuss this and related matters.

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
H. Denton
E. Case
E. Christenbury
M. Young

DISTRIBUTION
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T. Speis
L. Rubenstein
W. Johnston
J. P. Knight
R. Houston
F. Schroeder
R. Bosnak
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C. Berlinger
F. Congel
W. Gammill
L. Hulman
D. Beckham
V. Moore
D. Ziemann
K. Kniel
T. Ippolito
D. Crutchfield
J. Lyons
ORB #5 file

*SEE PREVIOUS TISSUE FOR CONCURRENCE

OFFICE	DL:ORB #5	DL: ORB #5*	DL: ORAB*	DL:AD/SA	DL:DIR		
SURNAME	JLyons:cc	DCrutchfield	TIppolito	GLainas	DEisenhut		
DATE	4/22/82			4/22/82	4/23/82		

A copy of the transcript of the Commission briefing is available in the project manager's office, Room 309. We will need your SER input by April 30, 1982.

A meeting has been scheduled for April 23, 1982 at 10:00 a.m. in Room 542A with Gus Lainas and your Assistant Directors to discuss this and related matters.

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
As stated

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J. Lyons

OFFICE	ORB#5:DL	C-ORB#5:DL	AD:SA:DL	D:DL	ORAB		
SURNAME	JLyons:dn	DCrutchfield	GLainas	DEisenhut	T. Ippolito		
DATE	4/21/82	4/21/82	4/21/82	4/ /82	4/21/82		

OUTLINE OF GINNA RESTART SER

1.0	INTRODUCTION	DL/ORB #5
2.0	NOTIFICATIONS	EPLB
3.0	SEQUENCE OF EVENTS	DL/ORAB
	3.1 Summary	
	3.2 Cooldown	
	3.3 Draindown	
4.0	OPERATOR RESPONSE	RSB/PTRB/HFEB/OLI
	4.1 Procedures	
	4.2 Evaluation	
	4.3 Conclusions	
5.0	EQUIPMENT PERFORMANCE	
	5.1 B Steam Generator Tube Failure Analyses	DL/ORAB Lead sup MTEB, CMEB, MEB
	5.2 Pressurizer Power Operated Relief Valves	RSB/MEB/ASB
	5.3 Pressurizer PORV Block Valve Performance	MEB/RSB
	5.4 B Main Steam System	MEB/RSB
	5.5 Letdown Isolation	RSB/CSB/ASB
	5.6 Effluent Monitoring System	ETSB
	5.7 Sump A Level Indicator	ICSB
	5.8 Safety Injection Pump 1C	PSB
6.0	ANALYSIS	
	6.1 Comparison of Plant Response with Previous Analysis	RSB
	6.2 Steam Void Formation	RSB
	6.3 Calculation of Leak Rate	DL/ORAB
	6.4 Thermal Transient on Reactor Coolant System	GIB/RSB/MTEB
	6.5 Hydrogen Transfer	CMEB
	6.6 Fuel Performance	CPB
	6.7 Steam Generator Overfill	MEB/RSB
	6.8 Pressurizer Power Operated Relief Valve	RSB/MEB/ASB
	6.9 Plant Water Inventory	DL/ORAB

7.0 RADIOLOGICAL ASSESSMENT AEB/RAB

7.1 Reactor Coolant System and Steam Generator

7.2 Radiological Releases

7.3 Meteorological Data

7.4 Survey Teams

7.5 Sampling (Air, Snow, Water)

7.6 TLD Measurements

7.7 Estimated Offsite Doses

7.8 Additional Radiological Information

7.9 Recommendations

8.0 Conclusions and Recommendations DL/ALL GROUPS

Note: The first branch listed has lead responsibility for the item.

NUREG-0909 ISSUES TO BE ADDRESSED

Findings		Responsible Group	Subject Area
1.4.1	1	None	
(Facility	2	RSB	Procedure/Guidelines
Response)	3	None	
	4	RSB	RCP Trip
	5	RSB	RCS Depressurizations
	6	RSB/MEB/ASB	PORV use and failure
	7	ICSB	Failure to record SG valve openings
	8	GIB/RSB/MTEB	Thermal Shock
	9	RSB	RCP - Restart
	10	RSB/CSB/ASB	PRT use and failure
	11	RSB/MEB	S/G SV use/behavior
	12	ICSB/RSB	Use of non-safety equipment
	13	ICSB/RSB/DHFS	Post accident monitoring
1.4.2	1	RSB/PTRB	Operator Actions (general)
(Human	2	RSB/PTRB	Procedure Problems/use
Factors	3	RSB/PTRB	S/G isolation in 15 min.
Considera-	4	RSB/PTRB	Trip of RCP's
tions)	5	RSB/PTRB	Steam bubble not addressed in Procedures
	6	RSB/PTRB	No subcooling in SI termination criteria
	7	RSB/PTRB	No procedure for failed S/G SV or RV
	8	RSB/PTRB	Operator response to steam bubble
	9	RSB/PTRB	Use of Aux. FW to cool S/G
	10	RSB/PTRB	Isolation of S/G RV
	11	RSB/PTRB	Auto switch over to RWST and SI Reset
	12	RSB/PTRB	Failure to terminate letdown relief
	13	RSB/PTRB	Subcooling meter problems
	14	RSB/PTRB	PORV and Block Valve controls
	15	RSB/PTRB	Location of PORV control and RCS pressure/meter
	16	RSB/PTRB/HFEB/OLB	Indicator lights burned out
	17	RSB/PTRB/HFEB/OLB	Terminology problems on control panels and in procedures
1.4.3 (Radiological Consequences)		AEB/RAB	Radiological consequences relative to design basis
1.4.4	1	DEP/IE/EPLB	Licensee's Emergency Plan
(Institu-	2	DEP/IE/EPLB	No alternate evacuation site
tional Response)	3	DEP/IE/EPLB	State and county decided not to use Prompt Notification System
	4	DEP/IE/EPLB	State was not notified of RV steam bubble
	5	None	SRI effective
	6	DEP/IE/EPLB	Lack of Region I and HQ coordination
	7	DEP/IE/EPLB	HPN adequate, ENS marginal
	8	DEP/IE/EPLB	HQ failed to make some notifications

Note: The first branch listed has lead responsibility for the item.

Findings		Responsible Group	Subject Area
1.4.5	1	AEB/RAB	Error in S/G gas analysis
(Post-	2	RSB	Slight boron dilution after event
Event	3	MTEB	Foreign objects in S/G's
Activities	4	MTEB	Tube rupture ballooning/fretting

Note: The first branch listed has lead responsibility for the item.

SER SUBJECTS

No.	Transcript Reference Pages	Subject Area	Responsible Branch
1	24 & 25	Is 15 minutes to conclude which S/G had tube rupture adequate	RSB
2	38	Ways to improve estimate of RCS leak rate during transient - rather than inferred readings	RSB, HFEB
3	46 & 47	Analyses to give better understanding of what may have happened to the reactor vessel. Including evaluation of the rate of temperture drop for various parts of the vessel for restart review.	GIB/RSB, ORAB, MTEB
4	50	Acceptability of auxiliary building ventilation system intake in an area close to the steam generator safety and relief valves	AEB/RAB
5	66	Significance of the fact that plugged tubes in periphery are always in a wedge area.	MTEB
6	75	Inspection methods to check S/G tube integrity due to corrosion process	MTEB
7	79	Acceptability of deviation from the procedures.	RSB/PRTB
8	83	Before restart a definite view on the cause of failure.	MTEB

Note: The first branch listed has lead responsibility for the item.

1. Page 1-3, last paragraph (ending at the top of page 1-4): The condensate system was contaminated because of the faulted B-generator dumping steam to the condenser earlier in the event. Since the secondary sides were tied together in the condensate system, the A-loop was also contaminated. The subsequent atmospheric dumping of A-generator steam amounts to intentional release to the atmosphere. Was it necessary? Were coolant activities checked prior to this decision to assure that no excessive radioactivity would be released to the atmosphere by this action? ORAB + RAB

2. Page 1-8, last paragraph (ending at the top of page 1-9): The report states that the tube wall thickness was less than 5% of nominal at the center of the rupture. In response to my question during the briefing, the staff response was that the >95% thickness loss was due to wear. If correct, this has two serious implications. Since this particular tube was inspected in August, 1981, one could conclude from this that either the inspection technique was not capable of detecting a substantial wall thickness loss, or the wear rate was extremely high (75% in five months). In addition, based on this one event, a thickness loss of 95% is necessary to cause tube burst. Therefore, are the present bases of the inspection program and the tube plugging criteria adequate to assure tube integrity? MTEB

3. Significant Finding on page 1-11: Whether the reactor vessel had been subjected to thermal shock and whether any damage was sustained should be pursued in the restart review. The referenced statement appears to have been with little basis. GIB/RSB/MTEB

4. What was the cause of the process computer malfunction for 16 minutes? Is the process computer qualified to the same standard as other class 1E equipment in the control room? + ICSB

5. Item #3 of Subsection 1.4.5 on page 1-23: The steam generator downcomer flow resistance plate modification was a generic item recommended by Westinghouse in 1975. The staff should develop a list of the plants that have made this modification, and examine the operating experience of peripheral tubes (including numbers of tubes plugged or shown degradation indications). Also the staff should examine whether the secondary sides of the steam generators of these plants have ever been inspected. ORAB

Generic, NA to restart

6. In relation to Q. 5, what is the staff's assessment on the effectiveness of a loose part monitoring system for the detection of any foreign objects and tube fretting inside steam generators? CPB

7. The descriptions of the B-steam generator water level in the first few minutes of the accident (Pages 3-3 and 3-7) are confusing and appear inconsistent with Figure 3-2. (A) What was the staff's estimate of the rate of water level fluctuation in the B-generator in the first five minutes? (B) Was the confusion attributable to the instrument problems, or the level actually fluctuated so fast? (A)-ORAB (B)-ICSB

8. A statement on page 3-9 states that it took 15 minutes to positively identify the existence of the problem in the B-generator. (A) Is this normal in terms of plant design or available equipment? (B) How long would it take for a B&W plant in a similar situation? RSB

9. In Table 3.8 on page 3-44, the peak leak rates for the Surry 2 and the Prairie Island 1 tube rupture accidents are different than those in NUREG-0651 (80 and 390 gpm, respectively). ORAB

10. Page 4-16, Subsection 4.2.2(1) indicates that the simulator responded in a similar manner to the course of this event, but at a slower rate (emphasis added). (A) Is the simulator not adequate? (B) Are changes needed in the simulator for future operator training? OLB

11. Subsection 4.2.2(6) on page 4-17 appears to indicate that there are ambiguities in the procedures, which allowed the operators to interpret as they saw fit. Staff should clarify. RSB/PTRB

12. Subsection 4.2.2(10) on page 4-17 indicates that the plant staff did not think they deviated from the applicable procedures. Yet the task force finding says that they did, but were prudent to deviate. Without passing judgement as to who is right and who is wrong, it is obvious that there is something wrong with the procedures. This question on the procedures should be further pursued in the restart review. RSB/PTRB

13. In several places the report references equipment failures. Will the Task Force or the restart effort examine these? For example,

- Page 1-10, No. 7 - main steam system valve position recorders failing to indicate the openings of safety valve and PORV's. ICSB

- Page 1-17, No. 13: Requiring a mental conversion which is simple in low stress situations is probably not wise. Does the staff for restart intend to address this? Note that it relies on the computer which at a later stage did break down. RSB/PTRB

- Page 1-18, No. 14: No obvious way to identify the difference in switch operation or whether the spring-loaded switch had to be held until the valve closed or opened fully. RSB/PTRB

- Page 1-18, No. 15: the control room operator is required to rely on someone else to describe the results of opening the PORV leading to a two-person bump and wait type of operation. RSB/PTRB

- Page 1-18, No. 16: Numerous indicator lights burned out. RSB/PTRB/HFEB/OLB

14. Page 1-16, No. 10: Does the staff intend to address whether the scenario in the second sentence would have led to a more serious event? RSB/PTRB

80-284

April 23, 1982

MEMORANDUM FOR: R. J. Mattson, Director
Division of Systems Integration

R. H. Vollmer, Director
Division of Engineering

S. H. Hanauer, Director
Division of Safety Technology

H. L. Thompson, Acting Director
Division of Human Factors Safety

B. K. Grimes, Director
Division of Emergency Preparedness

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: INPUT FOR GINNA RESTART SER

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a. whether 15 minutes for identifying a SGTR accident is acceptable (RSB),
b. the need for an independent staff computer code analysis of the thermal gradients that the RV experienced (RSB), and
c. whether the licensee has adequate capability (hardware and operator training) to recognize a large SGTR event in a timely manner (RSB).

~~8205060626 820423~~ XA 11pp.
~~PDR ADOCK 05000244~~

OFFICE ▶							
SURNAME ▶							
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A meeting has been scheduled for April 23, 1982 at 10:00 a.m. in Room 542A with Gus Lainas and your Assistant Directors to discuss this and related matters.

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing

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M. Young

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D. Crutchfield
J. Lyons
ORB #5 file

*SEE PREVIOUS TISSUE FOR CONCURRENCE

OFFICE	DL: ORB #5	DL: ORB #5*	DL: ORAB*	DL: AD/SA	DL: DIR		
SURNAME	JLyons:cc	DCrutchfield	TIppolito	GLainas	DEisenhut		
DATE	4/22/82			4/22/82	4/23/82		

A copy of the transcript of the Commission briefing is available in the project manager's office, Room 309. We will need your SER input by April 30, 1982.

A meeting has been scheduled for April 23, 1982 at 10:00 a.m. in Room 542A with Gus Lainas and your Assistant Directors to discuss this and related matters.

Darrell G. Eisenhut, Director
Division of Licensing

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OFFICE	ORB#5:DL	C-ORB#5:DL	AD-SA:DL	D:DL	ORAB		
SURNAME	JLyons:dn	DCrutchfield	GLainas	DEisenhut	TIppolito		
DATE	4/21/82	4/21/82	4/21/82	4/ /82	4/21/82		

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—3

OUTLINE OF GINNA RESTART SER

1.0	INTRODUCTION	DL/ORB #5
2.0	NOTIFICATIONS	EPLB
3.0	SEQUENCE OF EVENTS	DL/ORAB
	3.1 Summary	
	3.2 Cooldown	
	3.3 Draindown	
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	4.1 Procedures	
	4.2 Evaluation	
	4.3 Conclusions	
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	6.6 Fuel Performance	CPB
	6.7 Steam Generator Overfill	MEB/RSB
	6.8 Pressurizer Power Operated Relief Valve	RSB/MEB/ASB
	6.9 Plant Water Inventory	DL/ORAB

7.0 RADIOLOGICAL ASSESSMENT AEB/RAB

7.1 Reactor Coolant System and Steam Generator

7.2 Radiological Releases

7.3 Meteorological Data

7.4 Survey Teams

7.5 Sampling (Air, Snow, Water)

7.6 TLD Measurements

7.7 Estimated Offsite Doses

7.8 Additional Radiological Information

7.9 Recommendations

8.0 Conclusions and Recommendations DL/ALL GROUPS

Note: The first branch listed has lead responsibility for the item.

NUREG-0909 ISSUES TO BE ADDRESSED

Findings		Responsible Group	Subject Area
1.4.1	1	None	
(Facility	2	RSB	Procedure/Guidelines
Response)	3	None	
	4	RSB	RCP Trip
	5	RSB	RCS Depressurizations
	6	RSB/MEB/ASB	PORV use and failure
	7	ICSB	Failure to record SG valve openings
	8	GIB/RSB/MTEB	Thermal Shock
	9	RSB	RCP - Restart
	10	RSB/CSB/ASB	PRT use and failure
	11	RSB/MEB	S/G SV use/behavior
	12	ICSB/RSB	Use of non-safety equipment
	13	ICSB/RSB/DHFS	Post accident monitoring
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(Human	2	RSB/PTRB	Procedure Problems/use
Factors	3	RSB/PTRB	S/G isolation in 15 min.
Considera-	4	RSB/PTRB	Trip of RCP's
tions)	5	RSB/PTRB	Steam bubble not addressed in Procedures
	6	RSB/PTRB	No subcooling in SI termination criteria
	7	RSB/PTRB	No procedure for failed S/G SV or RV
	8	RSB/PTRB	Operator response to steam bubble
	9	RSB/PTRB	Use of Aux. FW to cool S/G
	10	RSB/PTRB	Isolation of S/G RV
	11	RSB/PTRB	Auto switch over to RWST and SI Reset
	12	RSB/PTRB	Failure to terminate letdown relief
	13	RSB/PTRB	Subcooling meter problems
	14	RSB/PTRB	PORV and Block Valve controls
	15	RSB/PTRB	Location of PORV control and RCS pressure/meter
	16	RSB/PTRB/HFEB/OLB	Indicator lights burned out
	17	RSB/PTRB/HFEB/OLB	Terminology problems on control panels and in procedures
1.4.3 (Radiological Consequences)		AEB/RAB	Radiological consequences relative to design basis
1.4.4	1	DEP/IE/EPLB	Licensee's Emergency Plan
(Institu-	2	DEP/IE/EPLB	No alternate evacuation site
tional	3	DEP/IE/EPLB	State and county decided not to use Prompt Notification System
Response)	4	DEP/IE/EPLB	State was not notified of RV steam bubble
	5	None	SRI effective
	6	DEP/IE/EPLB	Lack of Region I and HQ coordination
	7	DEP/IE/EPLB	HPN adequate, ENS marginal
	8	DEP/IE/EPLB	HQ failed to make some notifications

Note: The first branch listed has lead responsibility for the item.

Findings		Responsible Group	Subject Area
1.4.5	1	AEB/RAB	Error in S/G gas analysis
(Post-	2	RSB	Slight boron dilution after event
Event	3	MTEB	Foreign objects in S/G's
Activities	4	MTEB	Tube rupture ballooning/fretting

Note: The first branch listed has lead responsibility for the item.

SER SUBJECTS

No.	Transcript Reference Pages	Subject Area	Responsible Branch
1	24 & 25	Is 15 minutes to conclude which S/G had tube rupture adequate	RSB
2	38	Ways to improve estimate of RCS leak rate during transient - rather than inferred readings	RSB, HFEB
3	46 & 47	Analyses to give better understanding of what may have happened to the reactor vessel. Including evaluation of the rate of temperture drop for various parts of the vessel for restart review.	GIB/RSB, ORAB, MTEB
4	50	Acceptability of auxiliary building ventilation system intake in an area close to the steam generator safety and relief valves	AEB/RAB
5	66	Significance of the fact that plugged tubes in periphery are always in a wedge area.	MTEB
6	75	Inspection methods to check S/G tube integrity due to corrosion process	MTEB
7	79	Acceptability of deviation from the procedures.	RSB/PRTB
8	83	Before restart a definite view on the cause of failure.	MTEB

Note: The first branch listed has lead responsibility for the item.

1. Page 1-3, last paragraph (ending at the top of page 1-4): The condensate system was contaminated because of the faulted B-generator dumping steam to the condenser earlier in the event. Since the secondary sides were tied together in the condensate system, the A-loop was also contaminated. The subsequent atmospheric dumping of A-generator steam amounts to intentional release to the atmosphere. Was it necessary? Were coolant activities checked prior to this decision to assure that no excessive radioactivity would be released to the atmosphere by this action? ORAB / RAB

2. Page 1-8, last paragraph (ending at the top of page 1-9): The report states that the tube wall thickness was less than 5% of nominal at the center of the rupture. In response to my question during the briefing, the staff response was that the >95% thickness loss was due to wear. If correct, this has two serious implications. Since this particular tube was inspected in August, 1981, one could conclude from this that either the inspection technique was not capable of detecting a substantial wall thickness loss, or the wear rate was extremely high (75% in five months). In addition, based on this one event, a thickness loss of 95% is necessary to cause tube burst. Therefore, are the present bases of the inspection program and the tube plugging criteria adequate to assure tube integrity? MTEB

3. Significant Finding on page 1-11: Whether the reactor vessel had been subjected to thermal shock and whether any damage was sustained should be pursued in the restart review. The referenced statement appears to have been with little basis. GIB/RSB/MT

4. What was the cause of the process computer malfunction for 16 minutes? Is the process computer qualified to the same standard as other class 1E equipment in the control room? ICSB

5. Item #3 of Subsection 1.4.5 on page 1-23: The steam generator downcomer flow resistance plate modification was a generic item recommended by Westinghouse in 1975. The staff should develop a list of the plants that have made this modification, and examine the operating experience of peripheral tubes (including numbers of tubes plugged or shown degradation indications). Also the staff should examine whether the secondary sides of the steam generators of these plants have ever been inspected. ORAB

Generic, NA to restart SG

6. In relation to Q. 5, what is the staff's assessment on the effectiveness of a loose part monitoring system for the detection of any foreign objects and tube fretting inside steam generators? CPB

7. The descriptions of the B-steam generator water level in the first few minutes of the accident (Pages 3-3 and 3-7) are confusing and appear inconsistent with Figure 3-2. (A) What was the staff's estimate of the rate of water level fluctuation in the B-generator in the first five minutes? (B) Was the confusion attributable to the instrument problems, or the level actually fluctuated so fast? (A)·ORAB (B)·ICSB

8. A statement on page 3-9 states that it took 15 minutes to positively identify the existence of the problem in the B-generator. (A) Is this normal in terms of plant design or available equipment? (B) How long would it take for a B&W plant in a similar situation? RSB

9. In Table 3.8 on page 3-44, the peak leak rates for the Surry 2 and the Prairie Island 1 tube rupture accidents are different than those in NUREG-0651 (80 and 390 gpm, respectively). ORAB

10. Page 4-16, Subsection 4.2.2(1) indicates that the simulator responded in a similar manner to the course of this event, but at a <u>slower rate</u> (emphasis added). (A) Is the simulator not adequate? (B) Are changes needed in the simulator for future operator training? OLB

11. Subsection 4.2.2(6) on page 4-17 appears to indicate that there are ambiguities in the procedures, which allowed the operators to interpret as they saw fit. Staff should clarify. RSB/PTRB

12. Subsection 4.2.2(10) on page 4-17 indicates that the plant staff did not think they deviated from the applicable procedures. Yet the task force finding says that they did, but were prudent to deviate. Without passing judgement as to who is right and who is wrong, it is obvious that there is something wrong with the procedures. This question on the procedures should be further pursued in the restart review. RSB/PTRB

13. In several places the report references equipment failures. Will the Task Force or the restart effort examine these? For example,

- Page 1-10, No. 7 - main steam system valve position recorders failing to indicate the openings of safety valve and PORV's. ICSB

- Page 1-17, No. 13: Requiring a mental conversion which is simple in low stress situations is probably not wise. Does the staff for restart intend to address this? Note that it relies on the computer which at a later stage did break down. RSB/PTRB

- Page 1-18, No. 14: No obvious way to identify the difference in switch operation or whether the spring-loaded switch had to be held until the valve closed or opened fully. RSB/PTRB

- Page 1-18, No. 15: the control room operator is required to rely on someone else to describe the results of opening the PORV leading to a two-person bump and wait type of operation. RSB/PTRB

- Page 1-18, No. 16: Numerous indicator lights burned out. RSB/PTRB/HFEB/OLB

14. Page 1-16, No. 10: Does the staff intend to address whether the scenario in the second sentence would have led to a more serious event? RSB/PTRB

Louis Frank

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

D/B

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

AGENDA

o	INTRODUCTION	R. C. Mecredy
o	INSPECTION RESULTS	A. E. Curtis
o	FAILURE ANALYSIS PROGRAM	L. F. Ermold
o	B-STEAM GENERATOR REPAIRS	J. C. Noon
o	TECHNICAL BASIS FOR REPAIRS	L. F. Ermold
o	SUMMARY	J. C. Hutton

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

OBJECTIVES

- DETERMINE THE FULL EXTENT OF DEFECTS
- DETERMINE THE TUBE FAILURE MECHANISM(S)
- RESTORE THE STEAM GENERATOR TO A CONDITION WHICH IS SAFE TO OPERATE WHILL MAINTAINING RADIATION EXPOSURES AS LOW AS REASONABLY ACHIEVABLE
- OBTAIN NRC CONCURRENCE FOR RETURN TO POWER

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

PURPOSE OF MEETING

- TO REVIEW STEAM GENERATOR EVALUATION REPORT DATED APRIL 26, 1982
- SUMMARIZE RESULTS OF TUBE FAILURE ANALYSIS PROGRAM
- PRESENT CONCLUSIONS AS TO REASONS FOR JANUARY 25, 1982 TUBE RUPTURE
- IDENTIFY ANY ADDITIONAL INFORMATION REQUIRED FOR TIMELY COMPLETION OF SER

