UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 15 1979

DOCKET NOS. 50-329
AND 50-330

APPLICANT: CONSUMERS POWER COMPANY

FACILITY: MIDLAND PLANT, UNITS 1 & 2

SUBJECT: SUMMARY OF APRIL 19-20, 1979 MEETINGS ON OPEN ITEMS REGARDING FSAR REVIEW

On April 19 and 20, 1979, the NRC staff met in Bethesda, Maryland with representatives of Consumers Power Company, Bechtel Associates, and Babcock and Wilcox to discuss open items associated with the staff's review of the FSAR for Midland Plant, Units 1 and 2. These open items are identified in the staff's letter of March 30, 1979 and served as the agenda for the meeting.

This is the second of two sets of meetings which were held to discuss the March 30, 1979 letter. The first set of meetings was held on April 10 and April 11, 1979. Further meetings to discuss items of the remaining branches will be scheduled.

Meeting attendees for April 19, 1979 are listed in Enclosure 1, and for April 20, 1979 in Enclosure 2.

The discussions in Enclosures 3 and 4 for this meeting summary correspond to the same-numbered items by technical review branches as specified in the March 30, 1979 letter.

D Hood

D. Hood, Project Manager
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
As Stated

ccs w/enclosures:
See next page

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ENCLOSURE 1

ATTENDANCE LIST
APRIL 19, 1979

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M. Bolotsky
H. Conrad
M. Boyle
R. Hermann
R. Stevens
B. D. Liaw
J. Strossnider
J. Shapaker

B&W COMPANY

C. E. Mahaney
L. M. Lesniak
J. Howard
J. Shively
S. Skaar
J. Pastor

CONSUMERS POWER COMPANY

J. Zabritski
E. Olfier
J. Webb
A. Birkle

BECHTEL

J. Kovach
M. A. Gerding
D. Messing
J. Clements

ENCLOSURE 2

ATTENDANCE LIST
APRIL 20, 1979

NRC - STAFF

D. Hood
B. Bosnak
S. Hou
R. Stephens
D. Pickett
C. Tinkler
J. Shapaker

B&W COMPANY

C. Mahaney
D. Howell
J. Howard
J. Shively
J. Anderton
G. Vames
J. Galford

CONSUMERS POWER COMPANY

J. Zabritski

BECHTEL

J. Clements
J. Lewis
J. Dunne
J. Cartwright

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ENCLOSURE 3

ITEMS DISCUSSED APRIL 19, 1979

Instrumentation and Controls Systems Branch

The technical reviewers explained that the open items are based upon review through FSAR Revision 15 as indicated in the staff's letter of March 22, 1979. The review of the Midland FSAR by this branch will resume in late July 1979 once prior workload commitments on other projects are completed, and assuming sufficient information for an efficient review has been submitted on the Midland docket at that time.

1 & 5 - Unresolved - The staff stated that BTP-ICSB-23 is replaced by R. G. 1.97 and that Midland is subject to the results of Generic Technical Activity A-34. The staff reiterated its position that the post-accident monitoring recorders be qualified to operate after the SSE. The applicant referenced a January 31, 1979 letter from R. DeYoung to J. Ward of AIF as the basis for its position that Position C1 of R. G. 1.97 is not applicable to Midland and also noted that no equipment on the market can be purchased which would meet Position C3. The staff cautioned that the January 30, 1979 letter predated the TMI-2 accident and that this matter is being examined again in light of such recent developments. The applicant stated it has formed its own task force to review the TMI-2 accident for implication to Midland 1 & 2. The Midland task force will include the adequacy of post-accident monitoring instrumentation and R. G. 1.97.

2 - Unresolved - Additional information regarding the ultrasonic reverse flow monitors will be submitted in July or August 1979.

3 - The staff is reviewing the response in FSAR, Revision 18.

4 - Unresolved - The reply will be submitted in May 1979.

6 - Unresolved - One specific example for which two way communication devices have affected nuclear power plant operation is given in I&E Bulletin 76-03 dated 3/15/76.

7 - Unresolved - The staff reiterated its position that the Midland design be revised to conform with BTP-ICSB #4.

8 - The staff's formal detailed drawing review will not begin until at least July 1979.

9 - Unresolved - The applicant will reply by FSAR Revision 20.

10 - Unresolved - This concern is based upon I&E Circular 78-19 and the term "bypass" is defined as used therein.

11. - Unresolved - Reply will be in FSAR Revision 20.

12. - Unresolved - Reply will be in FSAR Revision 21.

Power Systems Branch

1. - Unresolved - The applicant clarified that the reliability calculation presented in FSAR Revision 19 is for the whole ESFAS, not the sequencer alone. Further discussion of this item is needed.