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NSARB/NRC REVIEWS

- CONCURRENCE WITH PROGRAM CONCEPTS

  NSARB - 2/26

  NRC - 3/1

- APPROVAL OF REMOVAL OF METALLURGICAL SAMPLES

  NSARB - 2/26

  NRC - 3/1

- APPROVAL OF REPAIR PROGRAM

  NSARB - 3/16

  NRC - 3/23

- APPROVAL OF RETURN TO POWER

  NSARB - 5/10

  NRC - 5/19

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

STEAM GENERATOR CONFIGURATION AND OPERATING HISTORY

- CONFIGURATION
- OPERATING HISTORY
- CHEMISTRY
- PLUGGING
- SECONDARY SIDE MODIFICATIONS

GINNA STATION
STEAM GENERATOR EVALUATION
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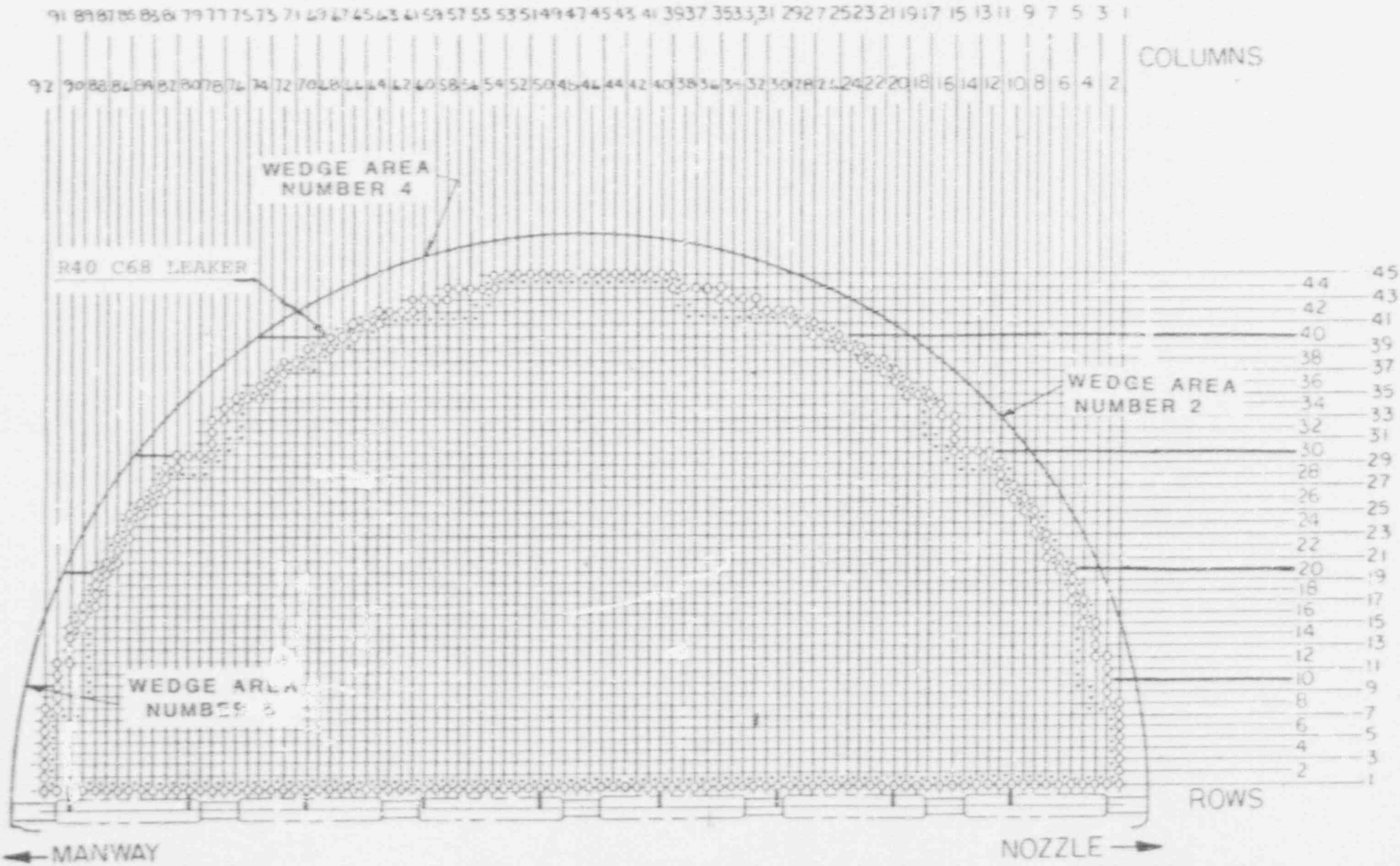
APRIL 30, 1982

JANUARY 1976
PERIPHERY AREA DEFECTS

B-HOT LEG (INLET)
RG&E Steam Generator

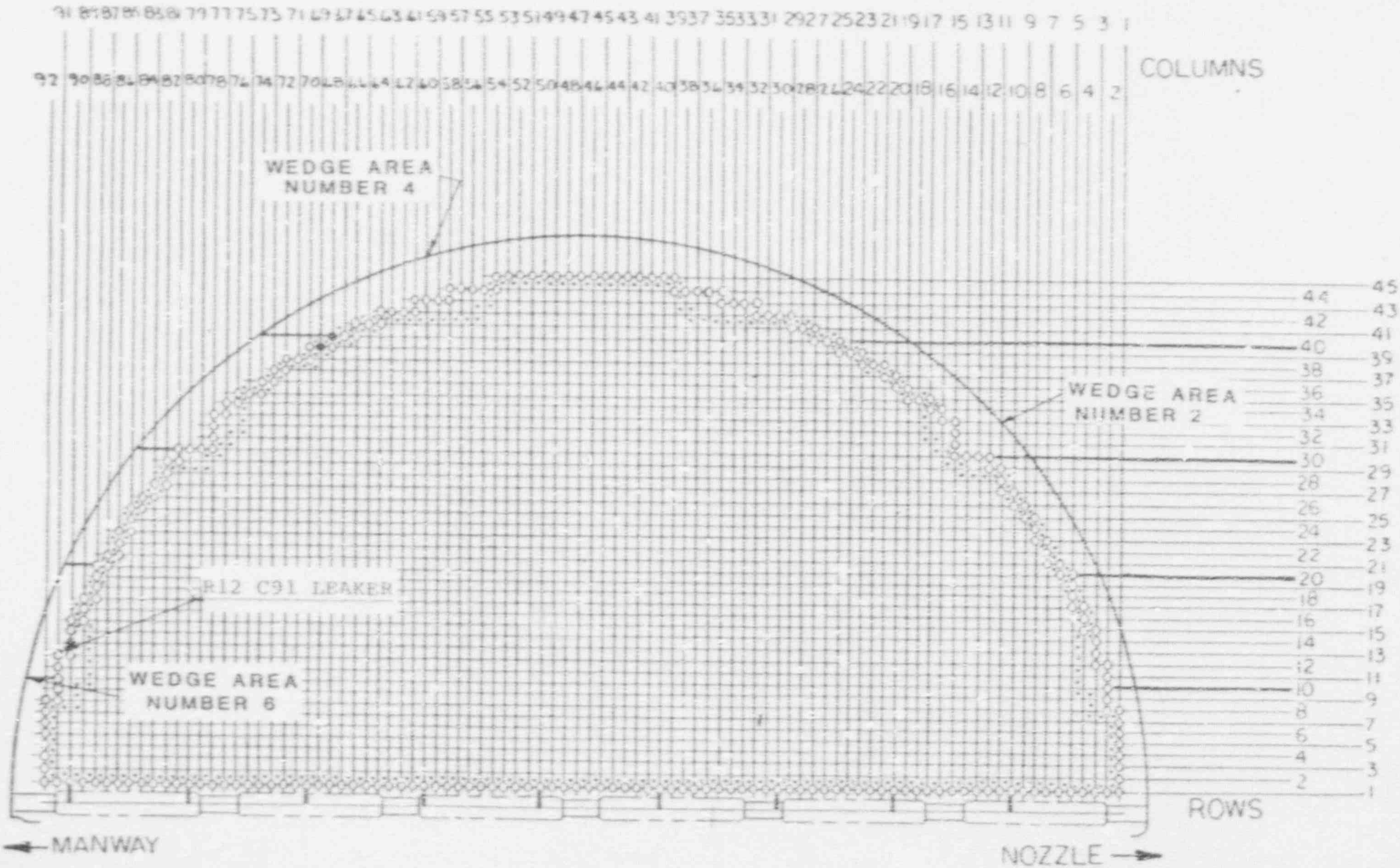


Sheet 1 of 10

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

MAY 1976
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)

RG&E Steam Generator



Sheet 2 of 10

GINNA STATION
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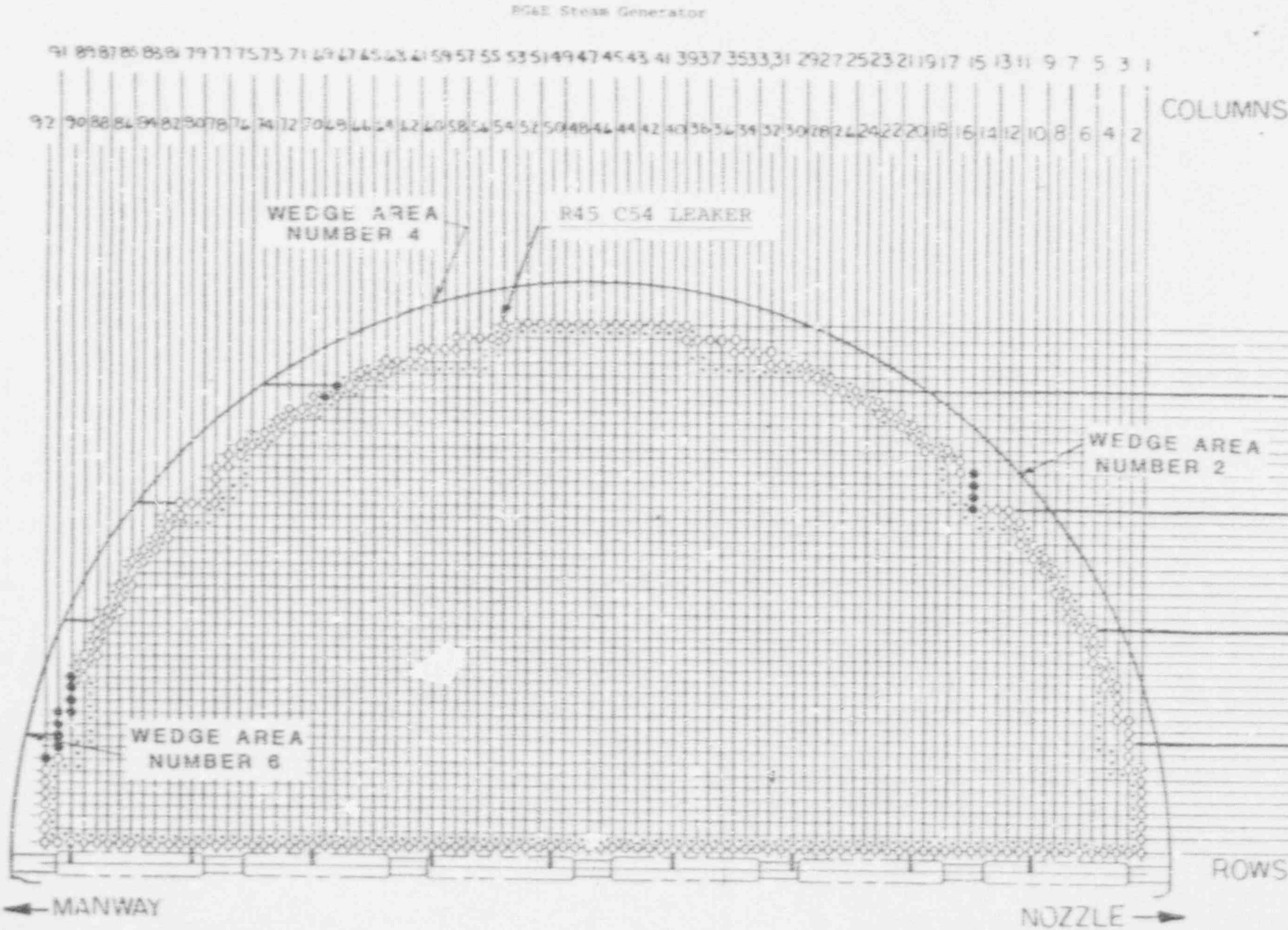
APRIL 30, 1982

JULY 1977
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



Sheet 3 of 10

GINNA STATION
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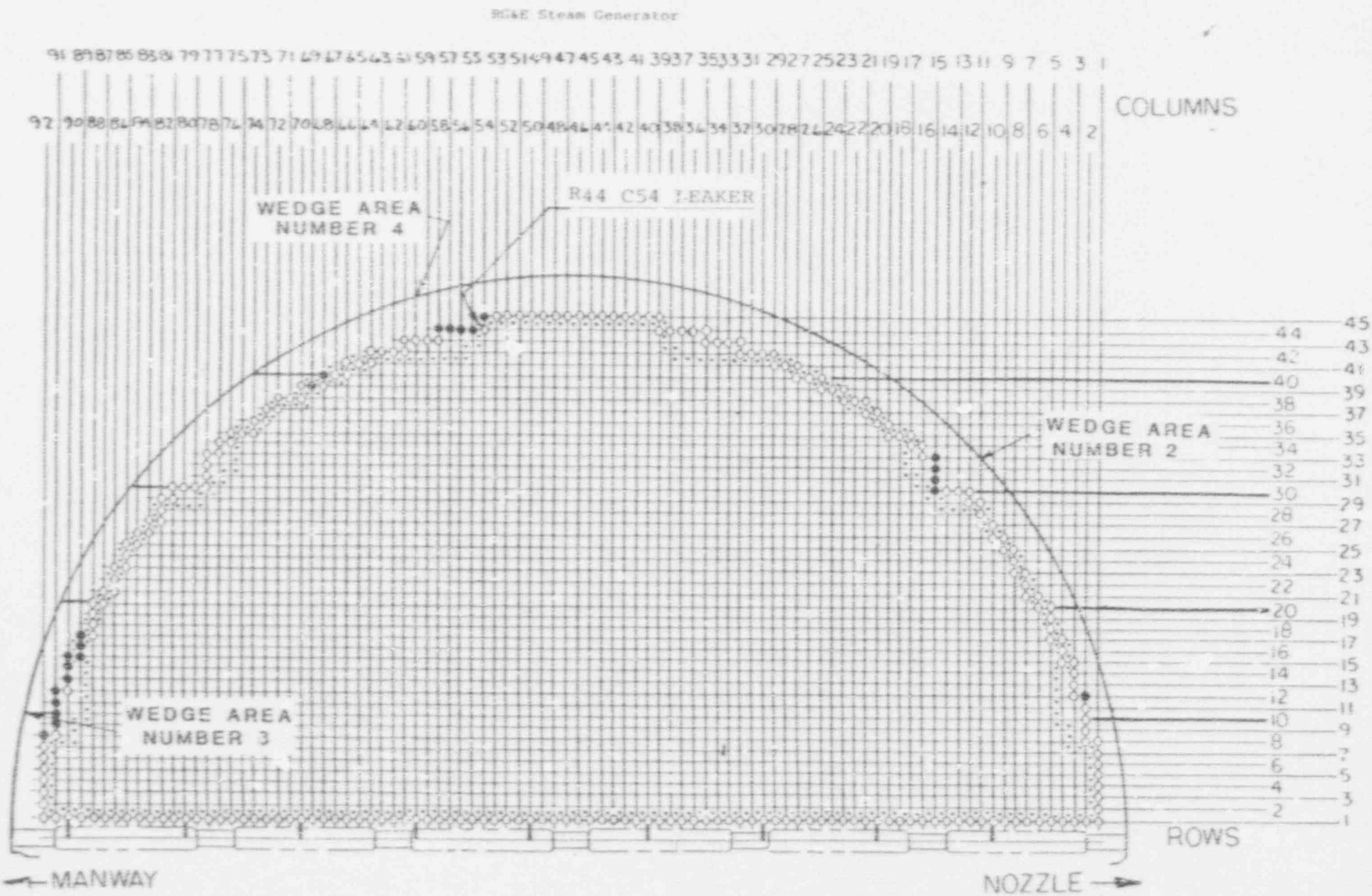
NRC MEETING
APRIL 30, 1982

JANUARY 1978
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



Sheet 4 of 10

GINNA STATION
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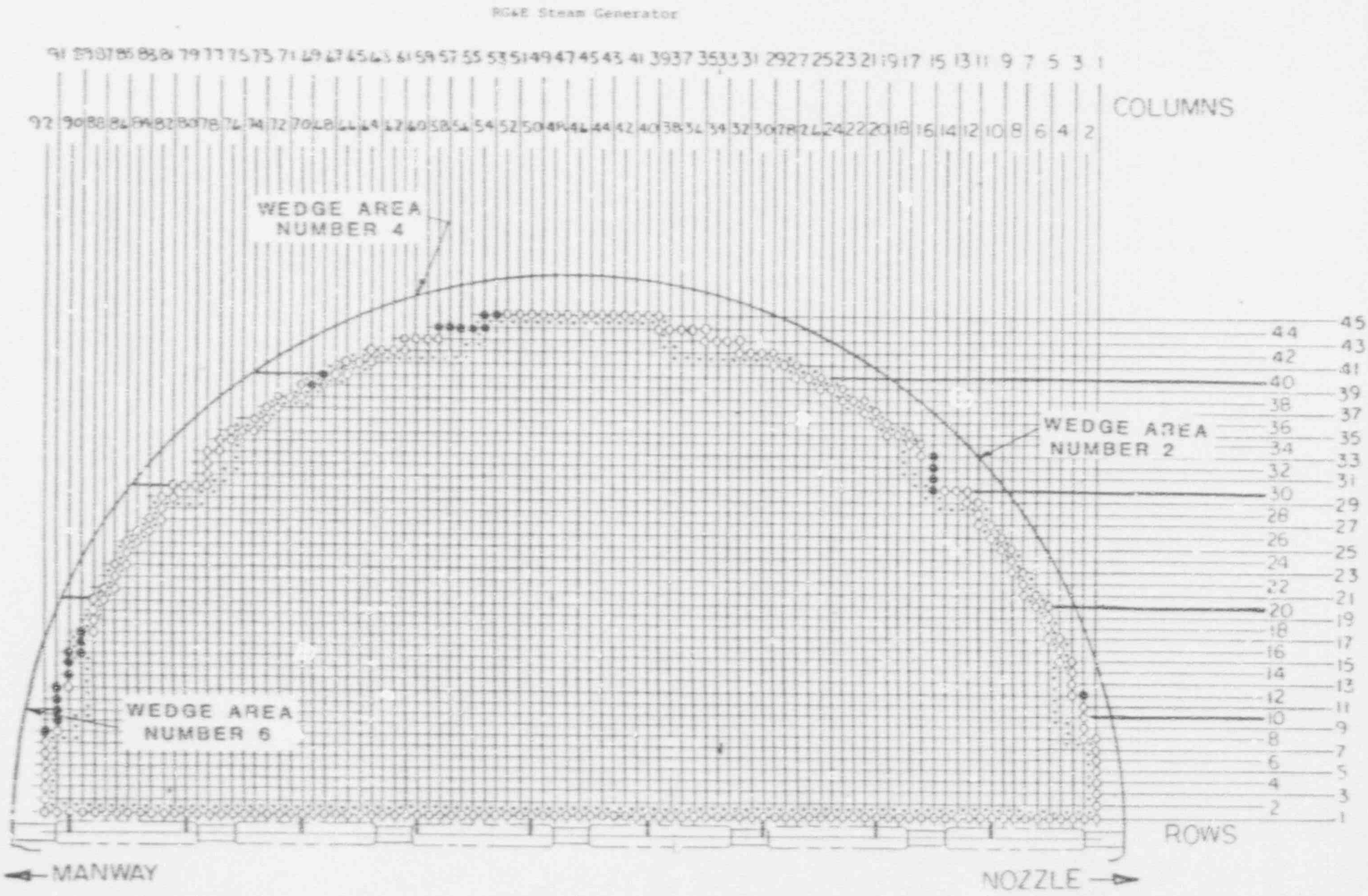
NRC MEETING
APRIL 30, 1982

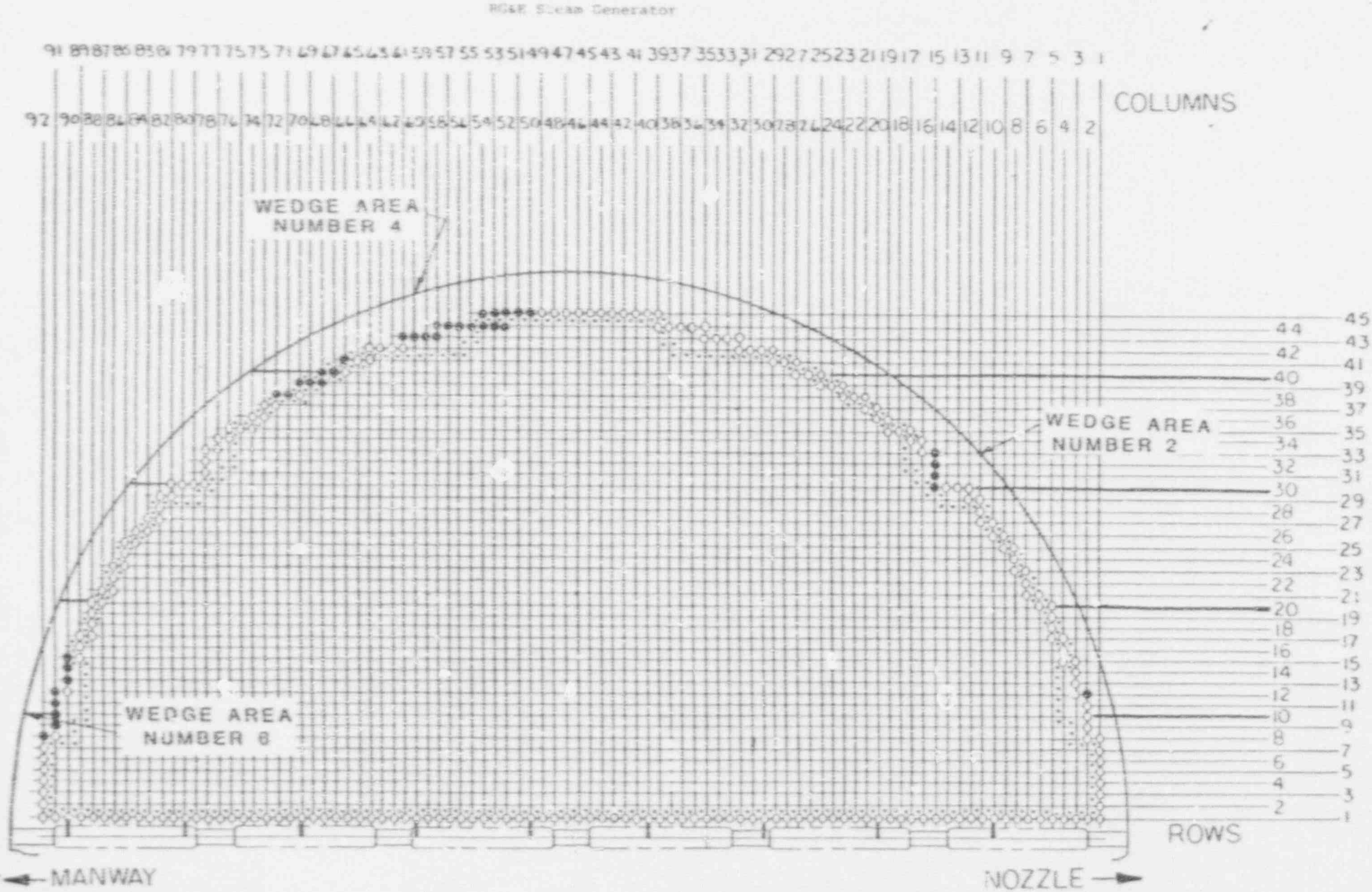
APRIL 1978
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



Sheet 5 of 10

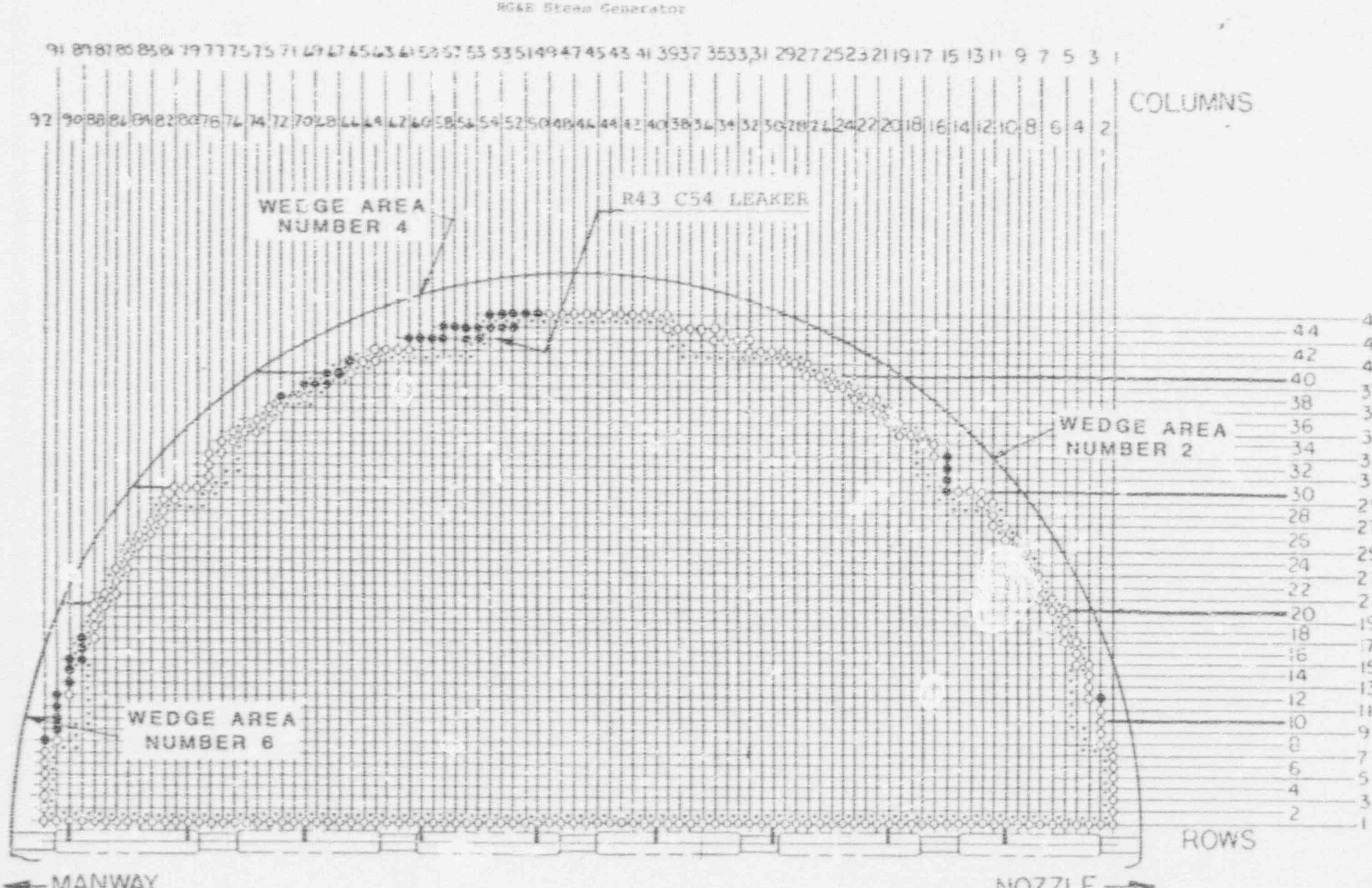
GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

FEBRUARY 1979
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

DECEMBER 1979
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



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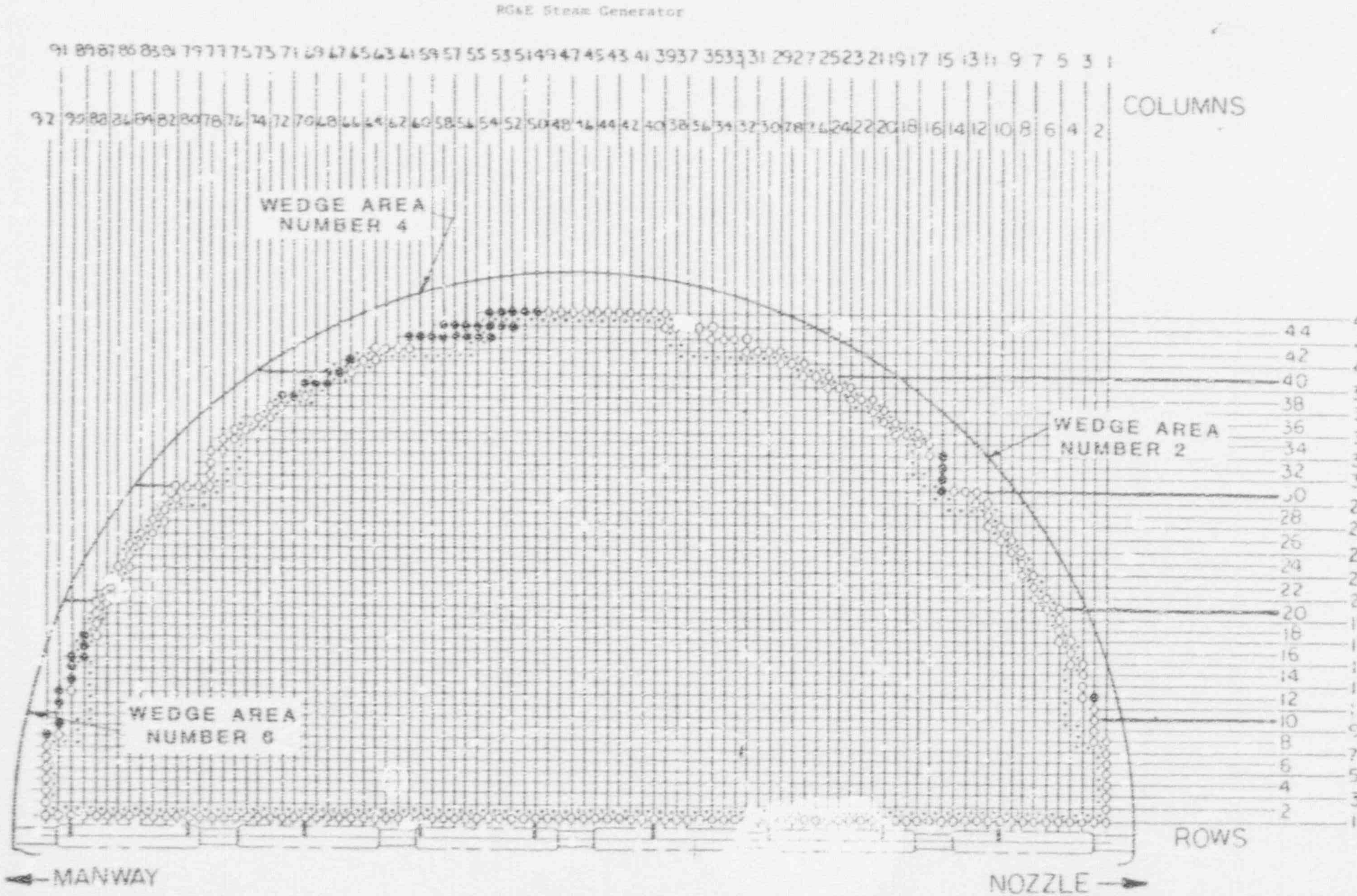
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

APRIL 1980
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



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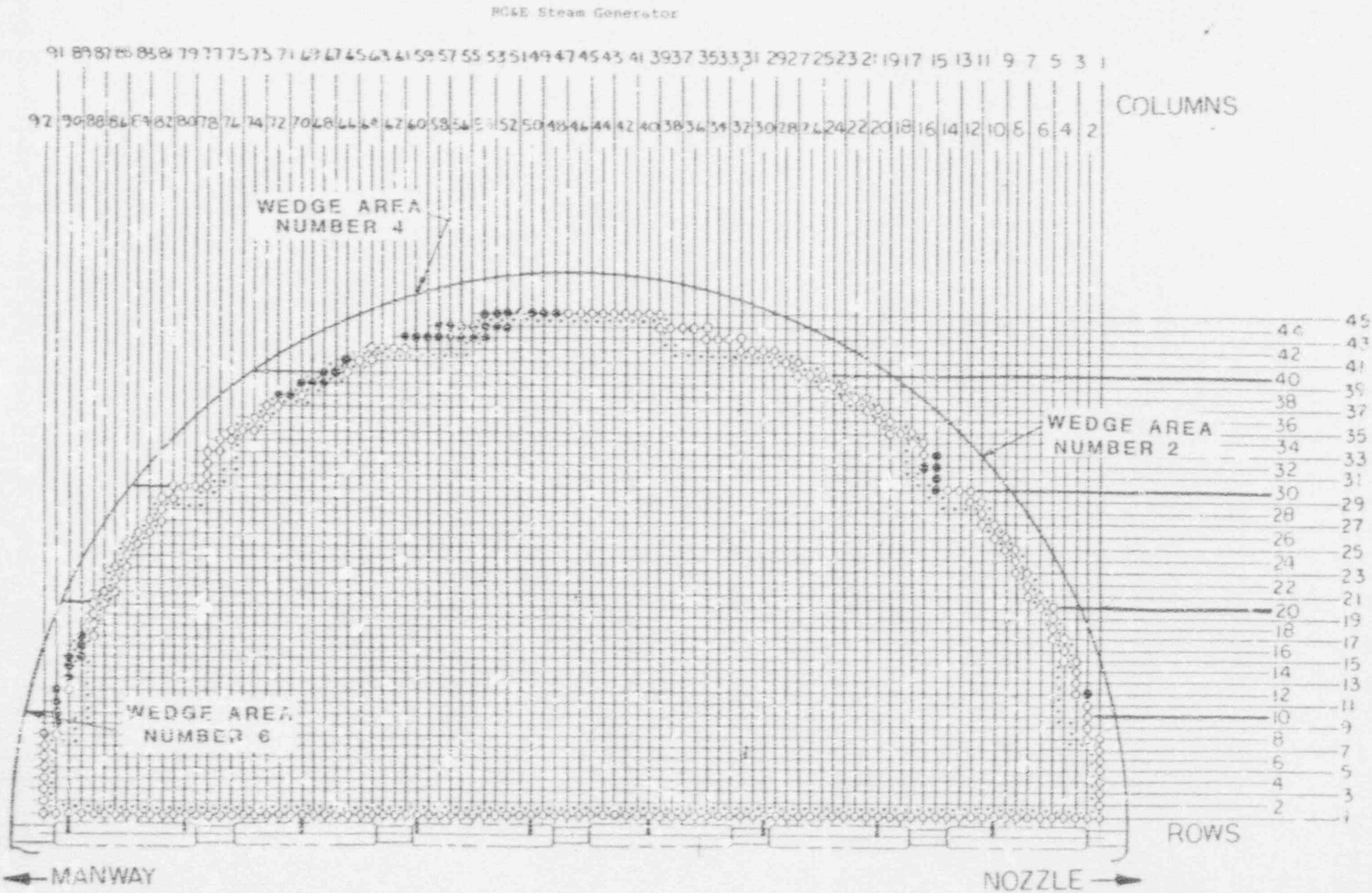
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

APRIL 1981
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)



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STEAM GENERATOR EVALUATION
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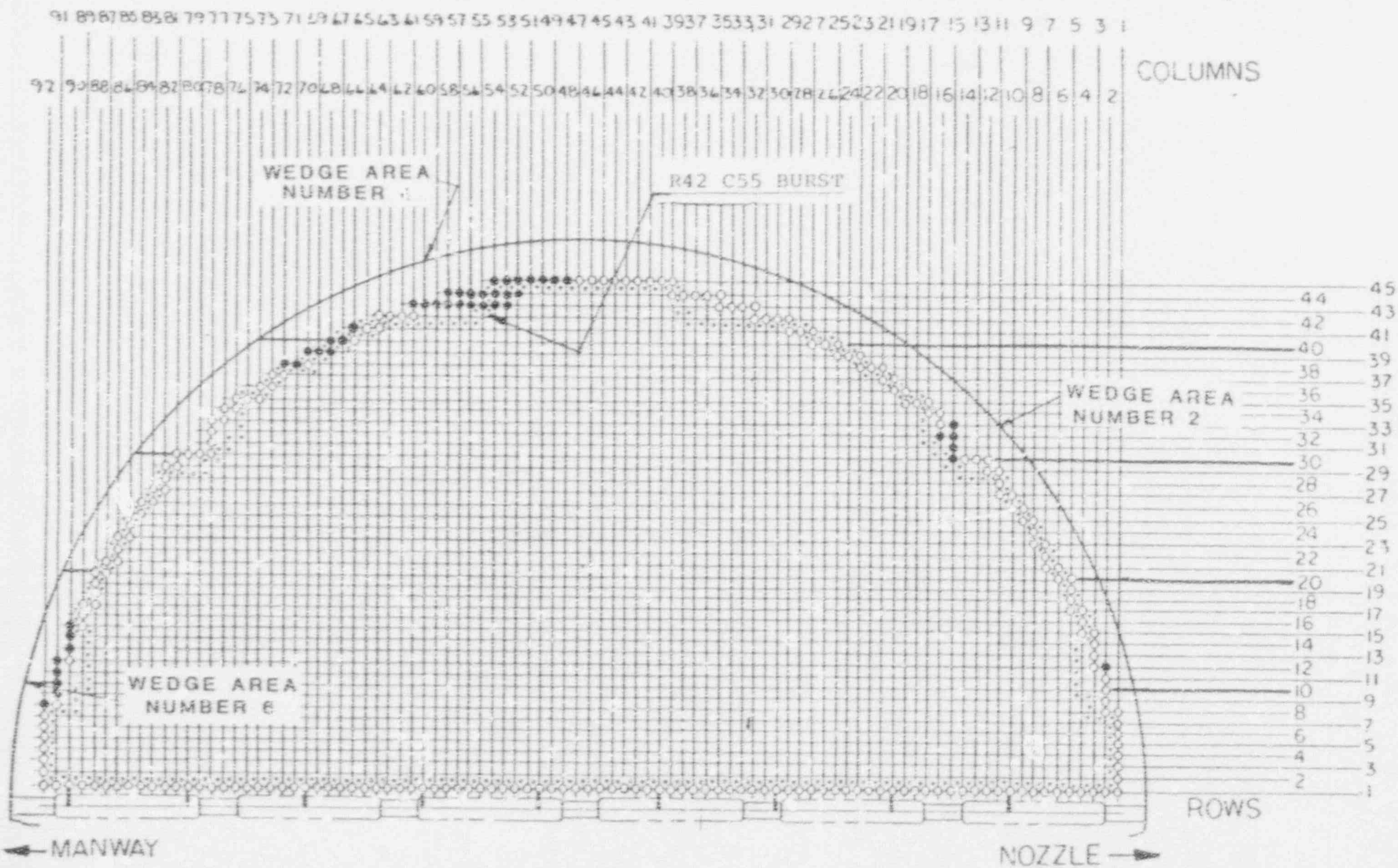
APRIL 30, 1982

JANUARY 25, 1982
PERIPHERY AREA DEFECTS
B-HOT LEG (INLET)

RG&E Steam Generator



GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

STEAM GENERATOR INSPECTION RESULTS

- EDDY CURRENT EXAMINATION Multi - 1978
- PROFILOMETRY EXAMINATION
  Geometric conditions
  50-100 tubes
- FIBER OPTICS INSPECTION
- TELEVISION VIDEO INSPECTION
- FOREIGN OBJECTS

absolute
conductivity
changes
IGA

400 KH - 200 KHz
mixing

"B" STEAM GENERATOR

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

CATEGORIZATION OF DEFECTS

CATEGORY	NO. 6 WEDGE AREA	R40C70 AREA	NO. 4 WEDGE AREA	NO. 2 WEDGE AREA
1. Structurally Degraded	R8C92 R10C91 R11C91 R12C91 R14C90 R15C90		R42C55M R43C59 R44C55M R43C53 R43C60 R44C56 R43C54M R43C61 R44C57 R43C55M R44C52 R44C58 R43C56M R44C53M R45C53 R43C57 R44C54 R45C54 R43C58	
2. Video OD Indication	R9C91 R13C90 R16C89 R17C89	R38C71 R38C72 R39C68 R39C69 R39C70	R45C51	
3. Eddy Current Signal	R15C89	R35C75* R40C67 R40C68 R41C66	R45C46 R45C47M R45C48 R45C49 R45C50	R12C2 R28C12* R30C15 R31C15 R32C15 R32C16 R33C15
4. Preventatively Plugged			R41C55 R42C54 R42C56	
TOTALS	11	9	28	7

M Metallurgical Samples

* plugged for cold leg, support plate intersection, O.D. indications

R45C52 pulled April 1978

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

METALLURGICAL EXAMINATION AND FAILURE ANALYSIS

- SITE PHOTOGRAPHY
- WESTINGHOUSE/BATTELLE COLUMBUS LABORATORIES
- MODEL FOR WEAR ORIENTATION COMPARISONS
- PHOTOGRAPHY AT 90° INCREMENTS
- RADIOGRAPHY AT 45° INCREMENTS
- DIMENSIONING
- OPTICAL METALLOGRAPHY OF TRANSVERSE SECTIONS
- SCANNING ELECTRON MICROSCOPE FRACTOGRAPHY
- KNOOP MICROHARDNESS DETERMINATION
- ELECTRODE DISCHARGE SPECTROSCOPY - CHEMICAL COMPOSITION
- PHYSICAL PROPERTY DETERMINATIONS

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

METALLURGICAL EXAMINATION
AND
FAILURE ANALYSIS CONCLUSIONS

- MICROSTRUCTURES WERE NORMAL
- MECHANICAL PROPERTIES WERE NORMAL
- NO EVIDENCE OF CORROSION OF ANY TUBE
- INITIAL METALLURGICAL SAMPLES DISPLAYED WEAR ZONES OF O.D. WALL REDUCTION WITH CIRCUMFERENTIAL STRIATIONS
- 38 OF THE 40 WEAR ZONES SHOWED KNOOP MICROHARDNESS VALUES INDICATIVE OF COLD WORK FROM THE WEAR PROCESS
- THE "BURST" R42 C55 OCCURRED WHERE THE WEAR ZONE HAD REDUCED THE WALL THICKNESS TO < 0.008 INCHES FOR APPROXIMATELY 4 INCHES IN LENGTH
- THE "BURST" R42 C55 FRACTURE FACE WAS PURE SHEAR (DIMPLED RUPTURE), THE NORMAL FRACTURE MODE FOR TENSILE OVERLOAD
- COLLAPSED AREAS OF PERIPHERY TUBES SHOWED EXTENSIVE COLD WORK 8-10 MILS IN DEPTH
- AREAS OF SEVER AND AXIAL BREAKAGE SHOWED FATIGUE STRIATIONS
- THE DEFECT 24" ABOVE TUBE SHEET ON R45 C47 WAS CAUSED BY RUBBING FROM SEVERED TUBE R45 C54 WHICH WAS WEDGED BETWEEN WRAPPER AND OUTER ROW OF TUBES

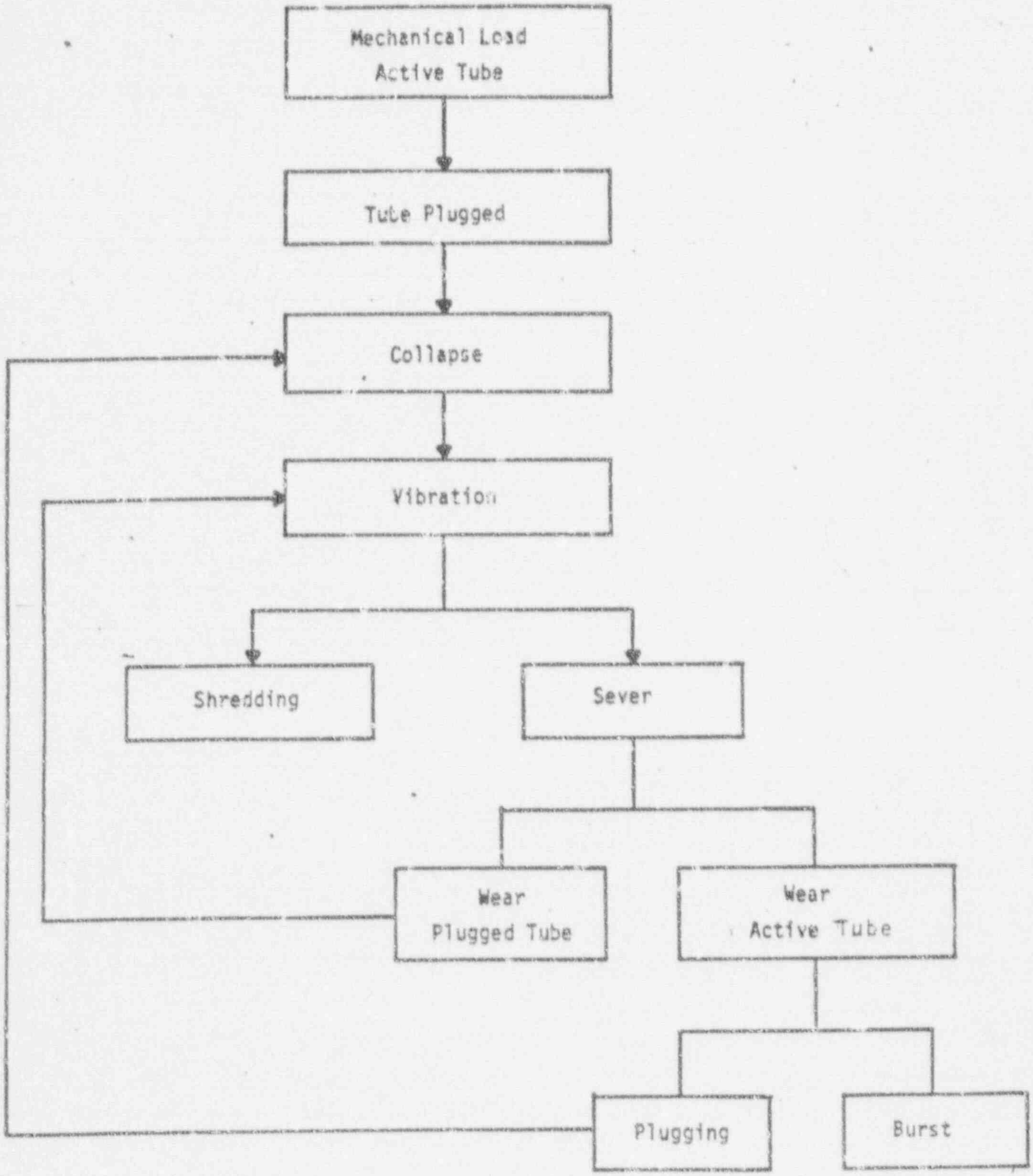
GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

POSTULATED FAILURE MECHANISM

- O INITIAL PLUGGING
- O COLLAPSE (Impact, change geometry ovality, external pressure leads to collapse
- O SEVERANCE
- O WEAR

continued loading leads to tube failure by fatigue

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982



Postulated Failure Mechanism Sequence

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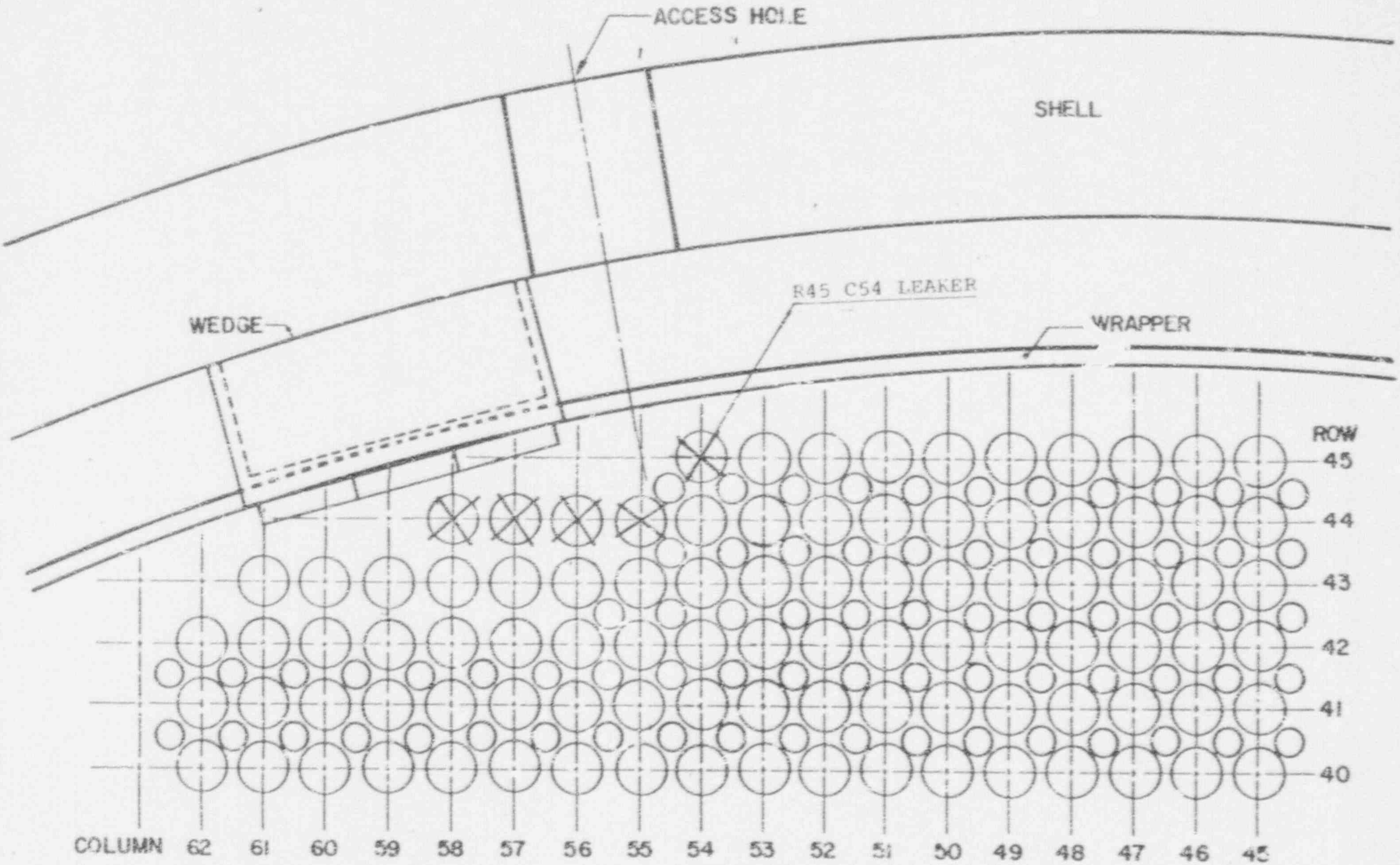
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
JULY 1977