2. - Unresolved - The applicant will reply by FSAR Revision 20.

3. - Not discussed.

4. - Further information will be submitted by FSAR Revision 20.

5. - The staff is reviewing the applicant's response in FSAR Revision 19.

6. - Unresolved - The staff requested worst case examples involving the maximum use of jumpers and identification of other equipment involved.

7. - Unresolved - The staff is reviewing the applicant's response and notes that Position No. 4 therein is not completed. An additional concern resulting from Arkansas Nuclear One, Unit 2 was described.

8a. - The applicant intends to indicate compliance with the staff position by FSAR Revision 20 and to provide supporting analyses by FSAR Revision 21.

8b. - Unresolved - The staff stated that one means of testing fuses is by measuring resistance on a sampling basis. Fuses which are installed are to be tested.

9. - The applicant will clarify the FSAR to indicate that each diesel is provided with a 7-day fuel oil storage tank to meet the engineered safety feature loads. The 3 1/2 day supply was based upon maximum rated diesel load.

Materials Branch (Metallurgy Section)

1. - Unresolved - The staff requires more specific information. In each thickness of liner and for each penetration for which fracture toughness test data is required, the applicant was requested to provide the following: the material and its specification, the applicable ASME code edition and addenda, the service metal temperature, the testing requirements, the acceptance criteria, and the actual fracture toughness test data. The staff noted that the information in Table 3.8-25 was insufficient in this regard.

Extra Item - The staff needs information on the hydrogen evaluation rates inside containment after the design basis accident for the full range of pH conditions. The staff requested data on pH as functions of time and temperature for the first 2 hours during which pH may be changing. The staff referred the applicant to BNL-NUREG-24532 "Hydrogen Release Rates from Corrosion of Zinc and Aluminum," dated May 1978. The applicant replied that it is participating in a study with Oak Ridge for the hydrazine spray additive, including effects on coatings, and will submit a summary about June 1979.

Additional Item - The applicant will clarify the use of stainless steel in the steam generator. Such use was said to be limited to the cladding.

Materials Branch (Materials Integrity Section)

1. - Unresolved - The applicant will upgrade its inspection plan to the applicable code edition, once approved.

2 & 3 - Unresolved.

Extra - The discussion also included a review of the staff requests by letter of March 26, 1979 to which the applicant will respond by a future amendment. The staff also requested a summary of the fracture toughness data for each ferritic steel component of the reactor coolant pressure boundary.

Mechanical Engineering Branch

8. - The applicant explained it can only comply with the "intent" of R. G. 1.121 because this guide was not written for the once-through steam generator design. The applicant will clarify that the criteria of the guide will be meet, but that the analyses performed must be modified as appropriate to the once-through design. The staff noted that B&W has not provided a report on tube plugging.

Related Item - The staff noted that this item (MEB, Item 8) was also related to AAB Item 3 regarding turbine missile hazard, which is made less probable by frequent testing of the turbine stop valve; in Generic Technical Activity A-5 the staff notes a concern that frequent testing of the turbine stop valves may lead to degradation of the OTSG tubes as was observed at Oconee. The staff also expressed concern that less frequent testing may only defer tube damage to a later time. The B&W representative stated that more recent tests completed at Three Mile Island Unit 2, coupled with Oconee results, show that tube degradation will not occur if turbine stop valve testing is performed at reduced load and if the valve design provides for restroking of the governor valves to close at a slow rate. The applicant has not completed its review of this B&W finding. The generic resolution of this item, once determined, will be applied to Midland by the staff.

Containment Systems Branch

5. - The applicant advised the staff that it is considering the deletion of the penetration pressurization system, including the isolation valve seal water subsystem. The applicant noted that this system was provided in the design as a result of the staff's requirement expressed in a letter from Mr. P.A. Morris dated March 28, 1969. The applicant stated that Appendix J to 10 CFR 50 did not exist in 1969 and that, in its opinion, Appendix J negates the need for this system. The applicant further noted that the staff had permitted no credit for this system. The staff stated it will further review the 1969 letter and advise the applicant further of its position.

ENCLOSURE 4

ITEMS DISCUSSED APRIL 20, 1979

Containment Systems Branch

3. - The staff noted that further information is needed regarding the nodalization sensitivity study as to the basis for which the applicant concludes the 28 node scheme for the reactor cavity subcompartment is acceptable. The FSAR states that the peak pressures converged in the 28 node model and that there was virtually no change in peak pressures in the 32 and 40 mode models. The applicant also indicated that the loads and moments acting on the vessel also converged and will address this convergence in the FSAR.