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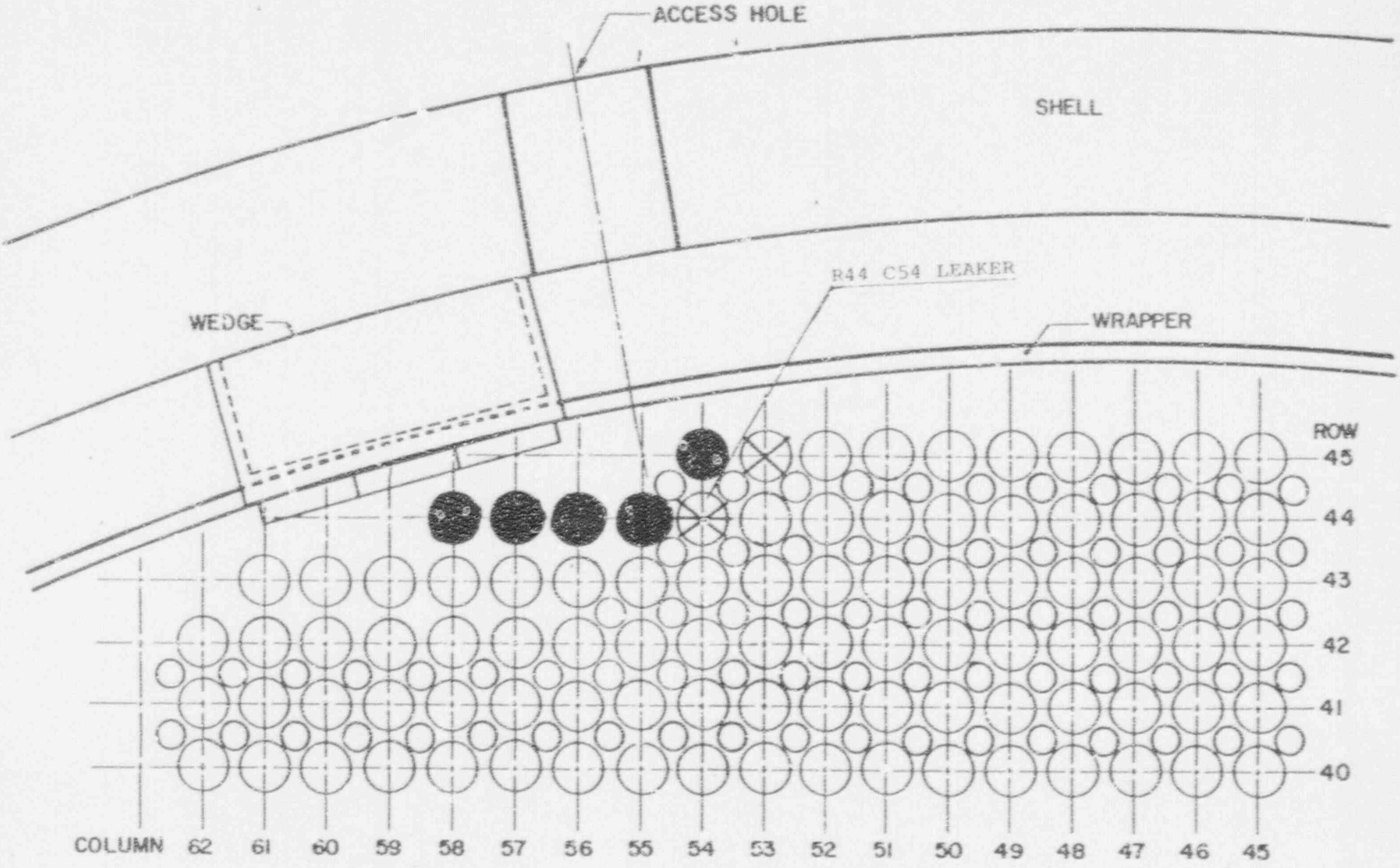
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
JANUARY 1978



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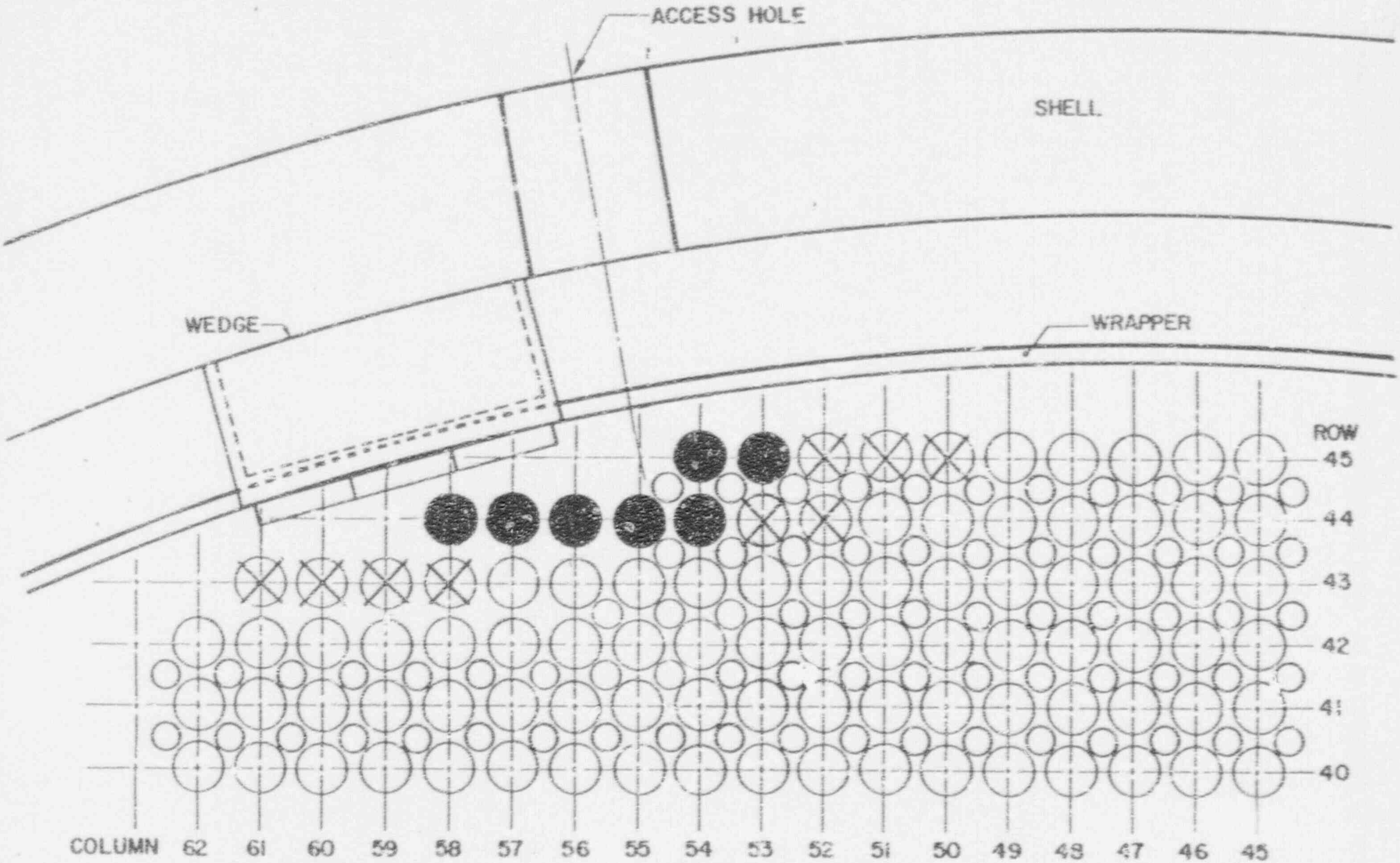
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
APRIL 1978



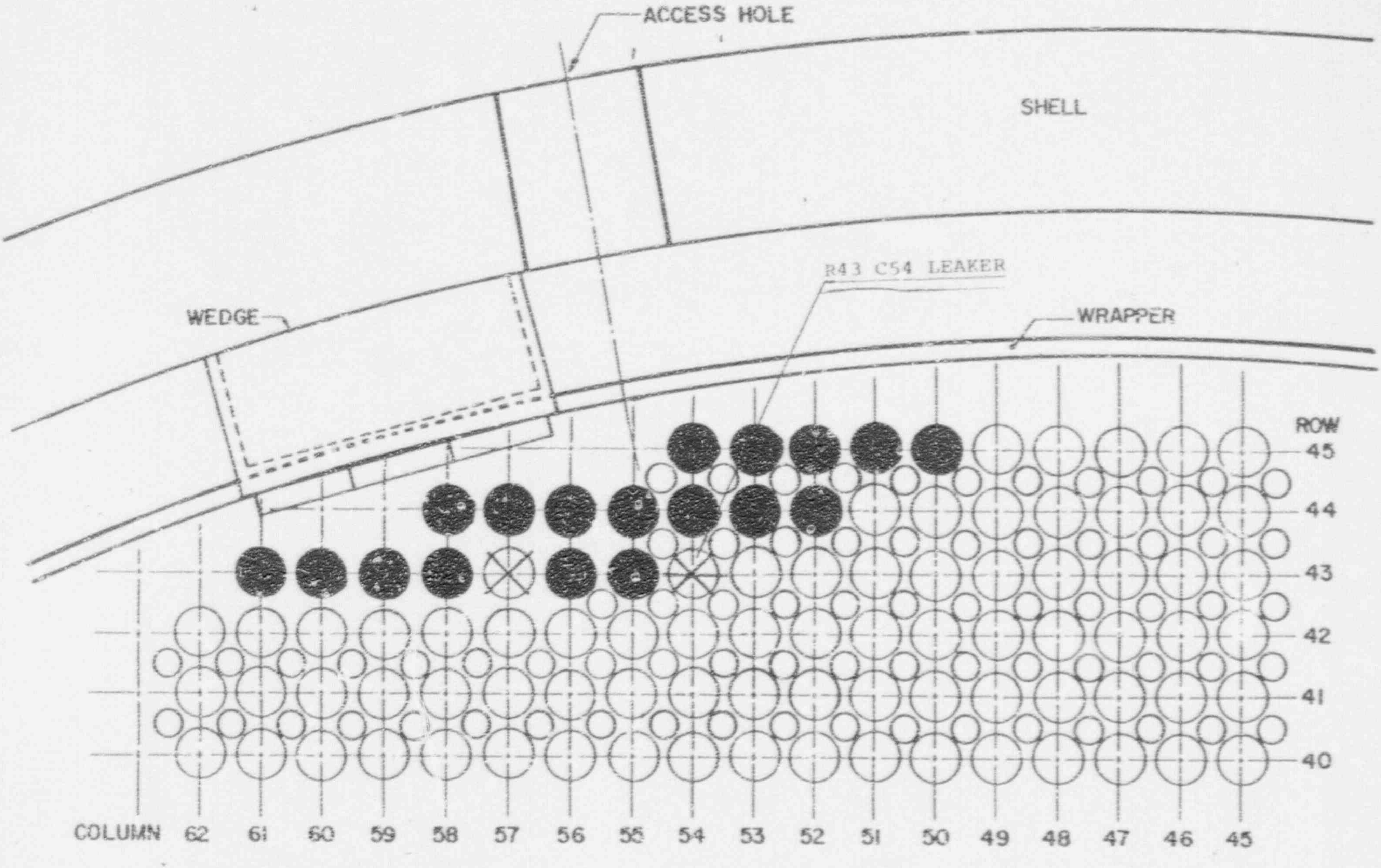
GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL ?, 1982

NO. 4 WEDGE AREA
FEBRUARY 1979

Sheet 4 of 8

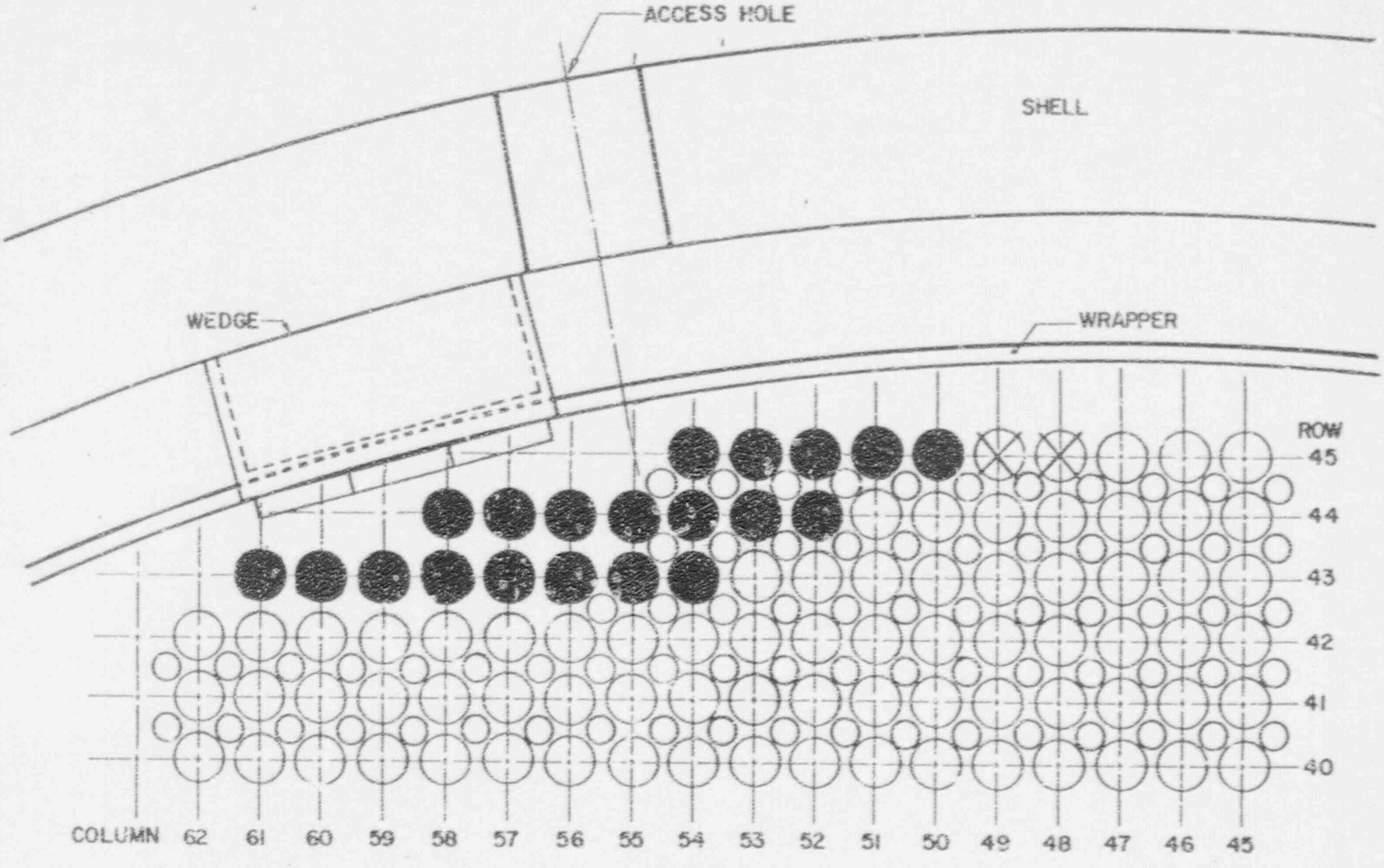
GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
DECEMBER 1979



GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
APRIL 1980



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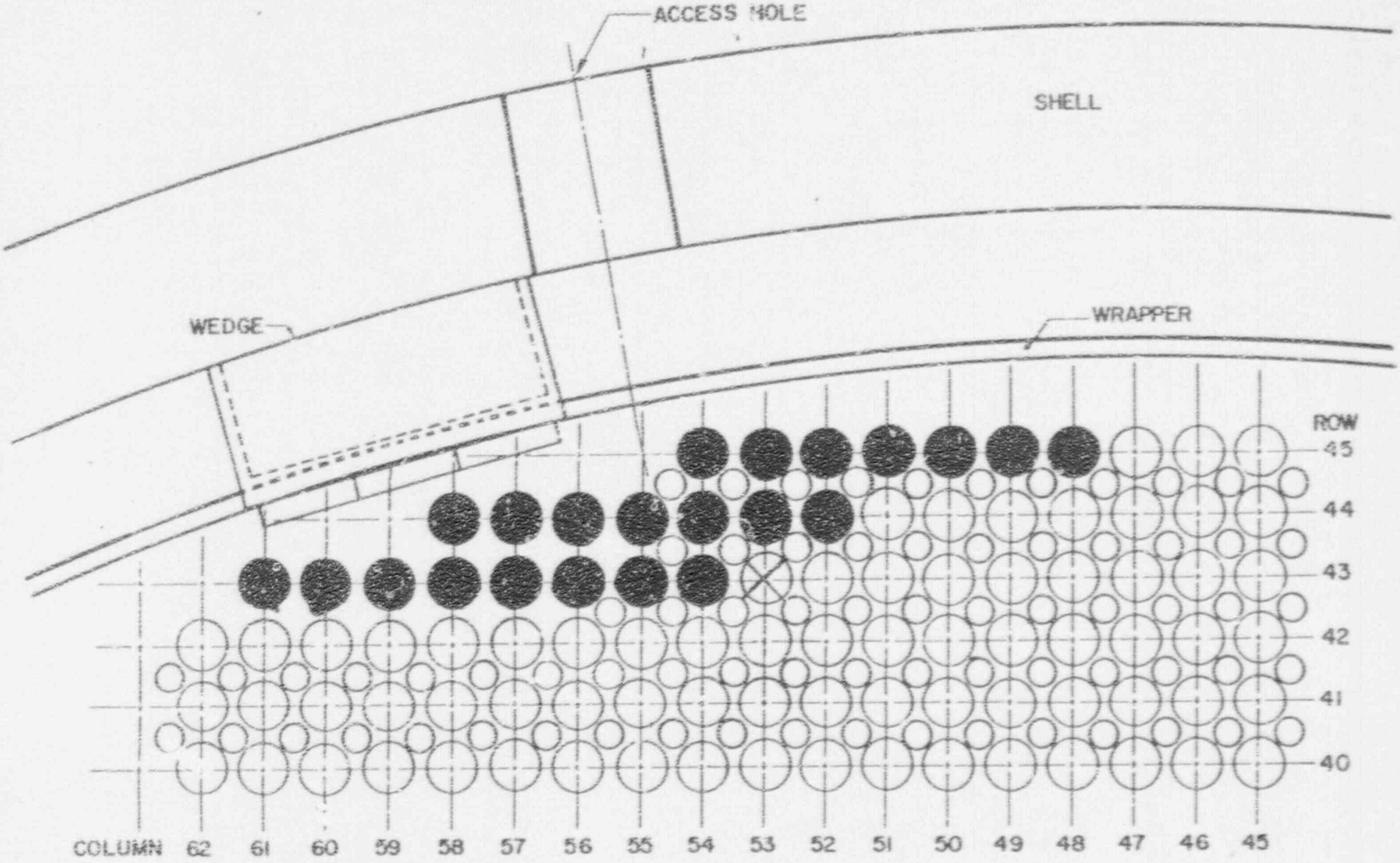
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

NO. 4 WEDGE AREA
APRIL 1981



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STEAM GENERATOR EVALUATION
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APRIL 30, 1982

NO. 4 WEDGE AREA
JANUARY 25, 1982

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

ANALYSIS AND TESTING PROGRAMS

DESIGN PARAMETERS

THERMAL HYDRAULIC EVALUATION

FLOW INDUCED VIBRATION

LATERAL IMPACT LOADS

AXIAL LOADS

COLLAPSE

FATIGUE

WEAR/BURST

COLLAPSE TESTING

FATIGUE TESTING

MODEL TESTING

CONCLUSIONS

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

DESIGN PARAMETERS

NO LOAD STEADY-STATE

$T_{hot} = T_{cold} = T_{av} = 547^{o}F$

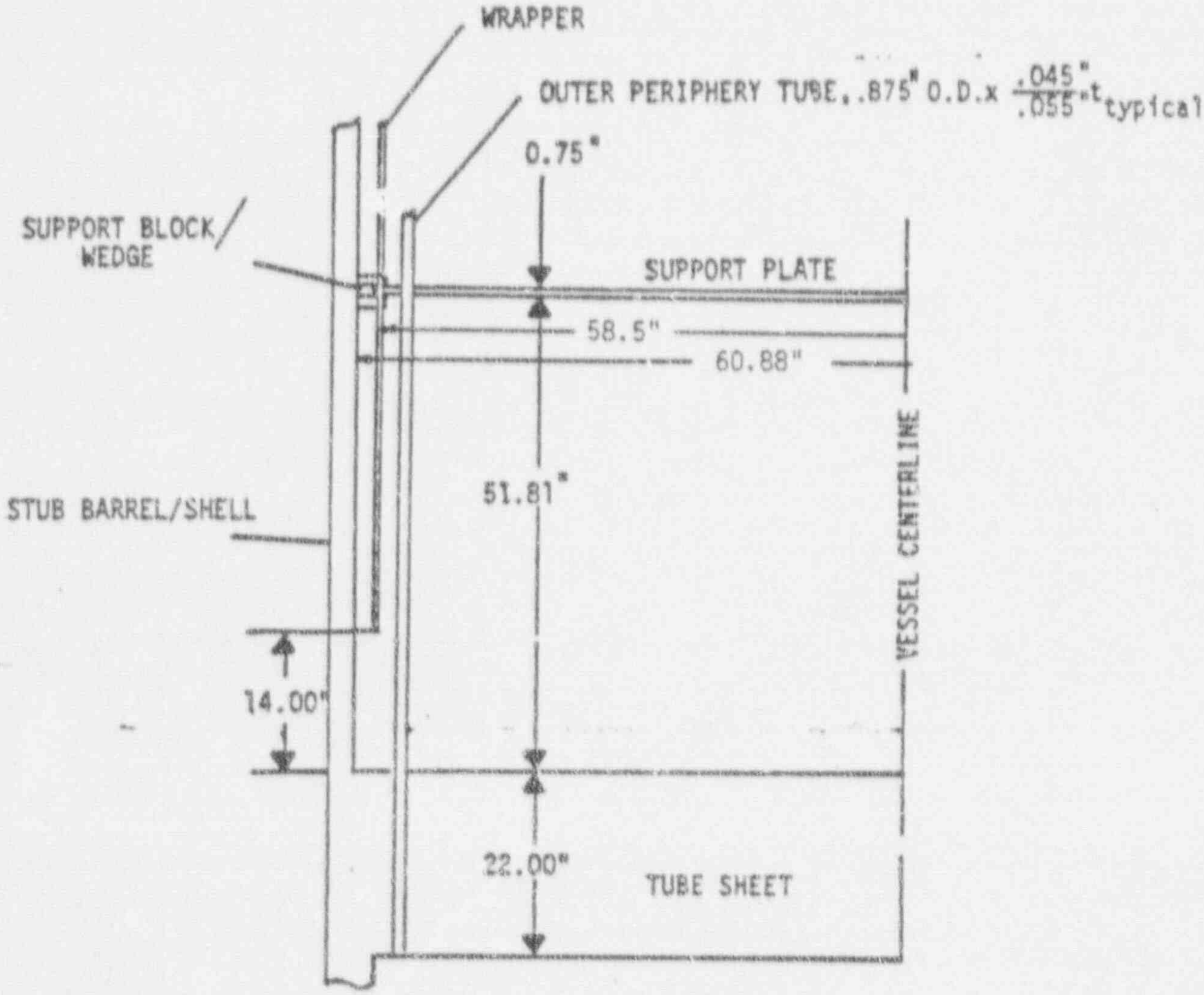
$P_p = 2250$ PSIA

$P_s = 1020$ PSIA

100% FULL-POWER

$T_{hot} = 603.8^{o}F$	$T_{cold} = 547.7^{o}F$
$T_{av} = 572.5^{o}F$	$T_{stm} = 516.3^{o}F$
$P_p = 2250$ PSIA	$P_s = 777$ PSIA
$T_{FW} = 414.7^{o}F$	RECIRC RATIO = 4.73

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982



Tubing

- Mill Annealed Inconel 600
- 7/8" O.D. X .050" Wall
- 2% Max. Ovality
- R.T. Yield 40 - 65 ksi
- S_a - 26.0 ksi at 10^6 Cycles

Tube Pitch 1.234"