The staff questioned why the nodes above the shield plug (i.e., nodes 11, 12, 13, 14, 25, 26, 27 & 28 of FSAR Figure 6D-7) were neglected in determining the loads and moments, and expressed concern when changes are made in nodalization between peak pressure cases and the loads and moments. It was noted that the loads and moments transients (time histories) vary between the various nodalization schemes and that it is difficult to determine the worst case. The staff concluded that the worst case would be one with the peak magnitudes. The applicant will submit a calculation of the loads and moments considering the region above the shield plug in order to justify its assumption that this region makes a negligible contribution to loads and moments.

The reactor cavity model used in the applicant's analyses assumes that insulation blocks all flow paths to the incore instrumentation tunnel. The staff noted that this assumption prevents uplift forces on the vessel in the calculations and is not conservative for this purpose. The staff requires that the applicant supplement its total asymmetric load analyses to determine uplift forces on the vessel without flow blockage by insulation.

The full cross-sectional area (1A) break in the primary cold leg piping submitted for the reactor cavity subcompartment analysis occurs in the piping penetration. The applicant's analysis assumes that the foam-glass insulation in the penetration is instantly displaced. The staff finds this assumption is very significant to the vessel calculations because it directs most of the blowdown to the steam generator subcompartment rather than into the reactor cavity. The staff also noted that inadequate discussion of the foam-glass insulation is given

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in FSAR Section 6.2.2.1.2.4, Tables 6.2-33 and-34. The applicant was asked to further describe in the FSAR the following characteristics of the foam-glass insulation: manufacturer, amount used, attachment methods, whether any sheet metal or other shroud surrounds the foam glass, and any other data relative to displacement or blocking assumptions. The applicant was also asked to address properties of the foam glass relative to the potential blocking of sump screens and compatability of the foam glass material with the post-LOCA environment (including the potential of any of the insulation materials to form a slurry which could harm pump seals, foul heat exchangers, or insulate fuel). The staff requires that the applicant either (1) revise the analyses assuming the insulation remains in place, or (2) demonstrate analytically or experimentally the instantaneous displacement of the insulation. The applicant noted the conservatism in the assumption of a full area (1A) break whereas physical restraints preclude in excess of a 0.24A break. The applicant will submit further analyses to justify the loads and moments provided.

The staff requested that the 40 node reactor cavity model of FSAR Figure 6D-9 be expanded by including a horizontal plane through the nozzle centerline. The applicant will submit loads and moments using this modification to the model. The staff also questioned the basis for the vertical plane in the nozzle region between nodes 5-30, 2-29, 16-32 and 19-31; if these boundaries are intended to represent the excore neutron detectors, then the staff notes that these vertical planes should be extended for the entire vessel length.

5. - The staff stated its position with regard to the applicants consideration to delete the penetration pressurization system which was discussed the previous day. The staff does not support deletion of the system and finds that such deletion would require re-review of site acceptability issues. The staff noted that this system provides post-accident consequence mitigating safety equipment without which the staff advised in 1969 that the proposed site was unacceptable. The subsequent issuance of Appendix J does not change the need for increased protective measures at specific sites. The staff also notes that such deletion would represent a departure from the principal architectural and engineering criteria specified in PSAR Section 1.4.4.

Auxiliary Systems Branch

2. - The staff stated that review of the information in the FSAR regarding the single failure proof hoisting mechanism for the cask handling crane is being reviewed in parallel with the generic review of the referenced topical report EDR-1. The staff noted that Tables 9.1-9 and 9.1-10 of the FSAR addresses certain positions of Regulatory Guide 1.104 as its affects the design of those portions of the auxiliary building crane that are not covered by the topical report. However, the staff also noted that position C.1.C through and including C.1.F, and positions C.3d and C.5.a of Regulatory Guide 1.104 are not addressed in the FSAR and relate to portions of the crane not covered by the topical report. The staff requested that these positions be addressed in the FSAR and that justification for any deviations or alternatives to these positions be provided.

Mechanical Engineering Branch

5. - The analysis of reactor internals for combined SSE and LOCA loads for the Midland plant, including asymmetric cavity pressurization, are based upon adjustments to a prior analysis performed for Davis-Besse 2 & 3. A B & W representative explained in greater detail how these adjustments were made. Viewgraph slides used for this presentation are shown by Enclosure 5. The Midland analyses will be submitted in March 1980.