GINNA STATION
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

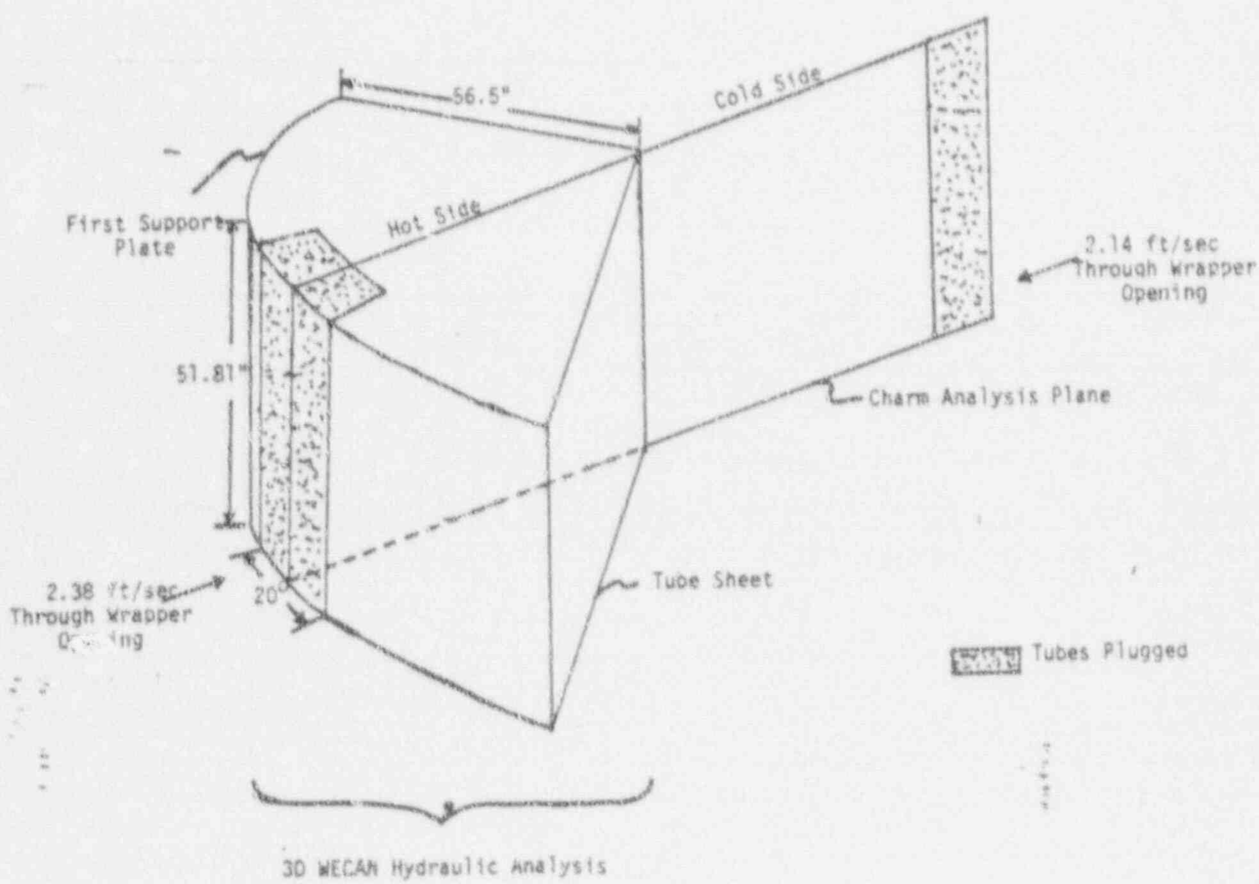
THERMAL HYDRAULIC EVALUATION

PURPOSE

DETERMINE FLUID VELOCITIES FOR USE IN EVALUATING FLUID-INDUCED LOADS

METHODS

CHARM ANALYSIS
WECAN ANALYSIS



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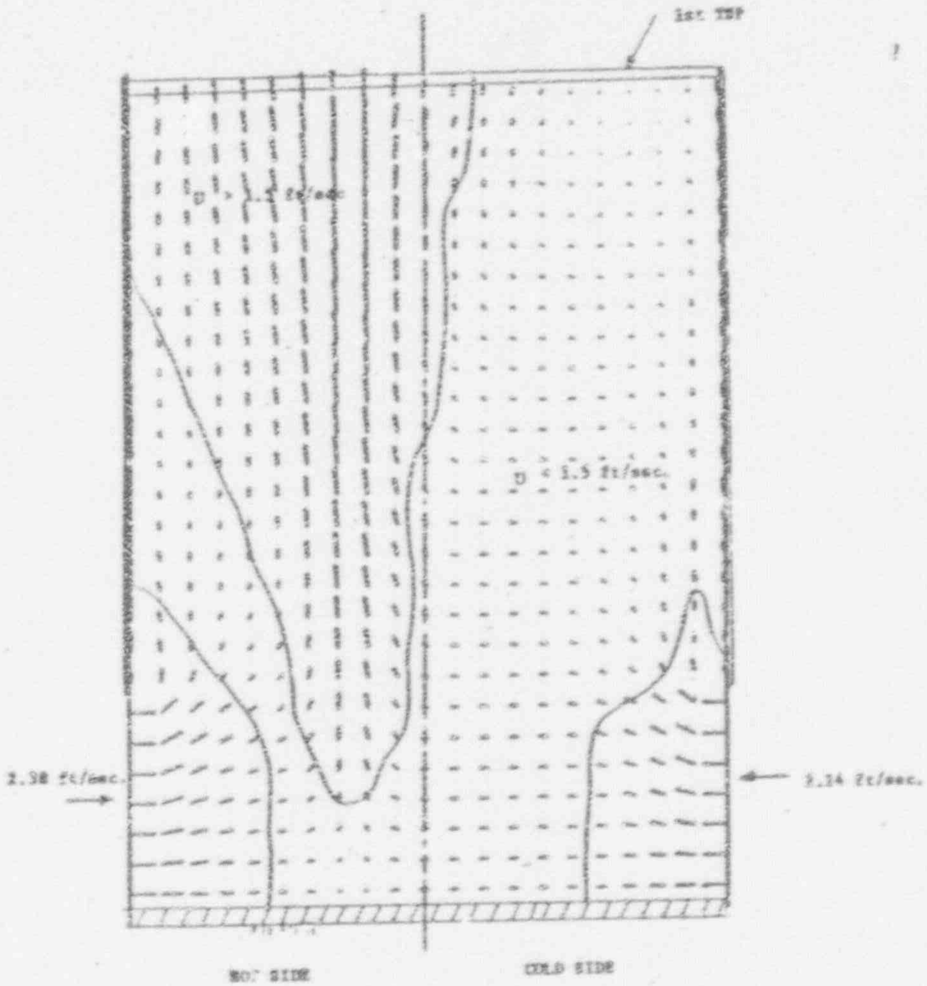
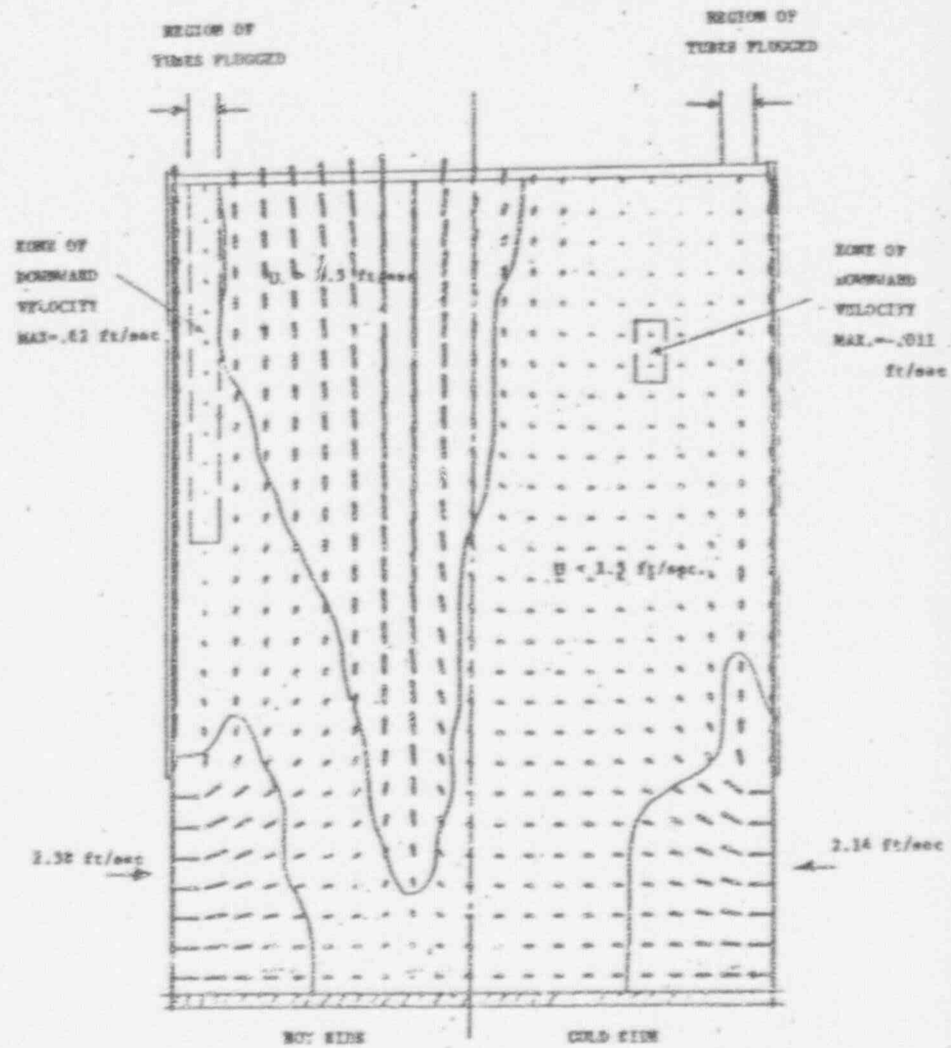
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982



GINNA BASE CASE



GINNA WITH TUBES PLUGGED

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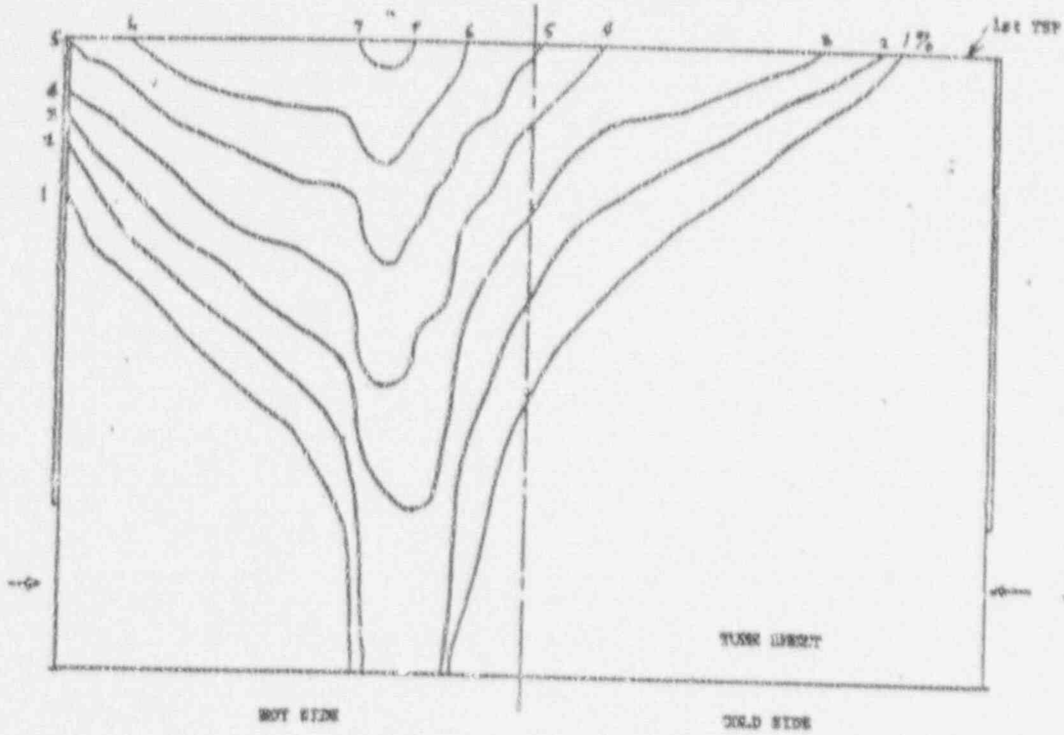
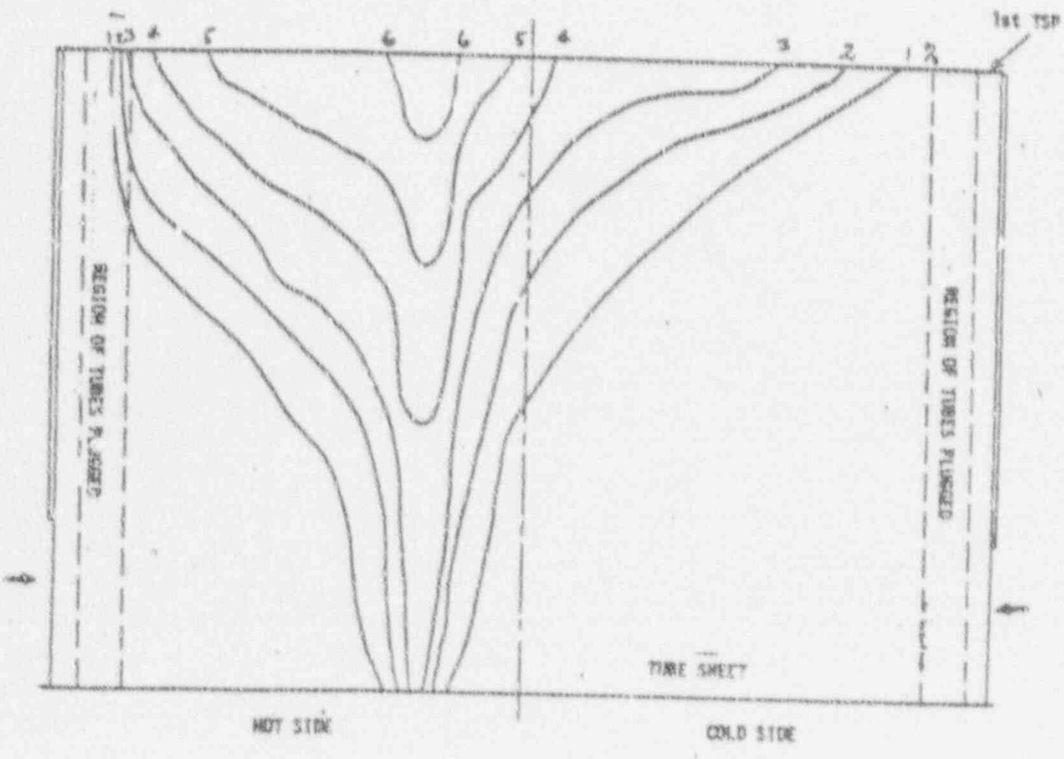
NRC MEETING
APRIL 30, 1982



QUALITY DISTRIBUTION OF GINNA BASE CASE



QUALITY DISYRIBUTION OF GINNA WITH TUBES PLUGGED

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NRC MEETING
APRIL 30, 1982

FLOW - INDUCED VIBRATION

PURPOSE

DETERMINE THE STABILITY CHARACTERISTICS AND MAGNITUDES OF FLOW-INDUCED VIBRATION DISPLACEMENTS AND/OR LOADS FOR HOT LEG TUBES BETWEEN THE TUBESHEET AND FIRST SUPPORT PLATE.

METHOD

- GROSS FLUID EFFECTS - DYNAMIC BEHAVIOR OF TUBES WITH VARIOUS CROSS-SECTIONS SUBJECTED TO CROSSFLOW.

- LOCAL FLUID EFFECTS - ESTIMATE MAGNITUDES OF DYNAMIC LOADS ACTING ON A TEAR IN STEAM GENERATOR TUBING.

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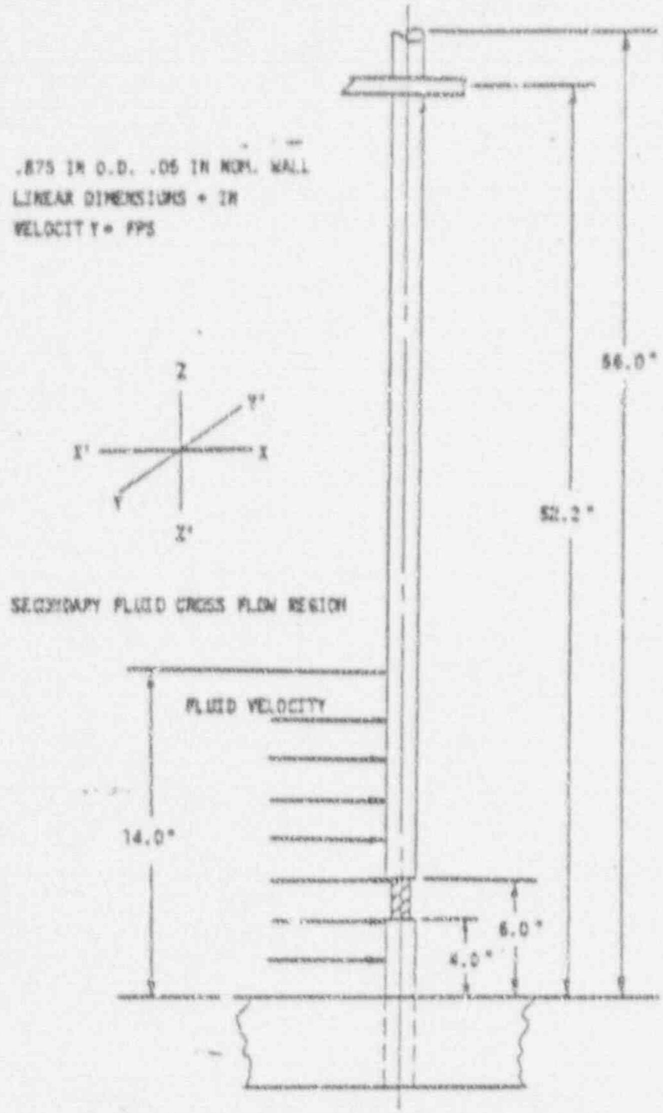
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982



BASIC ANALYSIS MODEL GEOMETRY

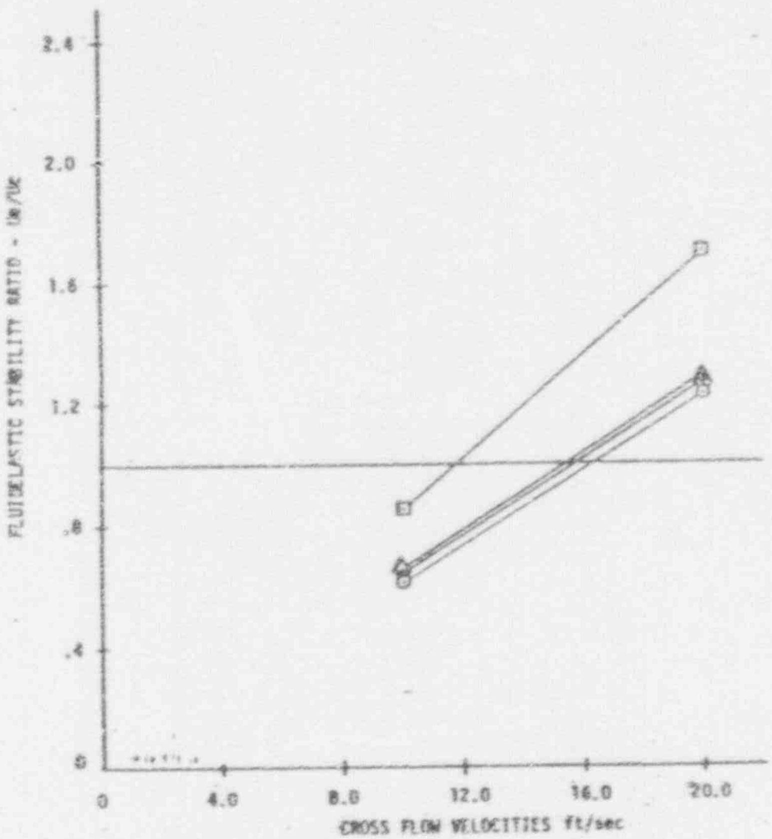
TUBE FUNDAMENTAL FREQUENCIES FOR
THE VARIOUS CASES ANALYZED

Cross Section	Fixed-Fixed	Fixed-Pinned	
	P=0	P=0	P= -1000 lbs
Cylinder	58.7 HZ	40.3 HZ	38.5 HZ
Flat	43.9 HZ	31.6 HZ	27.9 HZ
10 Ovalized	58.3 HZ	40.0 HZ	———
Kidney	58.2 HZ	39.9 HZ	———

STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982

FIXED-FIXED BOUNDARY CONDITIONS DAMPING RATIO = 0.01

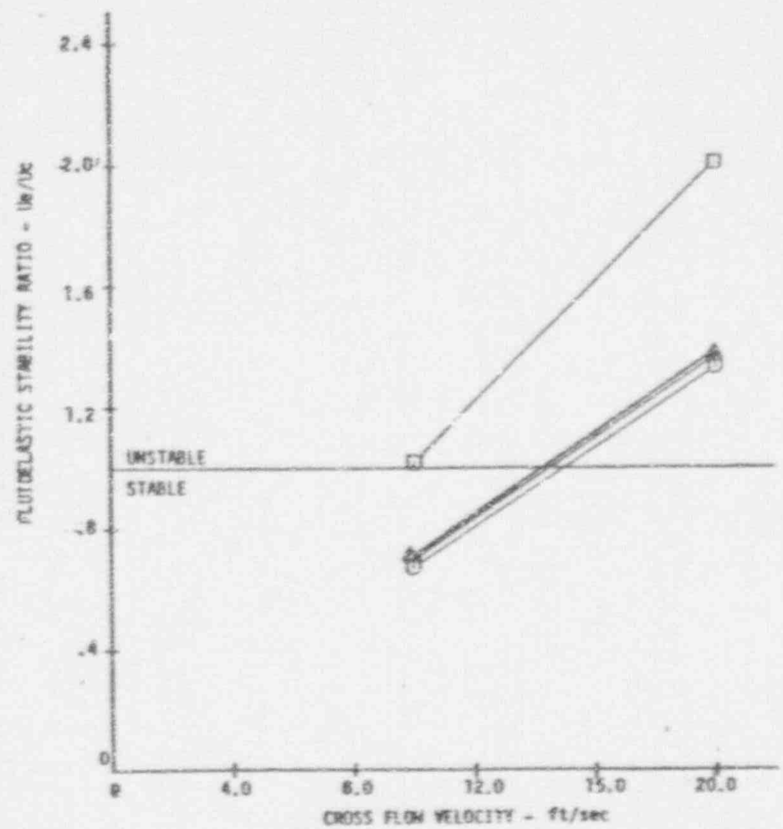
SYMBOL	CROSS SECTION	1ST MODE FREQ. - HZ
○	CIRCULAR	58.7
□	FLAT	48.9
◇	10% OVALIZED	58.3
△	KIDNEY	58.2



FLUIDELASTIC STABILITY RATIO OF TUBES AS A FUNCTION OF CROSS FLOW VELOCITY, FIXED-FIXED BOUNDARY CONDITIONS

FIXED-PINNED BOUNDARY CONDITIONS DAMPING RATIO = 0.01

SYMBOL	CROSS SECTION	1ST MODE FREQ. - HZ
○	CIRCULAR	40.3
□	FLAT	31.6
◇	10% OVALIZED	40.0
△	KIDNEY	39.5



FLUIDELASTIC STABILITY RATIO OF TUBES AS A FUNCTION OF CROSS FLOW VELOCITY, FIXED-PINNED BOUNDARY CONDITIONS

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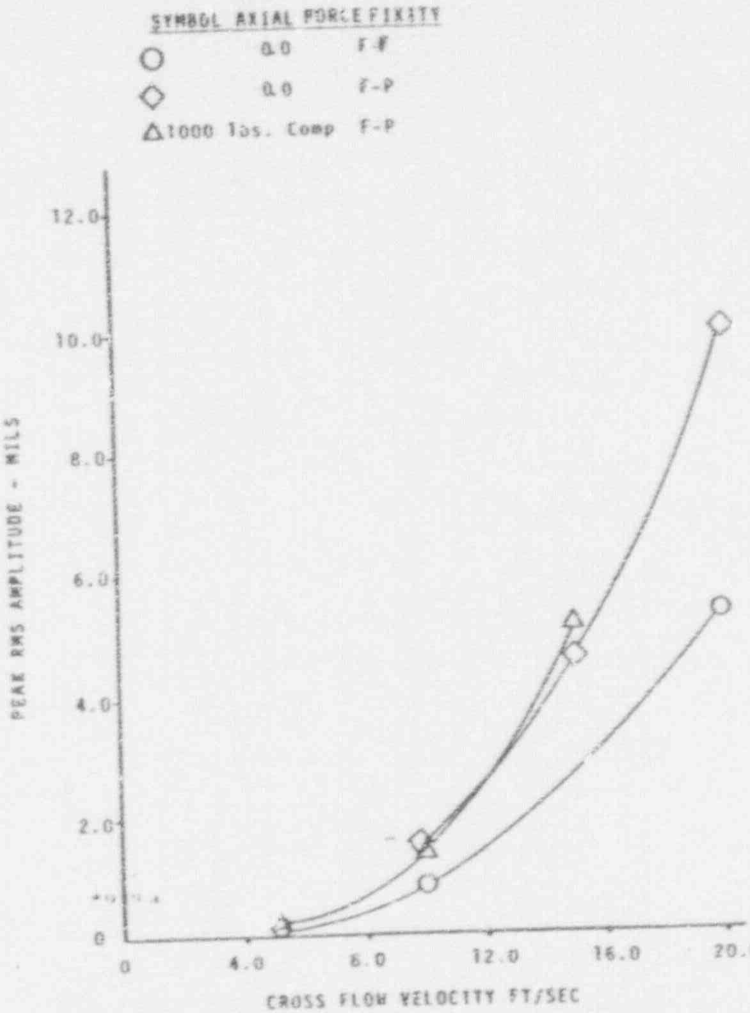
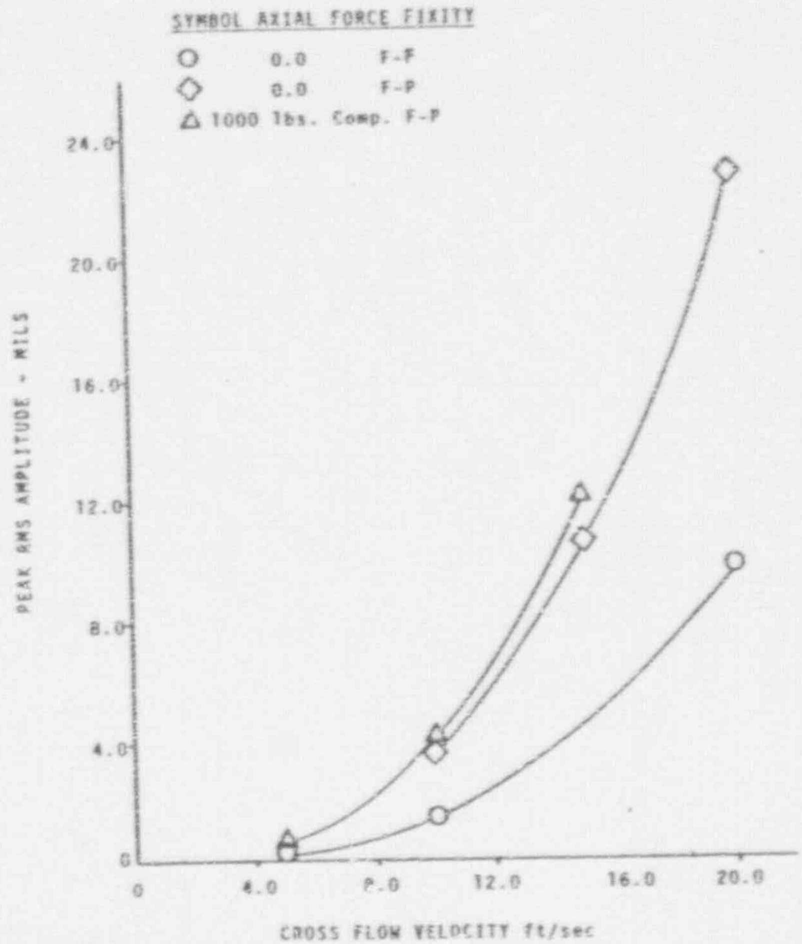
NRC MEETING
APRIL 30, 1982