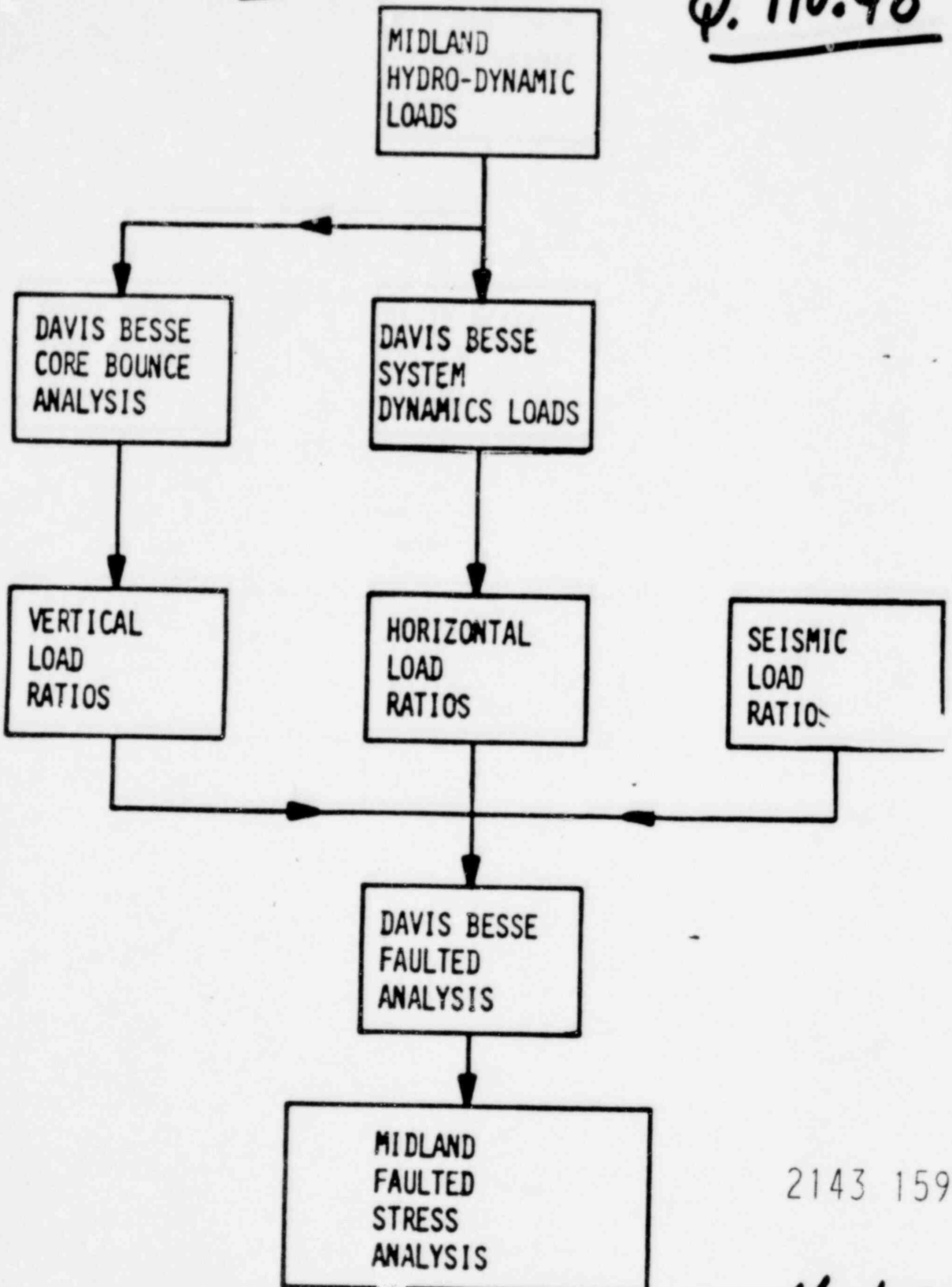
6. - Midland is the first B & W plant to perform dynamic, elastic-plastic analyses using break opening time histories for postulated guillotine breaks in the primary system. The ANSYS computer program is used to determine the guillotine break opening area and time. Pipe reaction forces are calculated using the CRAFT code. The analytical method is based upon iterating between pipe reaction forces and the pipe time history displacements and areas until area convergence is reached. Slides used for this B & W presentation are shown as Enclosure 6.

Operator Licensing Branch

1 & 2 - These items will be addressed by FSAR Revision 20.

Enclosure 5

Q. 110.48



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4/24/79

BASELINE STRESS ANALYSIS

- MOST RECENT STRESS ANALYSIS OF 177FA DESIGN
- INCLUDES ALL LOADS
    - ASYMMETRIC CAVITY PRESSURE
    - DYNAMIC SIDE LOADS ON CSA
    - SEISMIC LOADS
- INCLUDES LATEST ANALYSIS METHODS
- ASME CODE, SUBSECTION NG ANALYSIS
    - STRESS CATEGORIZATION
    - ALLOWABLE STRESS LIMITS

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4/201

MIDLANDS STRESS ANALYSIS

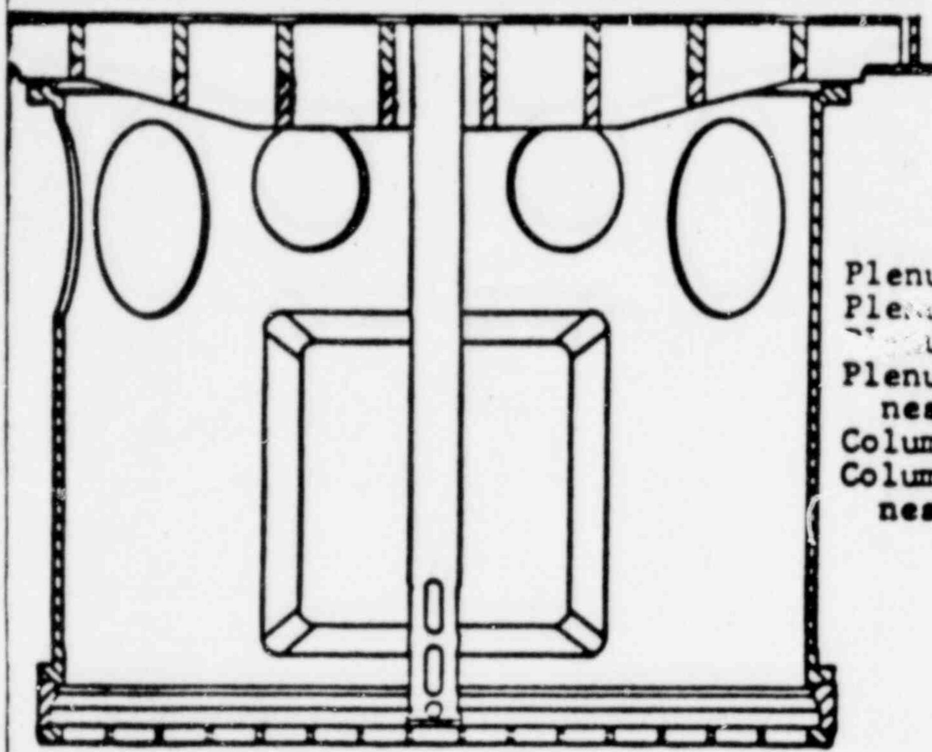
. EACH COMPONENT WILL BE ANALYZED

. EACH LOAD CAN BE RATIOED DIRECTLY

. SEISMIC LOADS WILL BE RATIOED AND ADDED

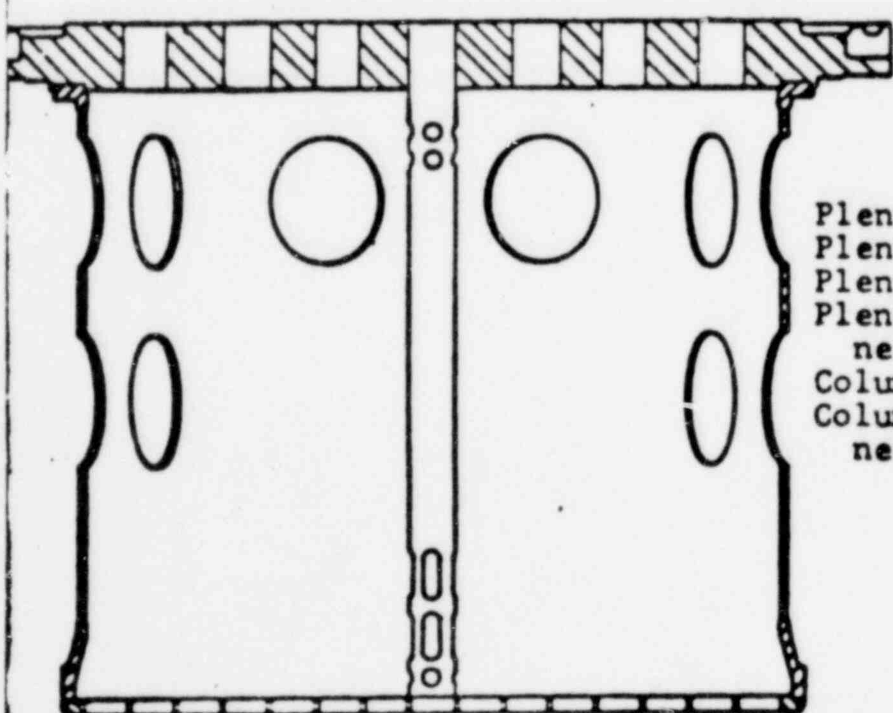
. DIFFERENCES FROM BASELINE WILL BE ADDRESSED