CYLINDRICAL CROSS SECTION TUBE



PEAK RMS AMPLITUDE DUE TO TURBULENCE
AS A FUNCTION OF CROSS FLOW VELOCITY

FLAT CROSS SECTION TUBE



PEAK RMS AMPLITUDE DUE TO TURBULENCE
AS A FUNCTION OF CROSS FLOW VELOCITY

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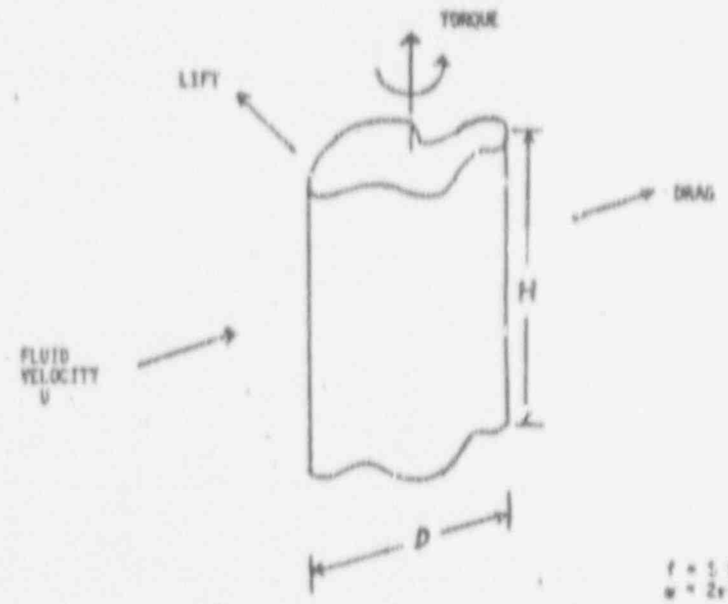
STEAM GENERATOR EVALUATION
NRC MEETING
APRIL 30, 1982



$f = S\,U/D$
$\omega = 2\pi f$

LIFT $= C_L\,(\rho U^2/2)\,HD\,\exp(i\omega t)$

DRAG $= C_D\,(\rho U^2/2)\,HD\,\exp(i\omega t)$

TORQUE $= C_L\,(\rho U^2/2)\,(HD^2/8)\,\exp(i\omega t)$

Tear Model for Calculating Vortex-Shedding Loads

VORTEX-SHEDDING INDUCED LOADS ON PROTRUSIONS FOR A FLUID VELOCITY OF 10.0 FT/SEC

H/D	H (Inches)	Lift (lbs)	Torque (in-lbs)	Lift & Torque Frequency (HZ)	Drag (lbs)	Drag Frequency (Hz)
1.0	0.1	.28 (10^{-2})	.35 (10^{-4})	240	.28 (10^{-3})	480
1.0	0.20	.11 (10^{-1})	.28 (10^{-3})	120	.11 (10^{-2})	240
1.0	0.30	.25 (10^{-1})	.95 (10^{-3})	80	.25 (10^{-2})	160
1.0	0.40	.45 (10^{-1})	.22 (10^{-2})	60	.45 (10^{-2})	120
1.0	0.50	.70 (10^{-1})	.44 (10^{-2})	48	.70 (10^{-2})	96
5.0	0.50	.14 (10^{-1})	.18 (10^{-3})	240	.14 (10^{-2})	480
5.0	1.00	.56 (10^{-1})	.14 (10^{-2})	120	.56 (10^{-2})	240
5.0	1.50	0.13	.47 (10^{-2})	80	.13 (10^{-1})	160
5.0	2.00	0.22	.11 (10-1)	60	.22 (10^{-1})	120
5.0	2.50	0.35	.22 (10^{-1})	48	.35 (10^{-1})	96

FOREIGN OBJECT INDUCED LOADS

Case	X_p (0) (inches)	U_{SS} (ft/sec)	U_A (ft/sec)	Model 1* Peak Load (lbs)	Model 2* Peak Load (lbs)
1	-5	2.3	0	26	107
2	0	.69	.69	.15	.28
3	0	.345	.69	.39	1.40

*Model 1: K_{SG} = 3717 lb/inches
M_{SG} = .03767 lb – sec^2/inches

Model 2: K_{SG} = 19439 lb/inches
M_{SG} = .01068 lb – sec^2/inches

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NRC MEETING
APRIL 30, 1982

ANALYSIS AND TESTING PROGRAMS

TUBE LOADINGS

- LATERAL IMPACT LOADS
- FLOW-INDUCED VIBRATIONS
- AXIAL LOADS
  - EXTERNAL PRESSURE
  - THERMAL GROWTH MISMATCHES
  - TUBE SHEET ROTATION AND MISALIGNMENT

TUBE DEGRADATION/FAILURE MECHANISMS

- COLLAPSE
- FATIGUE
- WEAR
- BURST

LABORATORY TESTING

- COLLAPSE
- FATIGUE
- FLOW MODEL

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APRIL 30, 1982

AXIAL LOADS ON PLUGGED TUBES

SOURCE	MAX. LOAD, LB.
• SECONDARY PRESSURE(1)	
- NO-LOAD SS, P = 1005 PSIG	-604.
- FULL-LOAD SS, P = 762 PSIG	-458.
• TUBE-TO-SHELL INTERACTION(2)	
- HOT STANDBY: TUBE-547°F, SHELL-476°F	-1940.
- PLANT LOADING: TUBE-517°F, SHELL-547°F	825.
- PLANT UNLOADING: TUBE-547°F, SHELL-497°F	-1375.
- FULL POWER SS: TUBE-517°F, SHELL-497°F	-550.
• PLUGGED-TO-ACTIVE TUBE INTERACTION(2)	
- FULL POWER SS: T_P=517°F, T_A=565°F	1410.

NOTES: (1) TUBE FREE TO MOVE AXIALLY AT TSP
(2) TUBE AXIALLY RESTRAINED AT TSP

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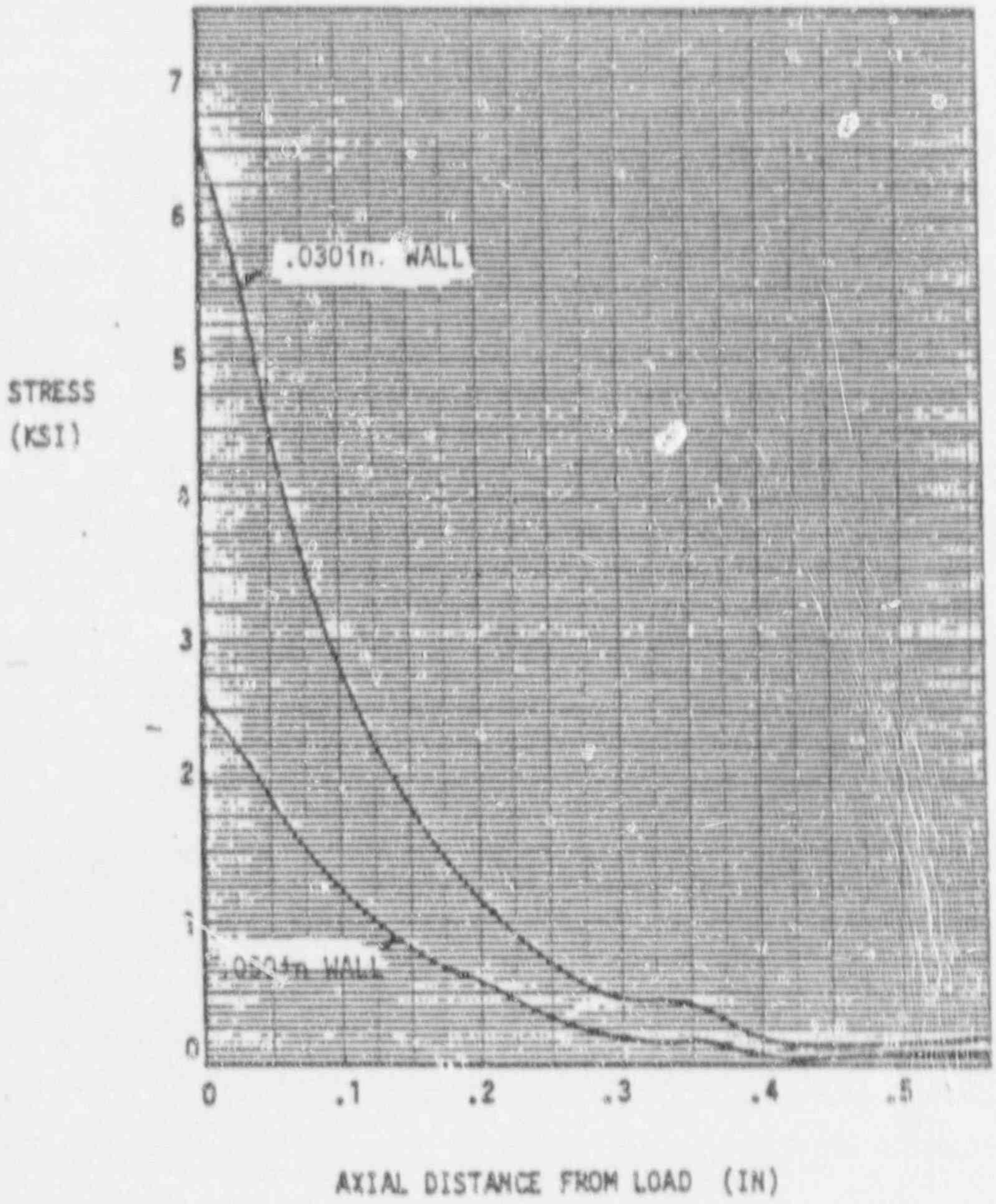
NRC MEETING
APRIL 30, 1982



HOOP STRESS IN OUTER FIBER
DUE TO 5 POUND RADIAL LOAD

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APRIL 30, 1982

SUMMARY OF COLLAPSE ANALYSIS

- FOR A NOMINAL 0.050 INCH WALL TUBE WITH 2% OVALITY AND 58 KSI YIELD STRENGTH, A LATERAL LOAD OF 75 LB. IS REQUIRED FOR INCIPIENT YIELDING.

- AXIAL LOAD OF 1000 LB. TENSION REDUCES THE INCIPIENT YIELD LOAD BY APPROXIMATELY 10.0 LB.

- HOWEVER, THE STRESS FIELD IS HIGHLY LOCALIZED. TO PRECIPITATE COLLAPSE IT IS REQUIRED THAT:

  1. THE LOAD BE SIGNIFICANTLY HIGHER IN ORDER TO OBTAIN A THROUGH-WALL PLASTIC ZONE (HINGE) AND,

  2. THE LOAD MUST ACT AT DIFFERENT TUBE LOCATIONS, IN ORDER TO PLASTICALLY DEFORM AN AREA 2 TO 3 TUBE DIAMETERS LONG AXIALLY.

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APRIL 30, 1982

SUMMARY OF PLUGGED TUBE FATIGUE ANALYSIS
(DESIGN MIN. WALL =0.045 INCH, NOMINAL OVALITY=2%)

. STRUCTURALLY STABLE TUBES

PRINCIPAL LOADING: THERMAL MECHANICAL DUTY CYCLES (LOW CYCLE FATIGUE)

CASE	LATERAL LOAD IMPACT	USAGE FACTOR(U)
1. NOMINALLY PLUGGED	NONE	0.036
2. NOMINALLY PLUGGED WITH NOTCH	NOTCH (SCF=4.0)	0.676

. STRUCTURALLY DEGRADED TUBES (LOCALLY COLLAPSED)

PRINCIPAL LOADING: FLOW-INDUCED VIBRATIONS (CROSS-FLOW TUBBULENCE)
(HIGH CYCLE FATIGUE)

CASE	LATERAL LOAD LB. (CONTINUOUSLY ACTING)	POSTULATED FAILURE (U=1.0)TIME, DAYS
3. FIXED-PINNED TUBE SPAN LOCALLY COLLAPSED OVER 2.0 INCH LENGTH NEAR TUBE SHEET	0.	NO FAILURE
	10.	30.
	25.	2.5
	50.	0.7

. SUMMARY

. STRUCTURALLY STABLE TUBES HAVE ACCEPTABLE FATIGUE MARGINS

. STRUCTURALLY DEGRADED TUBE, WHEN CONTINUOUSLY ACTED UPON BY A LATERAL LOAD, CAN FAIL IN HIGH CYCLE FATIGUE DUE TO FLUID INTERACTION.

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APRIL 30, 1982

WEAR CORRELATION

- BASIC
  - INCONEL ON INCONEL SPECIFIC WEAR COEFF. RANGE = 58 to 854 IN^2/lb.
  - WEAR VOLUME BASED ON RUB AREA = 8.0" LONG X 0.1" WIDE X 0.0434" DEEP
  - CONTACT FORCE BASED ON DRAG FORCE ON SEVERED TUBE = 3.0 LBS.
  - RUB VELOCITY = 5 in/sec USING ARCHARD THEORY

    RUB FREQUENCY = 50 HZ AND

    MEAN RUB AMPLITUDE = 0.050 INCH
  - ESTIMATED TIME RANGE FOR TUBE BURST, 59 DAYS TO 1.27 YEARS

- CONCLUSIONS
  - FOR THE CASE OF A SEVERED TUBE (AT TUBE SHEET) RUBBING ON A NEIGHBORING TUBE, CALCULATED FAILURE TIME PERIODS REASONABLY ENVELOP THE OBSERVED WEAR RANGE
  - WEAR BY UNSEVERED TUBE NOT POSSIBLE SINCE FLOW-INDUCED VIBRATION AMPLITUDES ARE SMALL AND WILL NOT RESULT IN TUBE CONTACT

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APRIL 30, 1982

BURST CORRELATION

- MINIMUM TUBE WALL AT BURST
  - ASME CODE NB - 3324.1
  - $P_m = S_u = 89.7$ KSI
  - $P_o = 780$ PSI
  - $P_i = 2250$ PSI
  - $t = 0.0066$ INCH

- MEASUREMENT OF (MINIMUM) WALL OF BURST TUBE (R42 C55)
  - $t = 0.008$ INCH

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NRC MEETING
APRIL 30, 1982

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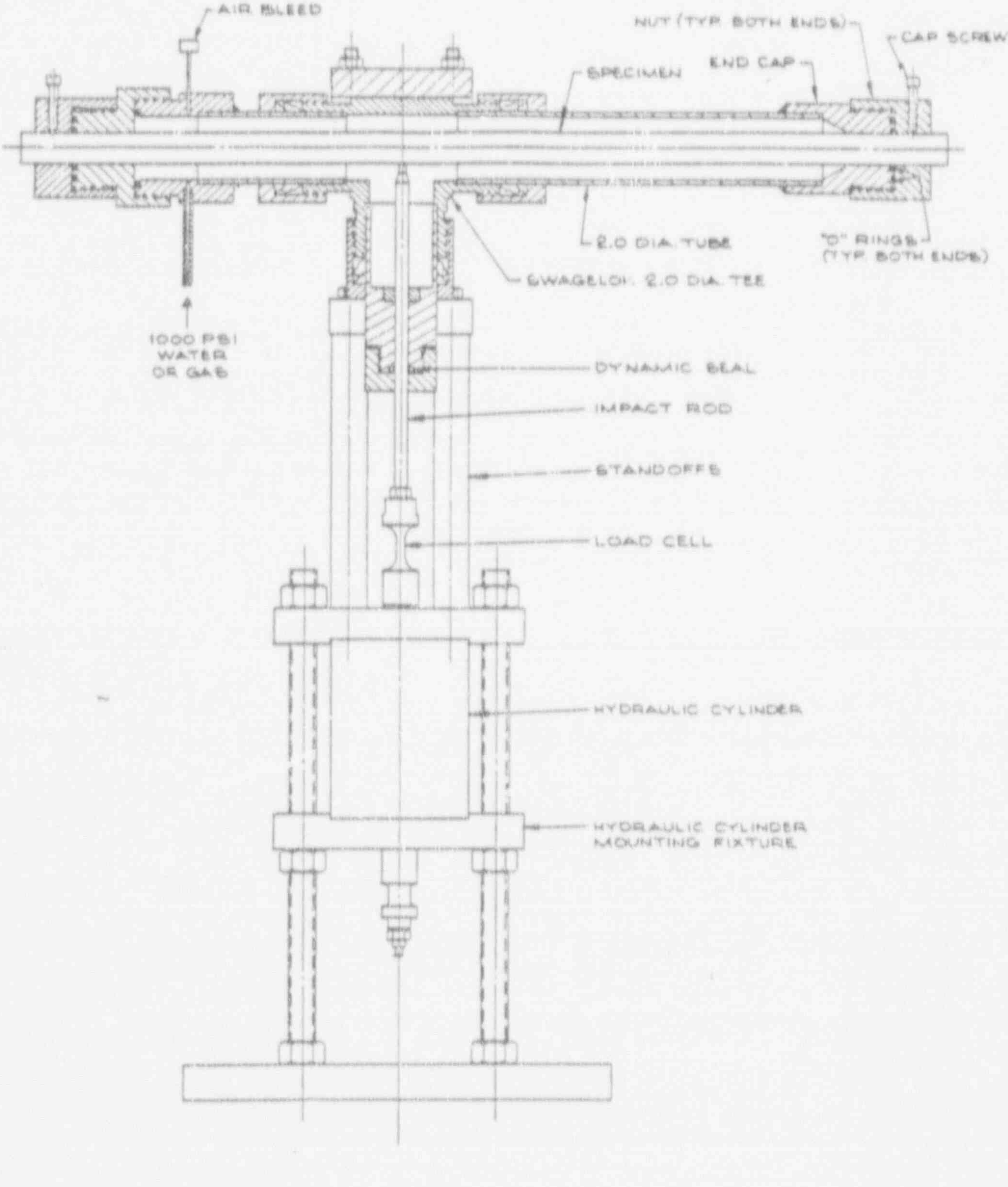

SCHEMATIC-TUBE COLLAPSE TEST

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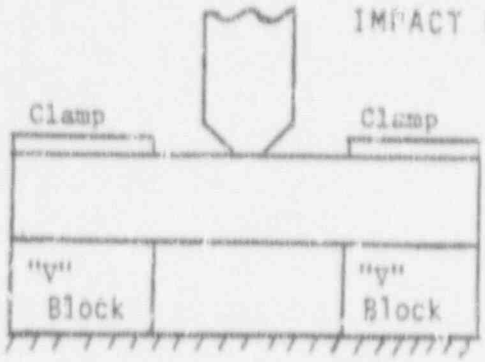
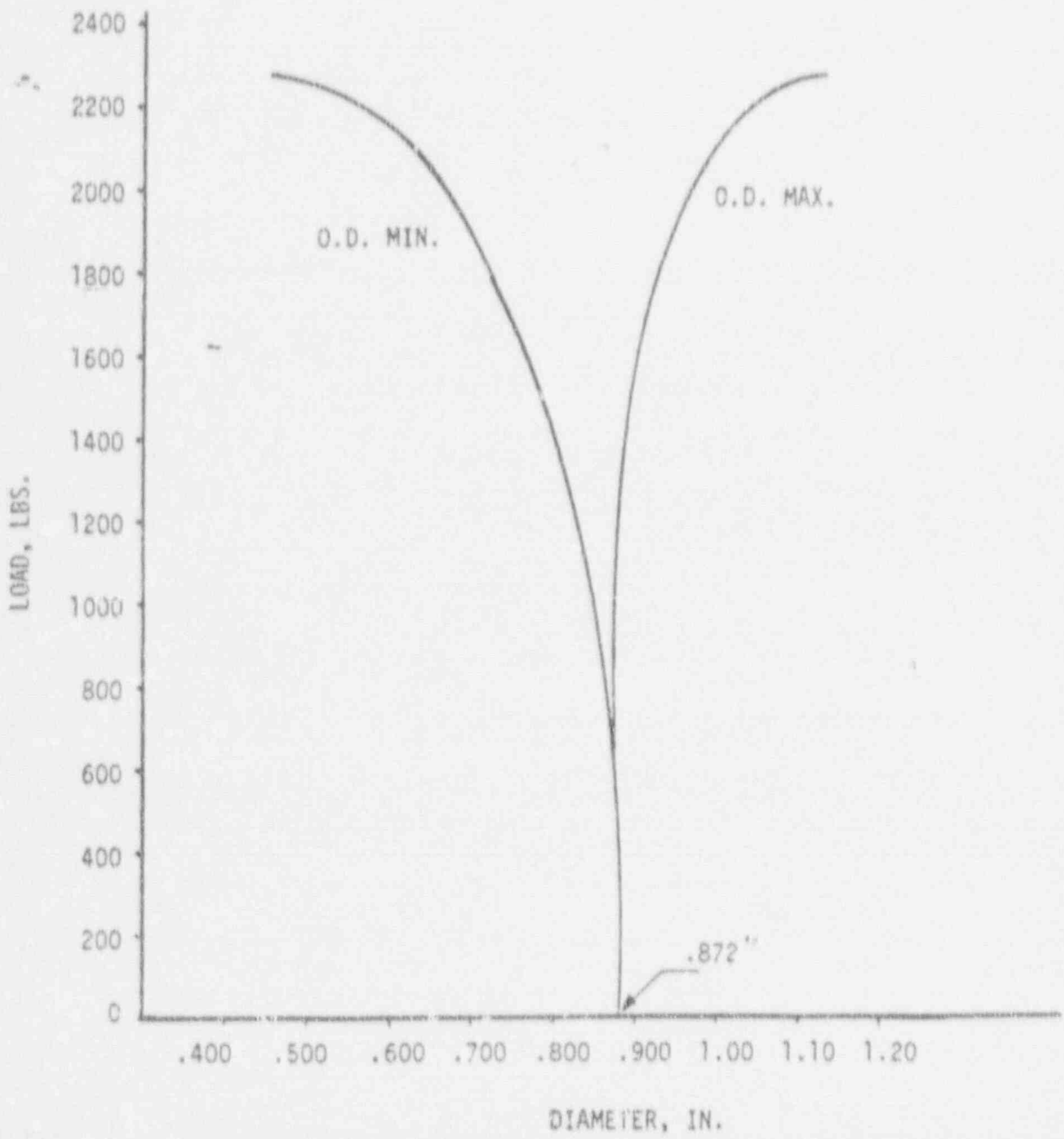
APRIL 30, 1982

LATERAL LOAD BENCH TEST-STATIC



IMPACT ROD - TIP .125" x 1.00", 90° TAPER

TUBE, O.D. = .872"
I.D. = .766"
OVALITY = "0" %



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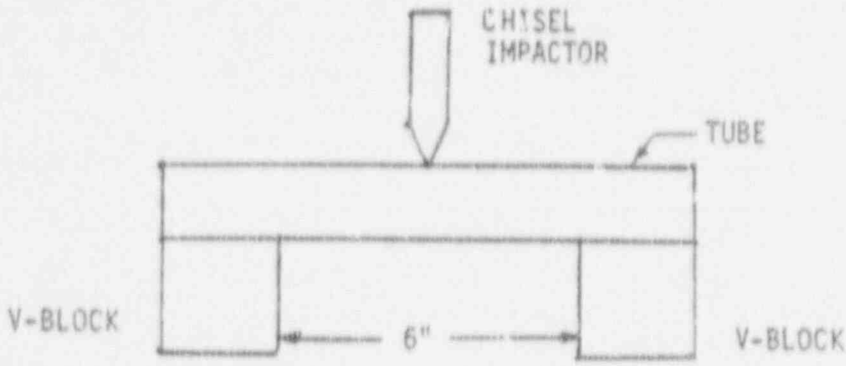
NRC MEETING
APRIL 30, 1982

LATERAL LOAD BENCH TEST-STATIC



LOAD (LBS)	OVALITY (%)	
	SHARP CHISEL	BLUNT CHISEL
0	0	0
550	1.1	0.2
700	2.1	0.8
850	3.9	1.9
1000	6.0	3.7

Chisel Face:

Sharp = 0.030" x 1.0"

Blunt = 0.125" x 1.0"

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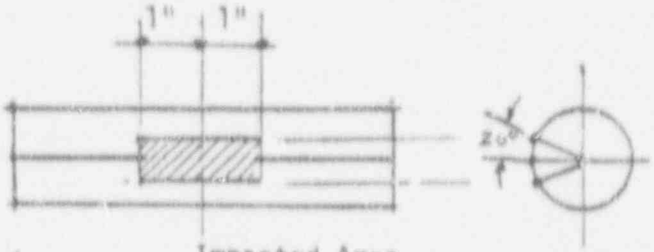
NRC MEETING
APRIL 30, 1982

DYNAMIC COLLAPSE TESTING



Impacted Area

- O EXTERNAL PRESSURE = 1000 PSI
- O LATERAL LOAD = 600 LBS
- O IMPACTING SEQUENCE:
  - a. IMPACT AT EVERY 1/4" ALONG 0-DEGREE PLANE FROM ONE END OF AREA TO THE OTHER. THEN OFFSET 1/8" AND BACKTRACK.
  - b. REPEAT STEP "a" ON THE + 20° PLANE
  - c. REPEAT STEP "b" ON THE -20° PLANE.

SEQUENCE NO.	SHARP CHISEL (1.00"x0.030")			BLUNT CHISEL (1.00"x0.125")		
	OD MAX/ODMIN (INCH)	ID MIN (INCH)	OVALITY(%)	ODMAX/ODMIN (INCH)	ID MIN (INCH)	OVALITY (%)
INITIAL	0.884/0.861	0.755	2.6%*	0.873/0.873	0.766	0%
1	--	0.724	--	--	0.734	--
2	--	0.699	--	--	0.695	--
3	--	0.680	--	--	0.664	--
4	--	0.660	--	--	0.647	--
FINAL	0.928/0.733	--	23.5%	0.923/0.766	--	18.5%
	NO COLLAPSE, DEVELOPED CRACK DURING SUBSEQUENT IMPACTS			TEST IN PROGRESS ΔP INCREASED TO 1500 PSI		

*OVALIZED IN A VISE

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NRC MEETING
APRIL 30, 1982

FATIGUE TEST SUMMARY

DEFORMATION SHAPE	CONFIGURATION	AXIAL TENSION LBS.	AMPLITUDE, INCH (PEAK TO PEAK)	FREQUENCY HZ	CYCLES TESTED	RESULTS
FLAT 2" LONG	FIXED-SUPPORTED	0.	.05 .07 .09 .12 .16 .20 .25 .30 .30 .40 .50	39.3	$.3 \times 10^6$ (EACH LEVEL) 3.6×10^6	NO FAILURE
FLAT 2" LONG	FIXED-FIXED	1000.	.05 .30	63.6	2×10^6	NO FAILURE
FLAT 2" LONG	FIXED-FIXED, NOTCHED AT TRANSITION	1000.	.30	63.6	$.9 \times 10^6$	FAILED - NOT AT NOTCH
KIDNEY-SHAPED 1" LONG	FIXED-FIXED	1000.	.30	67.7	6×10^6	NO FAILURE
KIDNEY-SHAPED 2" LONG	FIXED-FIXED, LATERALLY IMPACTED	1000.				

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NRC MEETING
APRIL 30, 1982

MODEL TEST

OBJECTIVES

NATURE AND EXTENT FOREIGN OBJECT MOBILITY

MAGNITUDE FOREIGN OBJECT IMPACT LOADS

STABILITY CHARACTERISTICS OF DEGRADED TUBES

NATURE AND EXTENT OF TUBE-TO-TUBE INTERACTION

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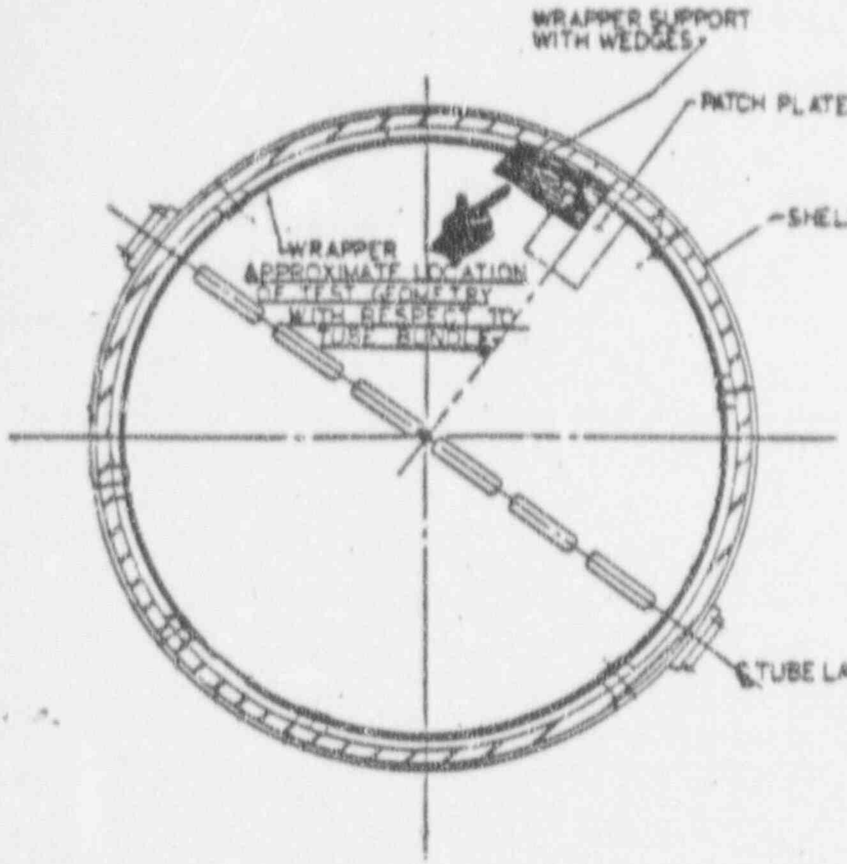
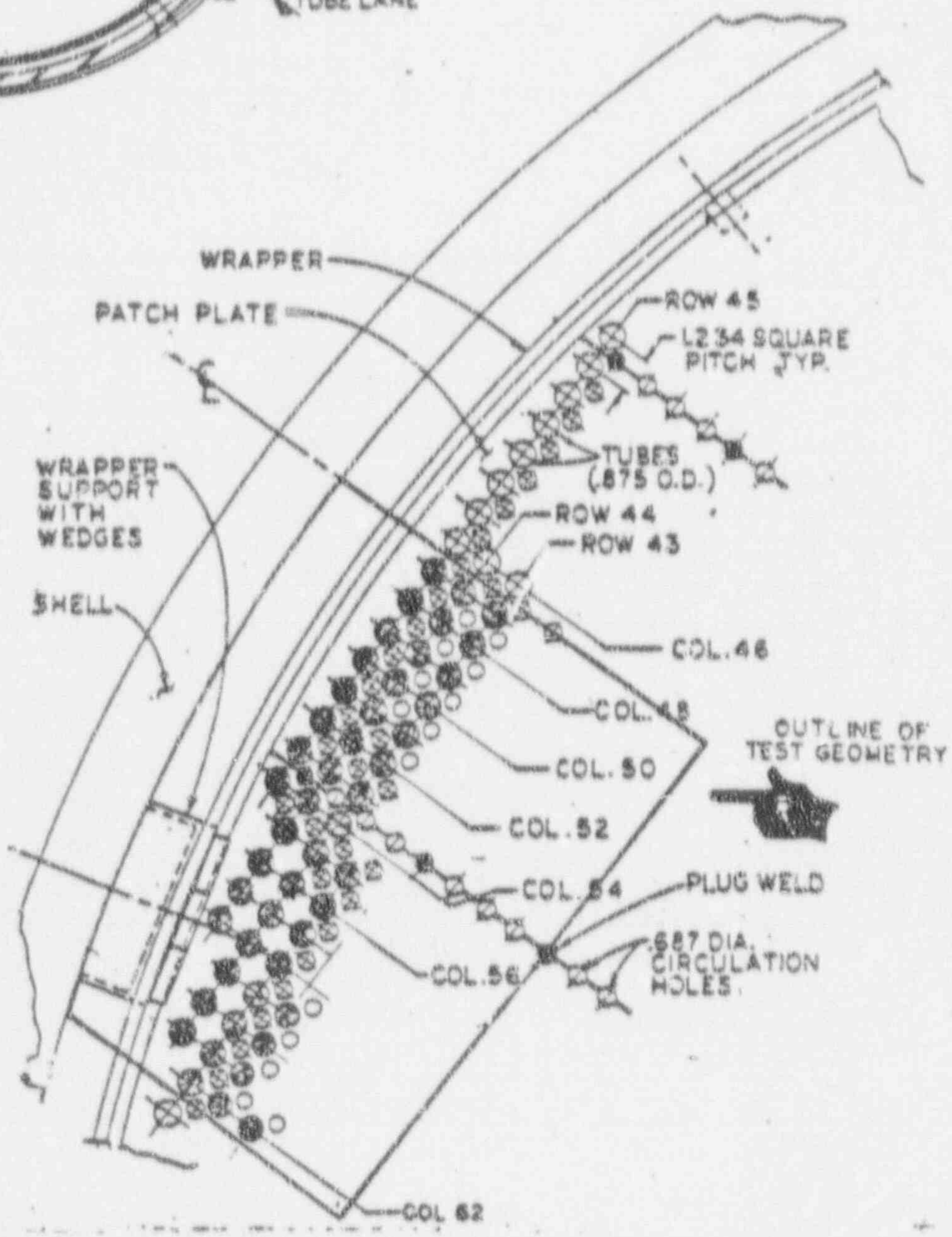
NRC MEETING
APRIL 30, 1982





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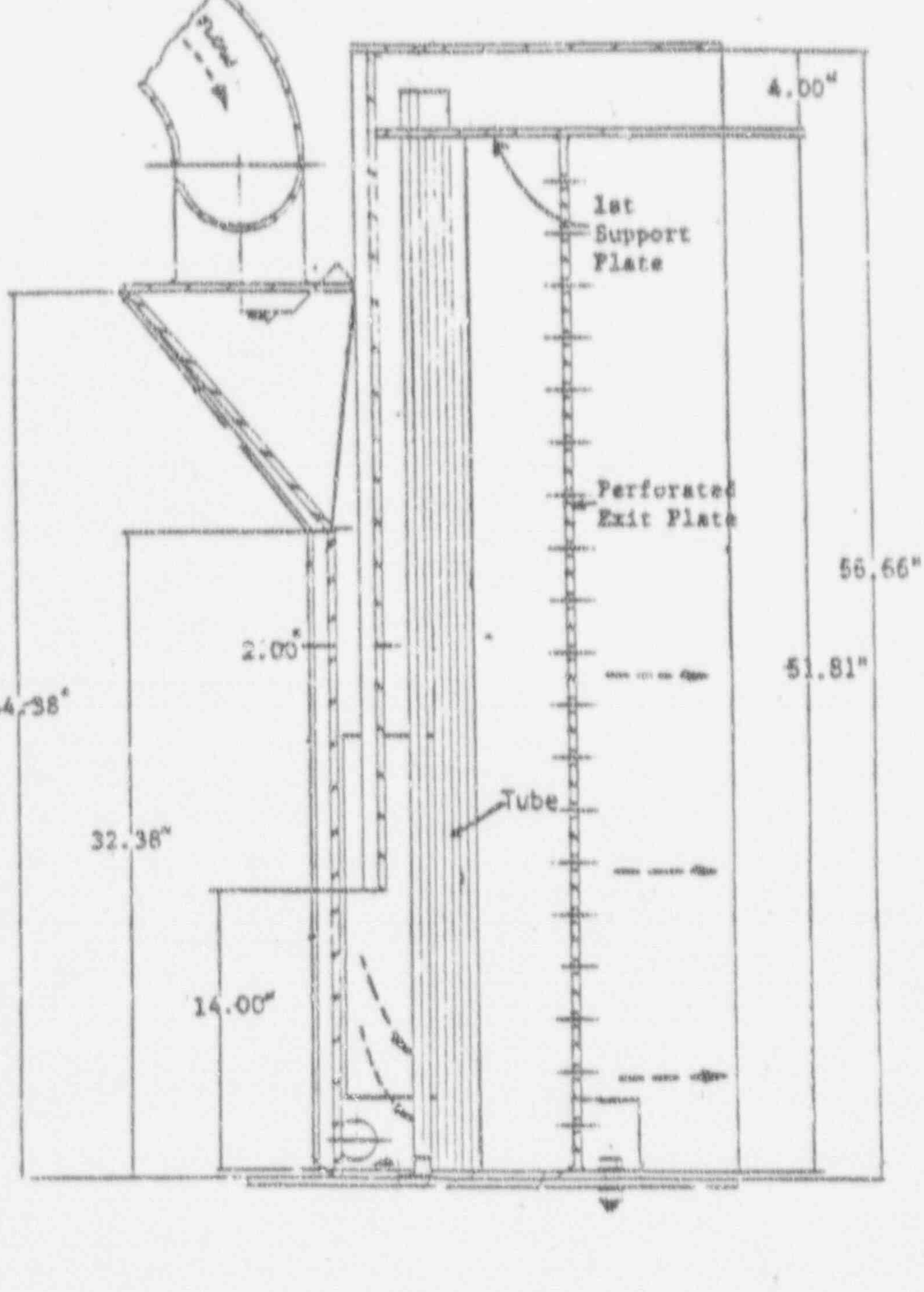
NRC MEETING
APRIL 30, 1982



Section View of Flow Test Model

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APRIL 30, 1982

MODEL TEST

CONCLUSIONS

OBJECT MOTION RANDOM IN NATURE AND OCCURRED FOR ALL ORIENTATIONS AND POSITIONS

MAXIMUM FOREIGN OBJECT IMPACT FORCES

- ACCELEROMETER DATA 120 - 180 LBS.
- FORCE TRANSDUCERS 200 - 350 LBS.

TUBES WITH UNDEGRADED AND LOCALLY DEGRADED CROSS SECTIONS WERE STABLE

SEVERED TUBE TENDED TO NESTLE BETWEEN NEIGHBORING TUBES AND INTERMITTENT IMPACTS WERE OBSERVED

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APRIL 30, 1982

CONCLUSIONS

COLLAPSE

- SIGNIFICANT LEVELS OF COLD WORK FOUND ON COLLAPSED TUBING SURFACE
- LARGE FOREIGN OBJECT REMOVED FROM STEAM GENERATOR CAPABLE OF PROVIDING 100 - 350 LBS. IMPACT LOADS
- LATERAL IMPACT LOADS IN THE RANGE OF 50 - 75 LBS. WILL IN COMBINATION WITH A 1000 PSI EXTERNAL PRESSURE CAUSE INCIPIENT YIELDING

FATIGUE

- STRUCTURALLY DEGRADED TUBE CAN FAIL DUE TO HIGH CYCLE FATIGUE WHEN SUBJECTED TO CONTINUING LATERAL IMPACT LOADS
- STRUCTURALLY DEGRADED TUBE WILL NOT FAIL DUE TO FLUID INDUCED VIBRATION
- NOMINAL PLUGGED TUBE WITH A NOTCH OR STRESS RISER WILL NOT FAIL IN FATIGUE
- FATIGUE TYPE STRIATIONS FOUND AT FAILED TUBING SURFACES

WEAR

- TUBE WEAR PATTERNS COMPATIBLE WITH ONE TUBE RUBBING AGAINST ANOTHER
- SUFFICIENT WEAR CAN OCCUR TO CAUSE BURSTING OF A TUBE
- EDDY CURRENT DATA SUBSEQUENT TO FEBRUARY 1979 CONSISTENT WITH PROPAGATION OF DEGRADATION BY WEAR PROCESS

BURST

- LABORATORY EXAMINATION INDICATED A PURELY DUCTILE FAILURE R42 C55
- CALCULATED AND OBSERVED BURST THICKNESS ARE CONSISTENT

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APRIL 30, 1982

REPAIR PROGRAM

- ACCESS HOLES
- TUBE REMOVAL
- TUBE PULL
- LOOSE PARTS
- MECHANICAL PLUG REMOVAL
- INSPECTIONS AND TESTS
- RADIATION EXPOSURE

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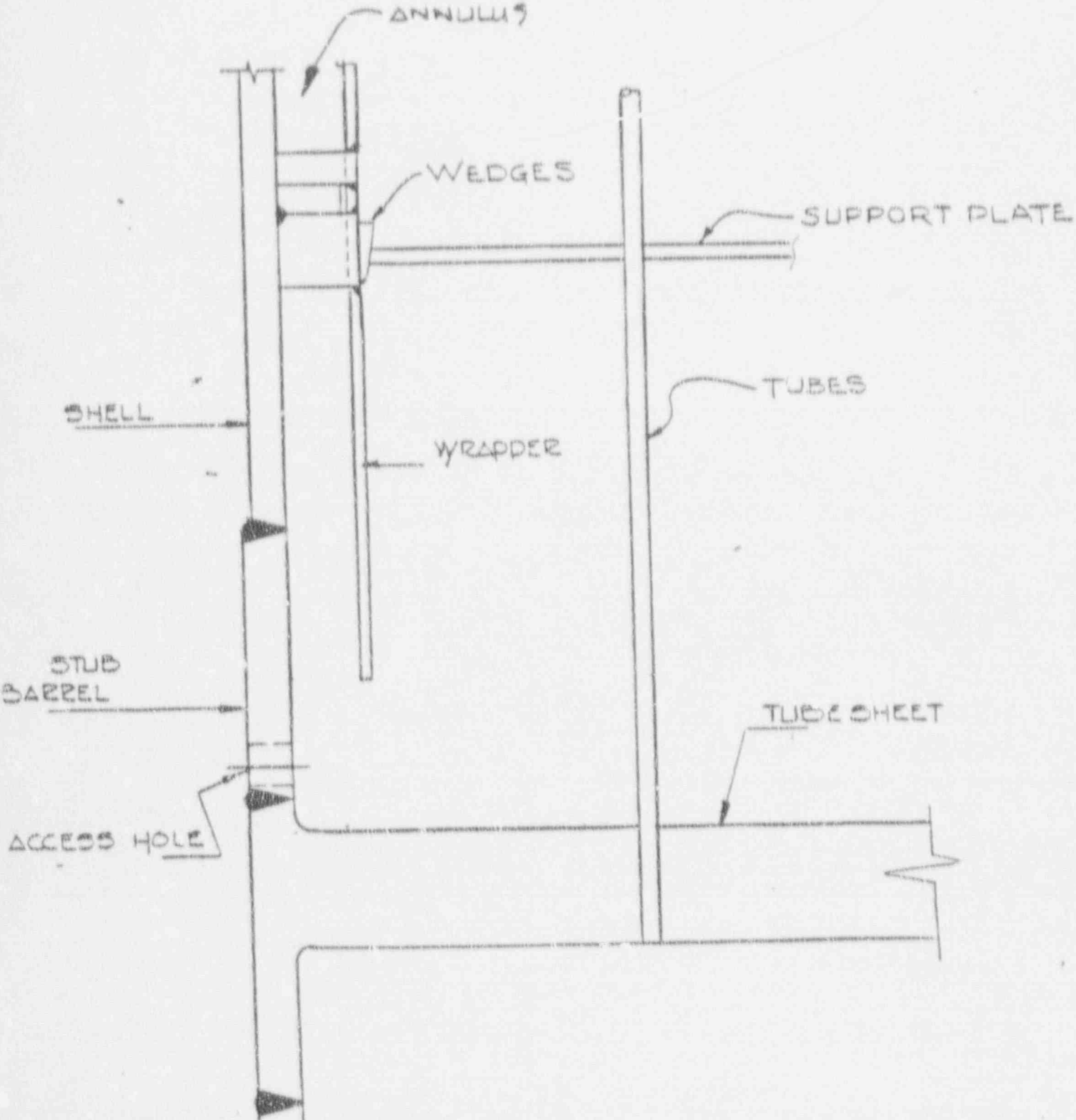
NRC MEETING
APRIL 30, 1982

ACCESS HOLE LOCATION



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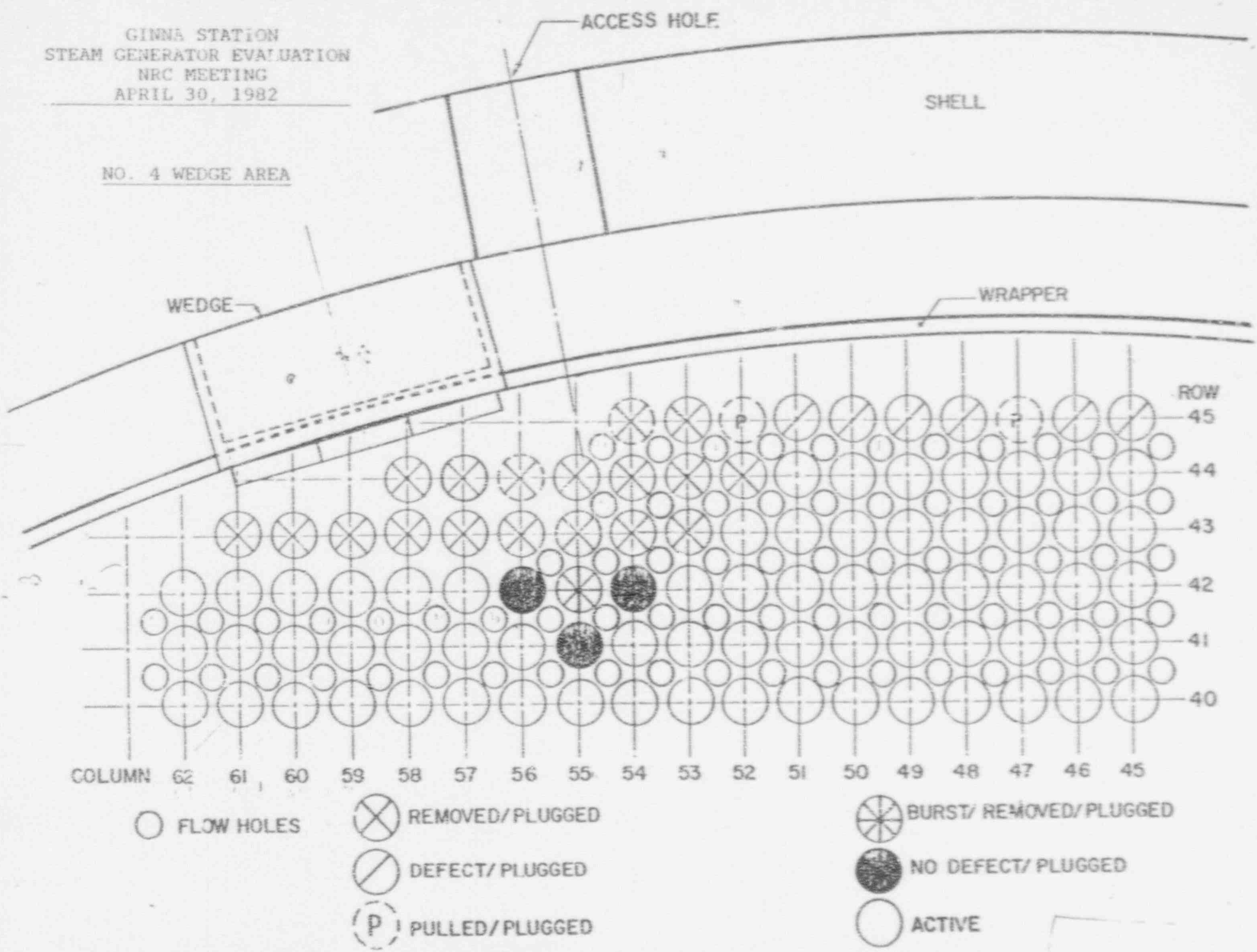
APRIL 30, 1982

NO. 4 WEDGE AREA



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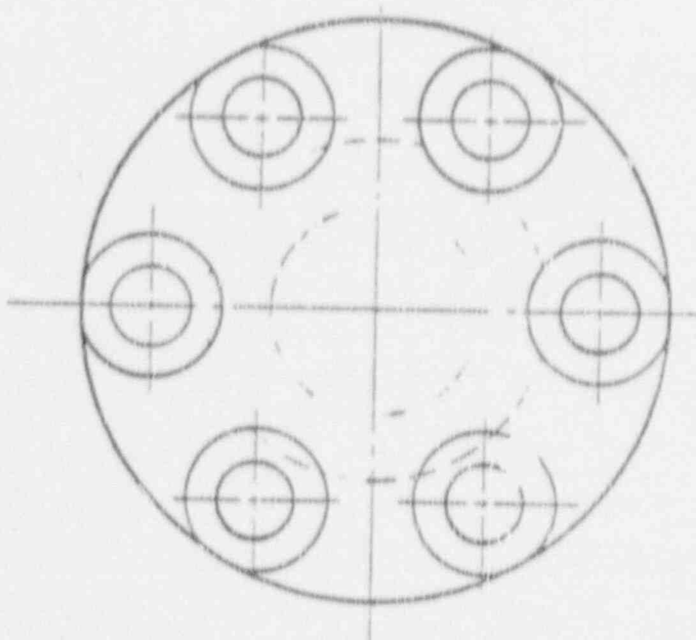
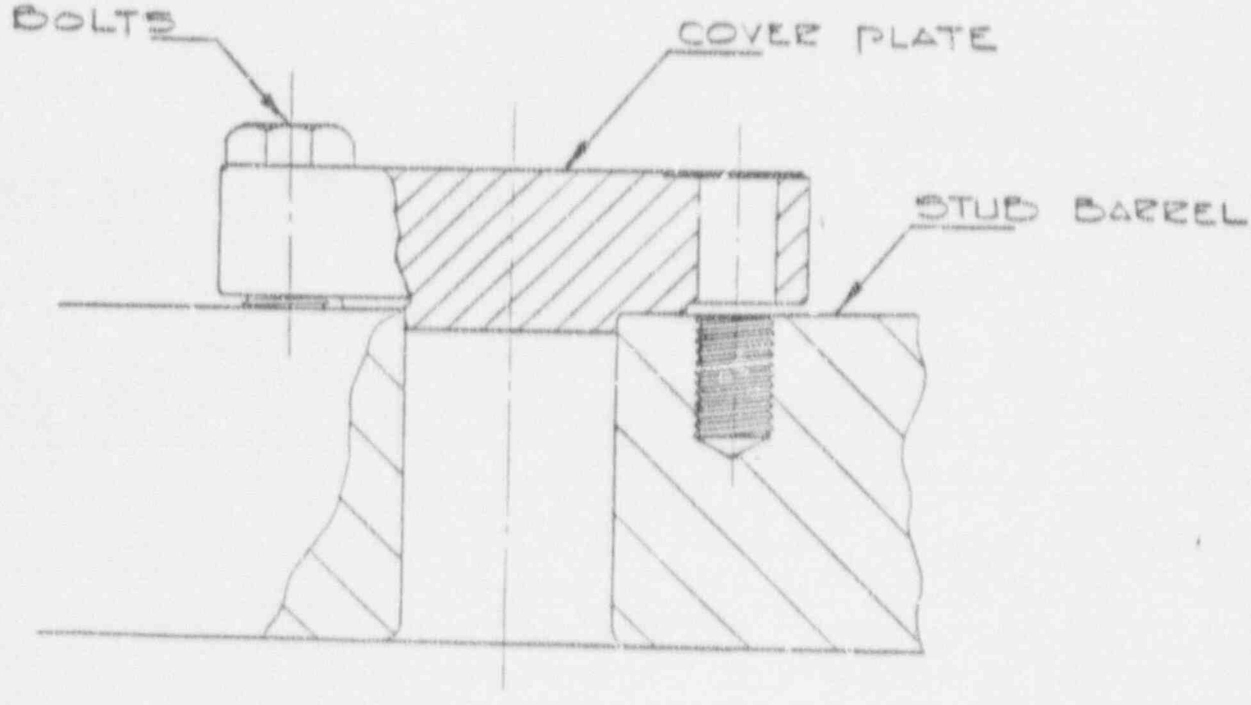
NRC MEETING
APRIL 30, 1982