. RESULT WILL BE CODE SAFETY FACTORS FOR EACH COMPONENT



Plenum Cover Depth	18.00"
Plenum Cylinder Height	103.25"
Plenum Cylinder O.D.	130.00"
Plenum Cylinder Thickness	1.50"
Column Weldment O.D.	8.625"
Column Weldment Thickness	.322"

MIDLAND



Plenum Cover Depth	12.00"
Plenum Cylinder Height	114.00"
Plenum Cylinder O.D.	130.00"
Plenum Cylinder Thickness	2.00"
Column Weldment O.D.	9.10"
Column Weldment Thickness	.350"

DAVIS BESSE 2 & 3

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POOR ORIGINAL

4/20/79

TO ADDRESS NRC ITEM 110.35:

"THE APPLICANT HAS USED DYNAMIC, ELASTIC-PLASTIC ANALYSES WITH THE ANSYS COMPUTER PROGRAM TO DEVELOP BREAK OPENING TIME HISTORIES FOR ALL POSTULATED CIRCUMFERENTIAL BREAKS IN THE PRIMARY SYSTEM. THIS IS THE FIRST B&W PLANT TO PERFORM SUCH ANALYSES. WE ANTICIPATE A MEETING FOR THE APPLICANT AND B&W TO MAKE A PRESENTATION ON HOW IT PERFORMS THIS ANALYSIS."

Enclosure 6

4/20/79

Guillotine pipe break parameters
for the Midland plant

- Break opening area

And

- Break opening time

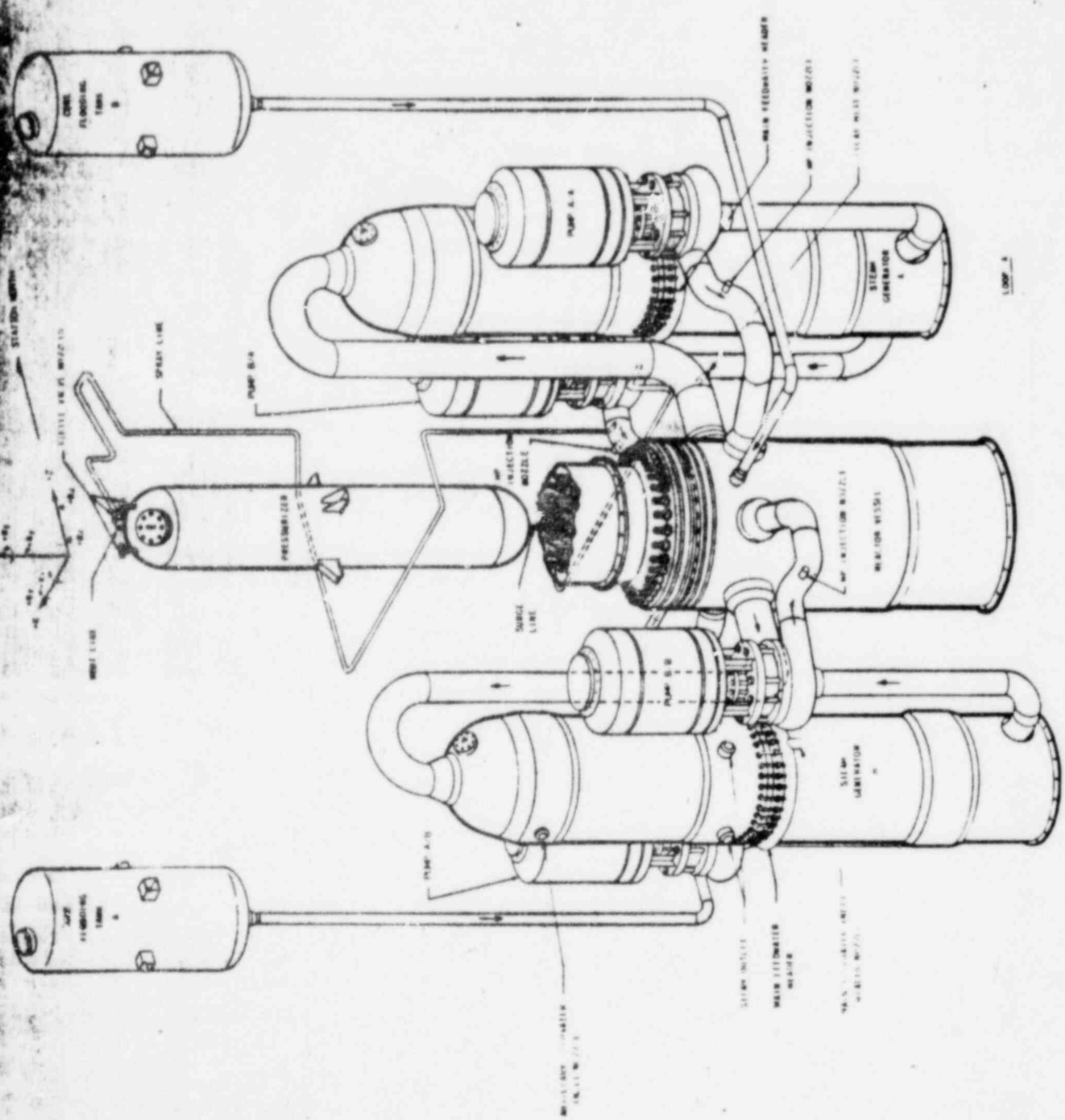
Their determination through the use of the
ANSYS COMPUTER PROGRAM