ACCESS HOLE COVER





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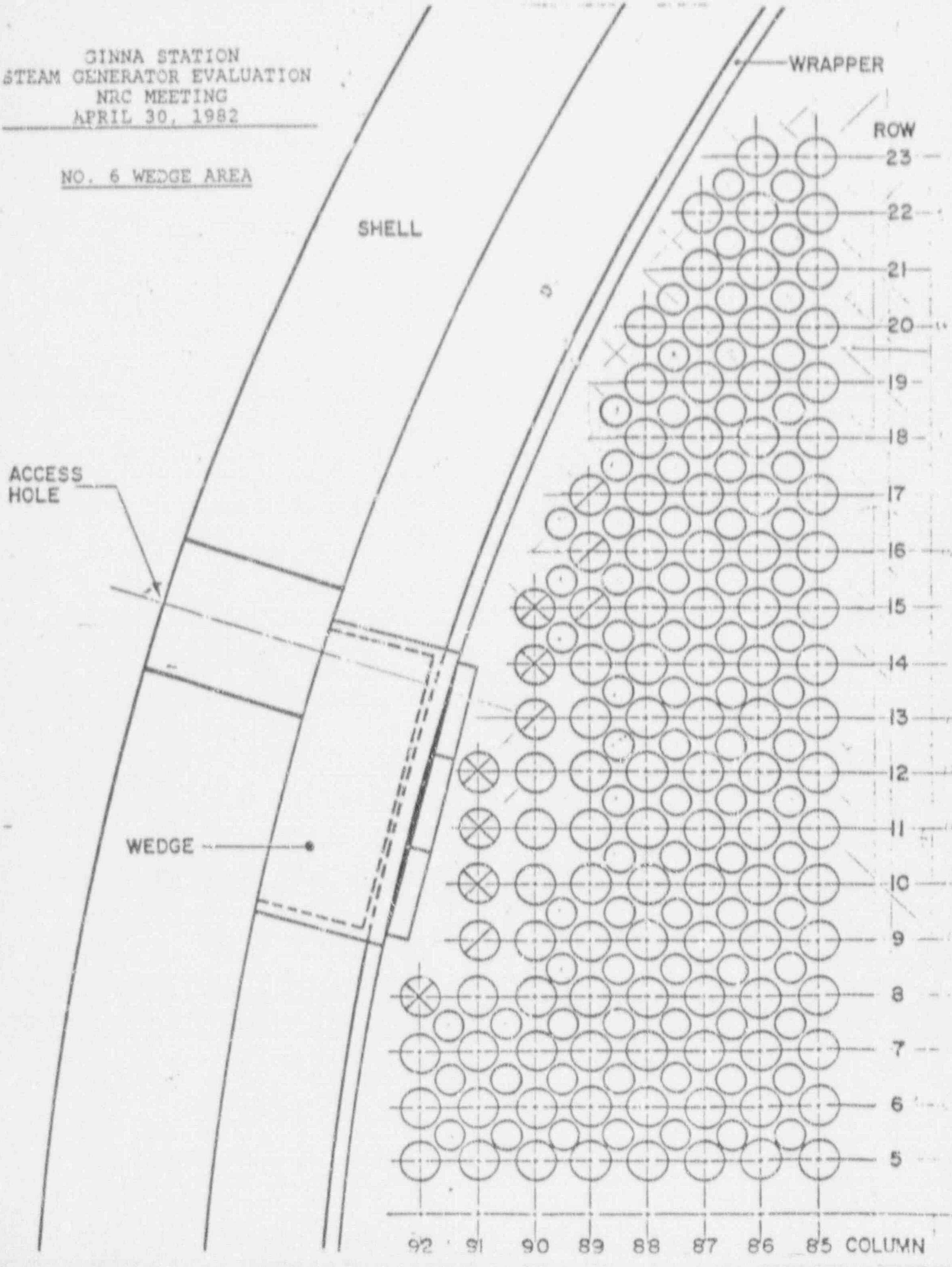
APRIL 30, 1982

NO. 6 WEDGE AREA



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NRC MEETING
APRIL 30, 1982

RADIATION EXPOSURE

o	PREPARATION	40 Man rem
o	PRE-REPAIR INSPECTIONS	80
o	ACCESS HOLES	30
o	TUBE REMOVAL	35
o	TUBE PULL	50
o	LOOSE PARTS REMOVAL	40
o	MECHANICAL PLUG REMOVAL	20
o	POST-REPAIR INSPECTIONS	25
o	CLOSEOUT	15
	TOTAL	335 Man rem

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APRIL 30, 1992

TECHNICAL BASIS REPAIR PROGRAM

- ACCESS PORTS
- THERMAL/HYDRAULIC EVALUATION
- STRUCTURAL EVALUATION
  - FATIGUE
  - COLLAPSE
  - GEOMETRIC STABILITY

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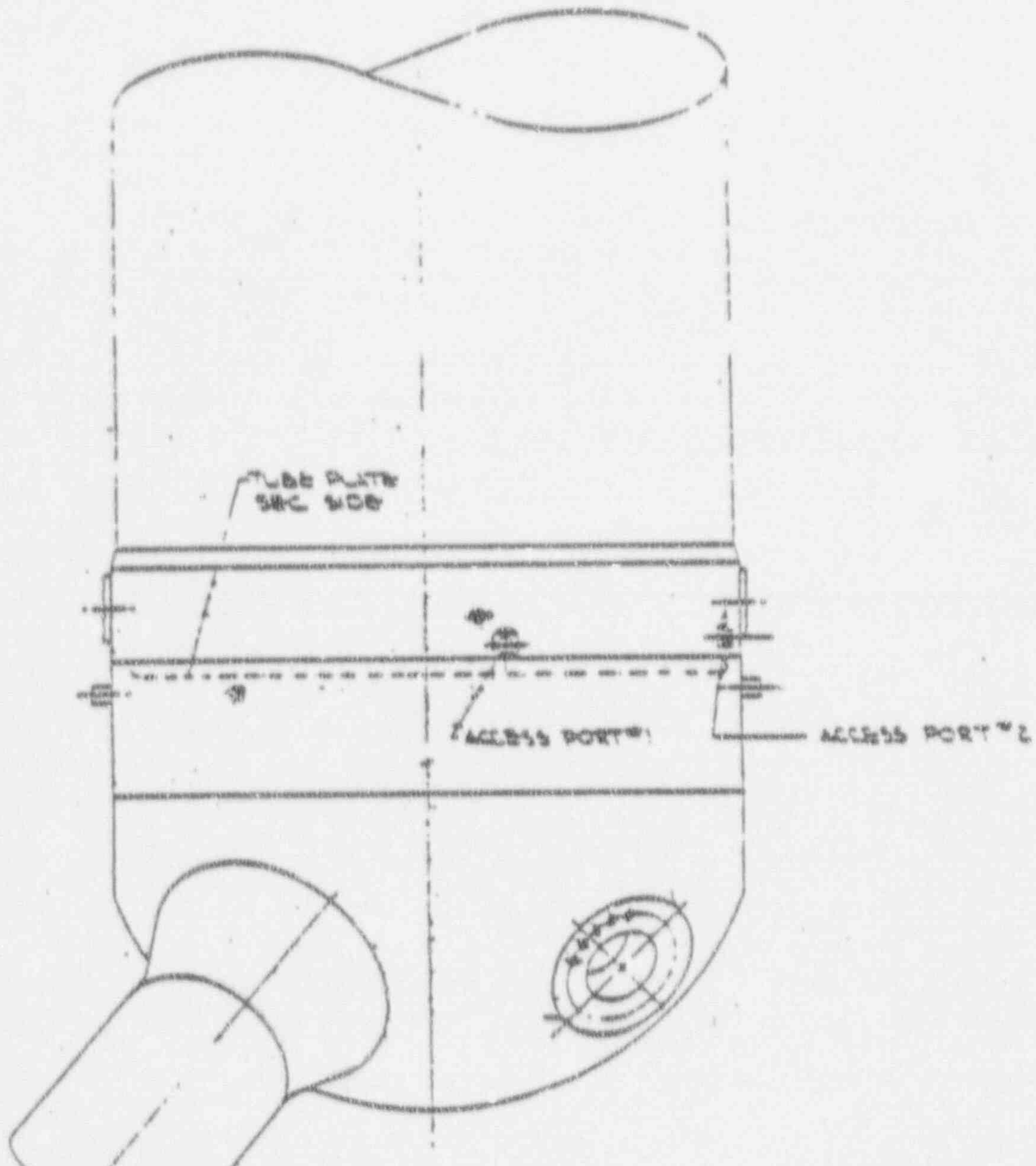
APRIL 30, 1982



Location of Access Ports

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NRC MEETING
APRIL 30, 1982

STRESS SUMMARY FOR 3 INCH DIAMETER ACCESS PORT

	Ratio of Maximum Stress to Allowable Stress		
Load Condition	Bolt	Cover	Shell
Design	0.63(1)	0.29(3)	— (6)
Normal and Abnormal	0.65(1)	<1.0	1.01(5)
	0.98(2)		
Test	0.65(1)	0.29(3)	— (6)
	0.62(2)		
Fatigue Usage Factor	0.85(4)	0.00	0.16
Bolt Replacement Interval	8 years	—	—

Notes: (1) Average Service Stress

(2) Maximum Service Stress

(3) Primary Membrane Plus Bending

(4) Fatigue usage factor based on specified replacement interval

(5) Acceptable per Code. A simplified elastic - plastic analysis was invoked for the fatigue evaluation.

(6) Primary stress limits are satisfied by Code rules for opening not requiring reinforcement.

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APRIL 30, 1982

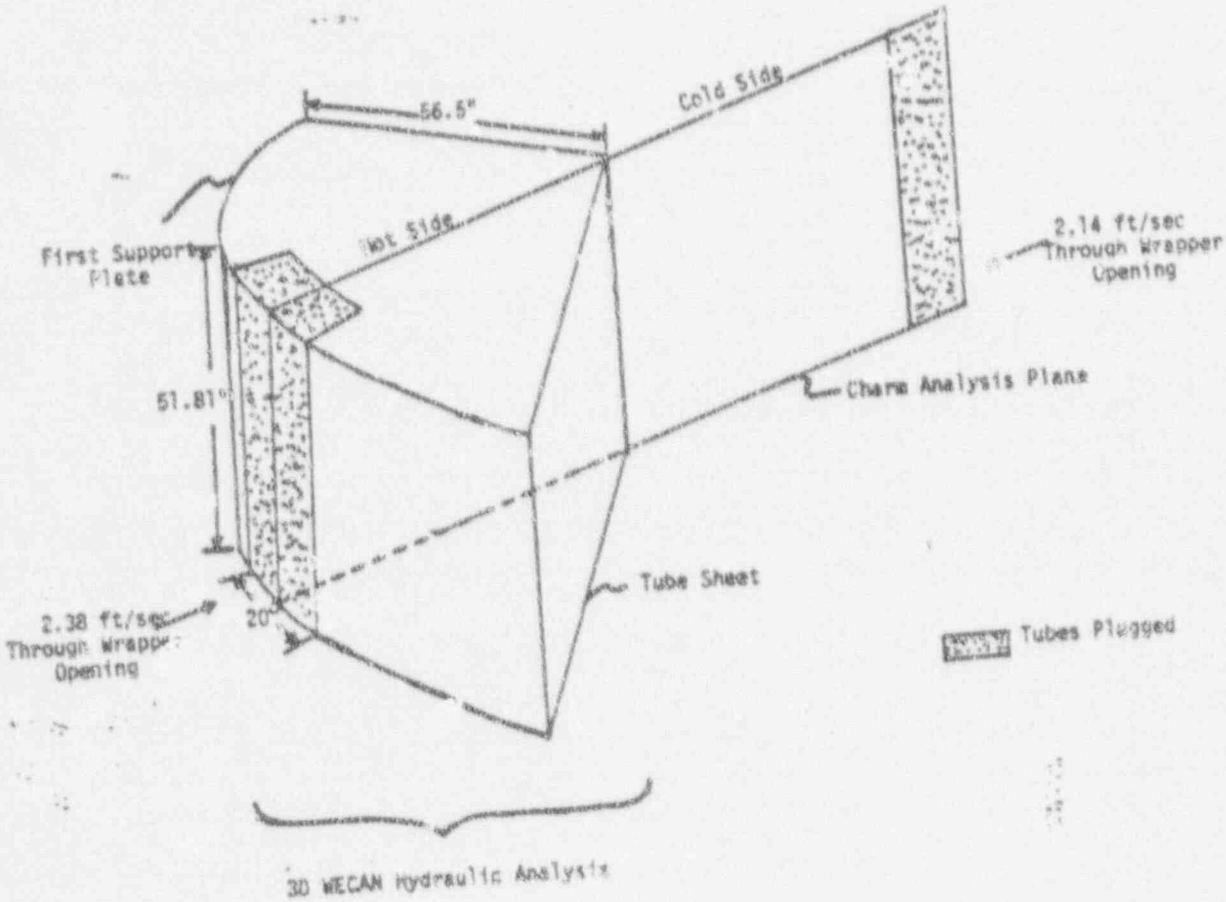
THERMAL HYDRAULIC EVALUATION

PURPOSE

DETERMINE FLUID VELOCITIES FOR USE IN EVALUATING FLUID-INDUCED LOADS

METHODS

CHARM ANALYSIS
WECAN ANALYSIS



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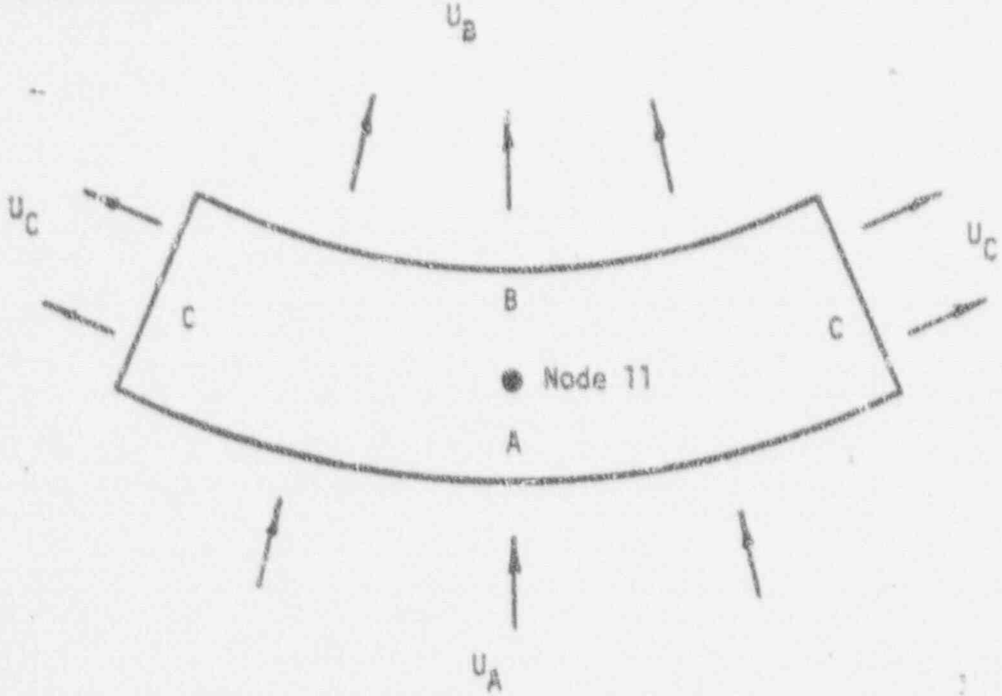
APRIL 30, 1982

BETWEEN-TUBE CROSSFLOW VELOCITIES IN AND NEAR THE TUBES REMOVED REGION

Location	Crossflow Velocity (nominal)	Crossflow Velocity (1 block of tubes removed)	Crossflow Velocity 2 blocks of tubes removed)
Perimeter Cell (Face A)	8.2 ft/sec	9.13 ft/sec	9.13 ft/sec
Perimeter Cell (Face C)	8.2 ft/sec	9.13 ft/sec	9.13 ft/sec
Perimeter Cell (Face B)	9.01 ft/sec	10.12 ft/sec	10.33 ft/sec
One Cell in from Perimeter (Face B)	8.54 ft/sec	9.48 ft/sec	10.95 ft/sec
Two Cells in from Perimeter (Face B)	8.21 ft/sec	9.16 ft/sec	10.46 ft/sec



MAXIMUM TUBE GAP VELOCITIES WITH AND WITHOUT TUBE REMOVAL

Case	Maximum Gap Velocity, ft/sec
Nominal	9.01
One block of tubes removed	10.12
Two block of tubes removed	10.95

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APRIL 30, 1982

SUMMARY OF TURBULENCE ANALYSES

- FIXED-FIXED BOUNDARIES
- CROSS-FLOW VELOCITY, 10.0 FT/SEC
- DAMPING RATIO, 0.01

CROSS SECTION OF DISTORTED ZONE	VIBRATION AMPLITUDES, MILS* TURBULENCE
CYLINDER (NOMINAL)	0.81
10 PERCENT OVALITY	0.83
KIDNEY	0.83

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FATIGUE

THERMAL/MECHANICAL LOADS

NOMINAL PLUGGED TUBE	.036 USAGE $<$ 1.0
NOMINAL PLUGGED TUBE WITH NOTCH OR STRESS RISER	.676 USAGE $<$ 1.0

HYDRAULIC LOADS

STABILITY	10.12 FT/SEC $<$ 14 FT/SEC
CROSS FLOW TURBULENCE (10 FT/SEC, NOTCH, F-P)	±11.24 KSI $<$ 13 KSI AT 10^{11}

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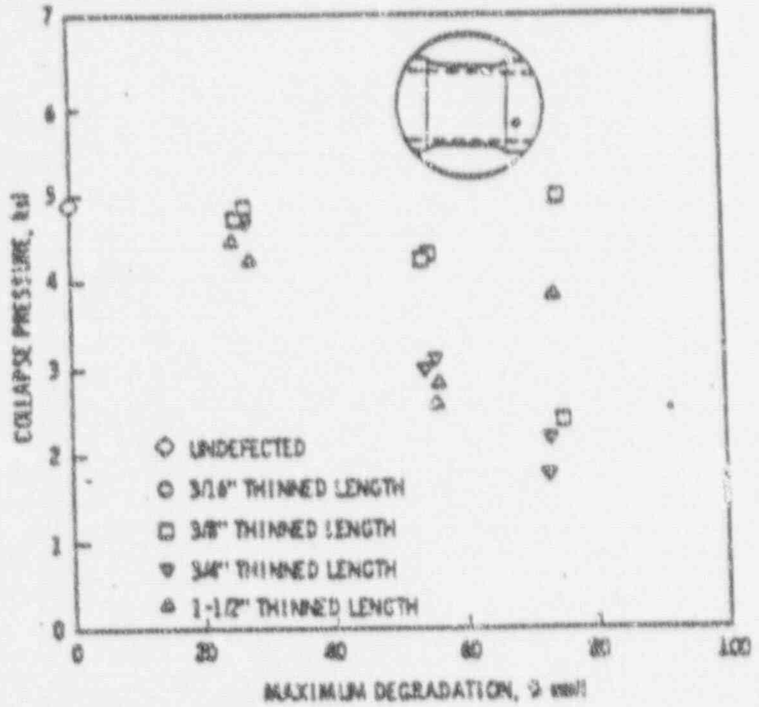
APRIL 30, 1982

COLLAPSE



COLLAPSE PRESSURE OF UNIFORMLY THINNED TUBING

- TESTING RESULTS
  - -COLLAPSE STRENGTH NOMINAL TUBE 5000 PSI
  - -80% UNIFORM THINNING FOR COLLAPSE AT 1020 PSI
- FOR A GIVEN % WALL DEGRADATION COLLAPSE PRESSURE FOR LOCALIZED THINNING WILL BE GREATER THAN FOR UNIFORM THINNING
- TUBES IN STEAM GENERATOR HAVE PASSED A PROOF COLLAPSE TEST

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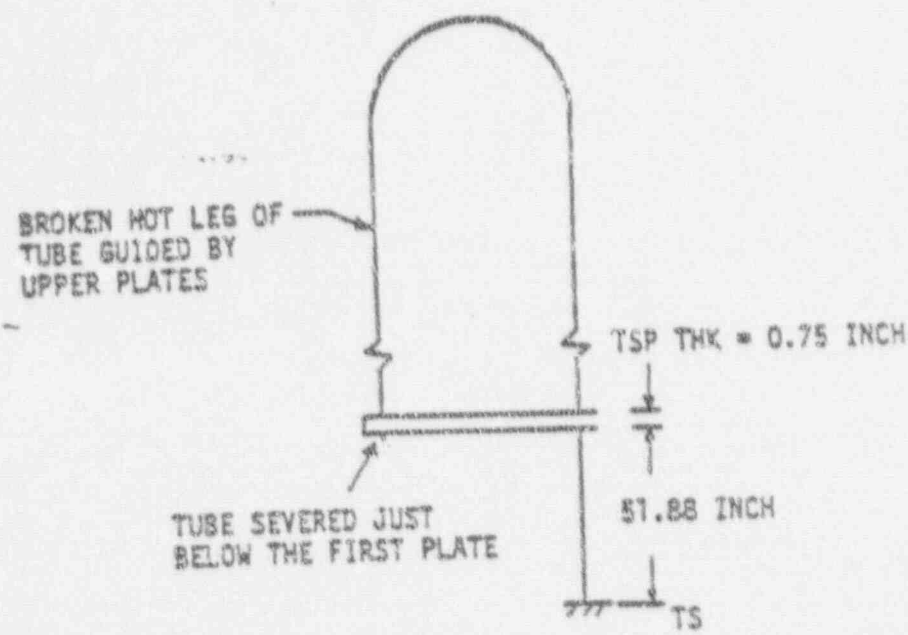
GEOMETRIC STABILITY

PRESSURE

THERMAL

LATERAL TUBE DEFLECTION



Schematic of a Partial Tube

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SUMMARY

- CAUSE OF TUBE RUPTURE
- WORK IN PROGRESS
- LOOSE PARTS MONITORING SYSTEM
- ADEQUACY OF REPAIRS
- INTERMEDIATE OUTAGE

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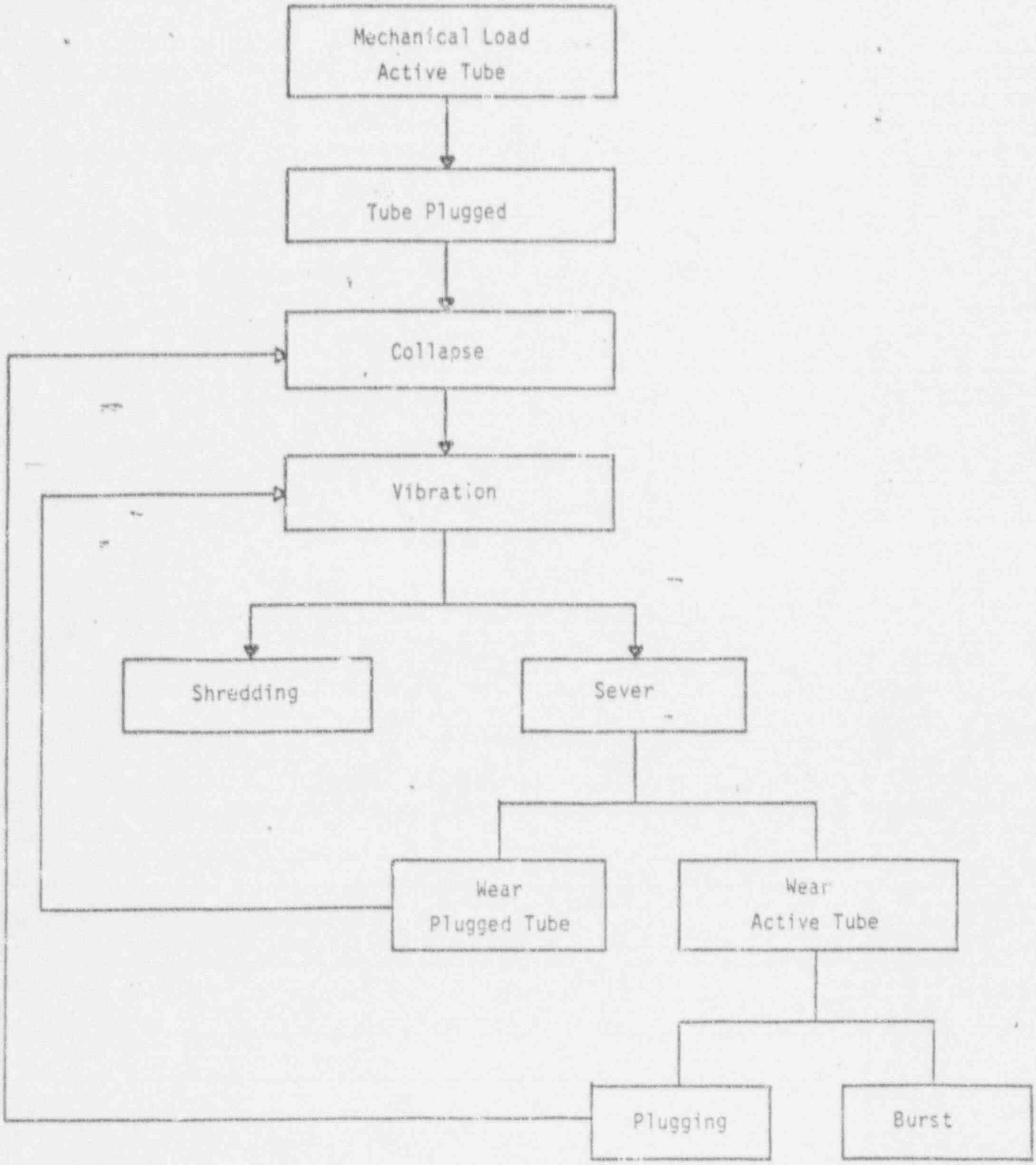
POSTULATED FAILURE MECHANISM SEQUENCE



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LOOSE PARTS MONITOR SENSOR LOCATIONS