4/20/79

Regulatory guide 1.46
Footnote 11

"CIRCUMFERENTIAL BREAKS ARE PERPENDICULAR TO THE PIPE AXIS, AND THE BREAK AREA IS EQUIVALENT TO THE INTERNAL CROSS-SECTIONAL AREA OF THE RUPTURED PIPE. DYNAMIC FORCES RESULTING FROM SUCH BREAKS ARE ASSUMED TO SEPARATE THE PIPING AXIALLY AND CAUSE WHIPPING IN ANY DIRECTION NORMAL TO THE PIPE AXIS."

4/20/79



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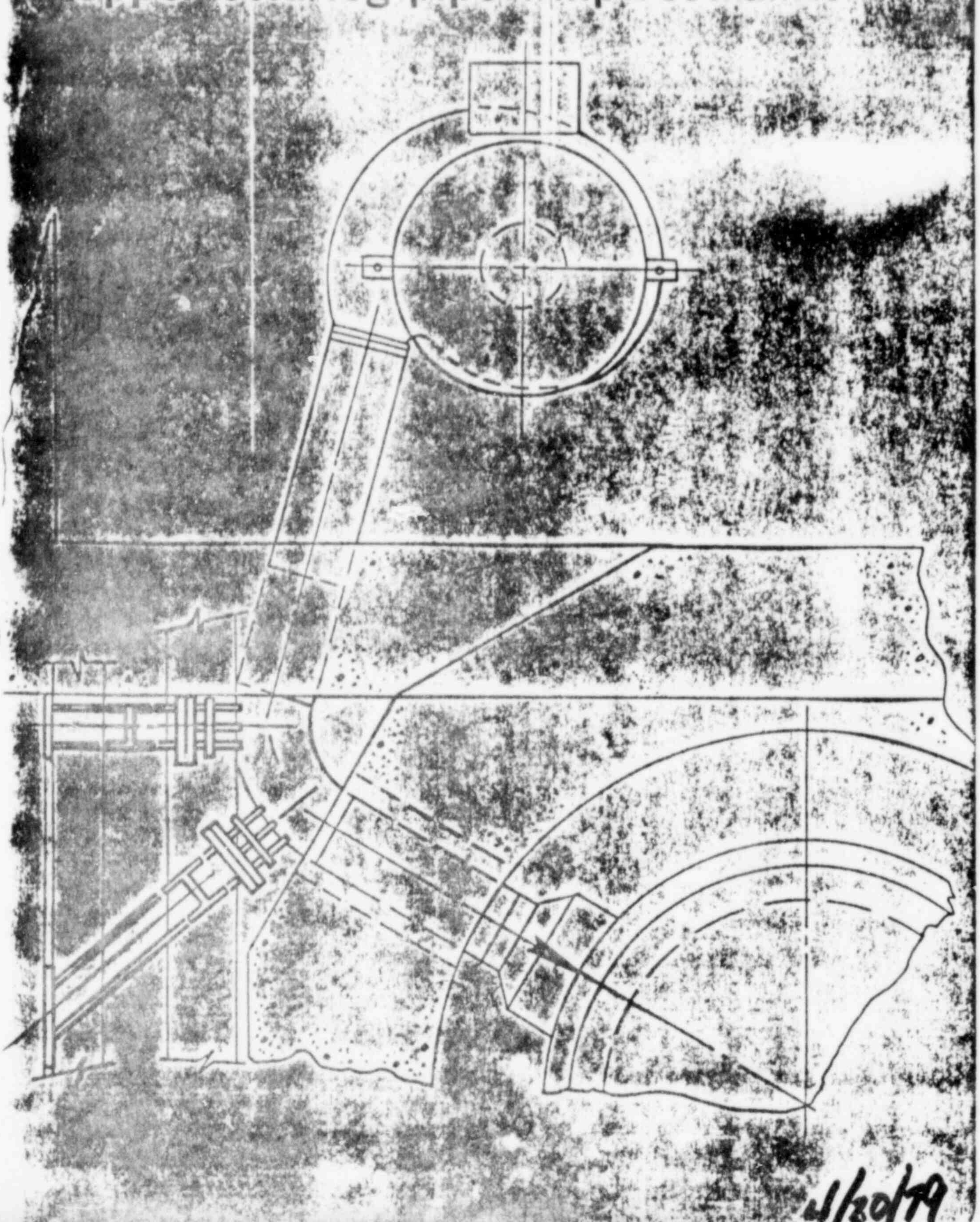
POOR ORIGINAL

Upper cold leg pipe whip restraints



4/20/79

POOR ORIGINAL

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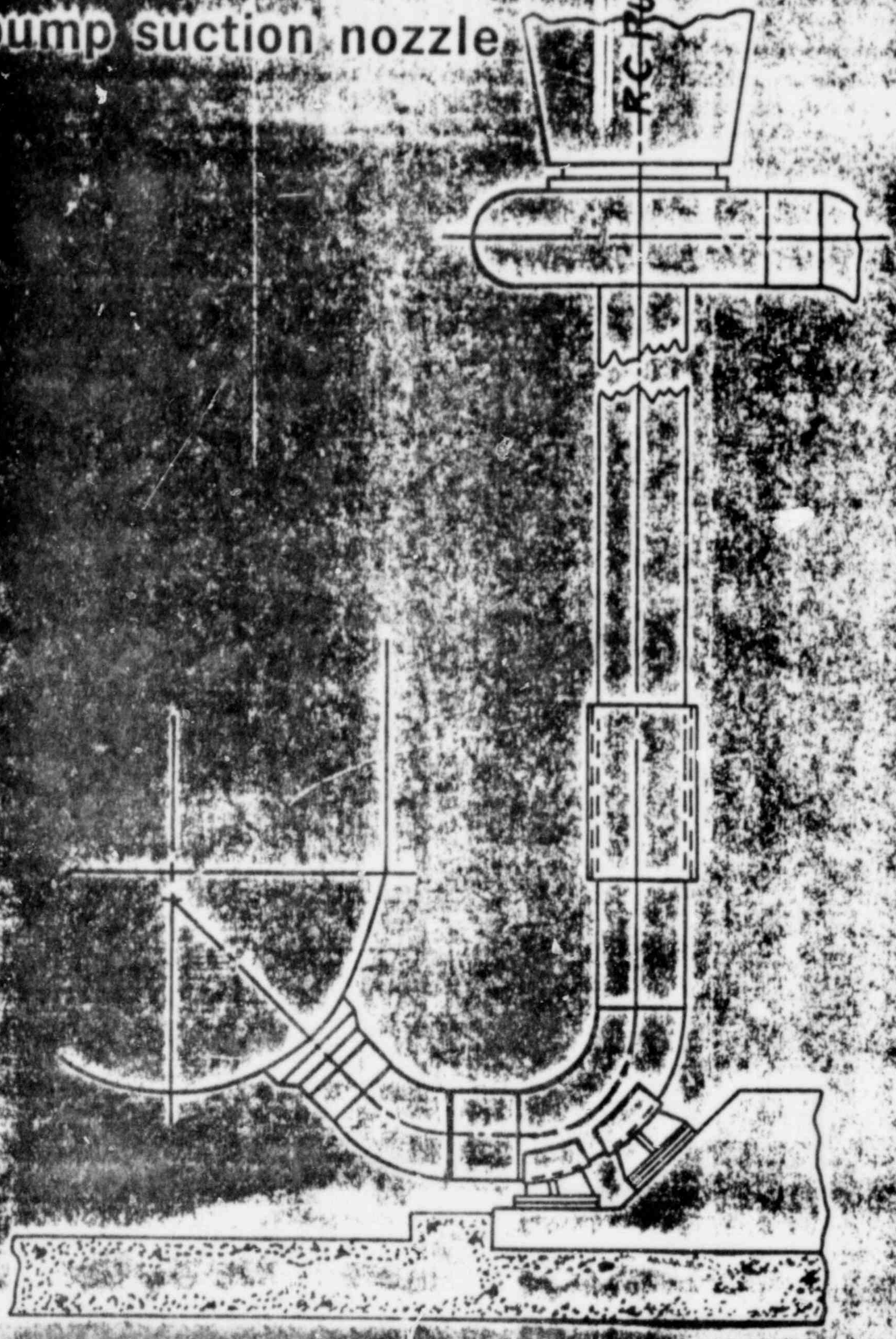
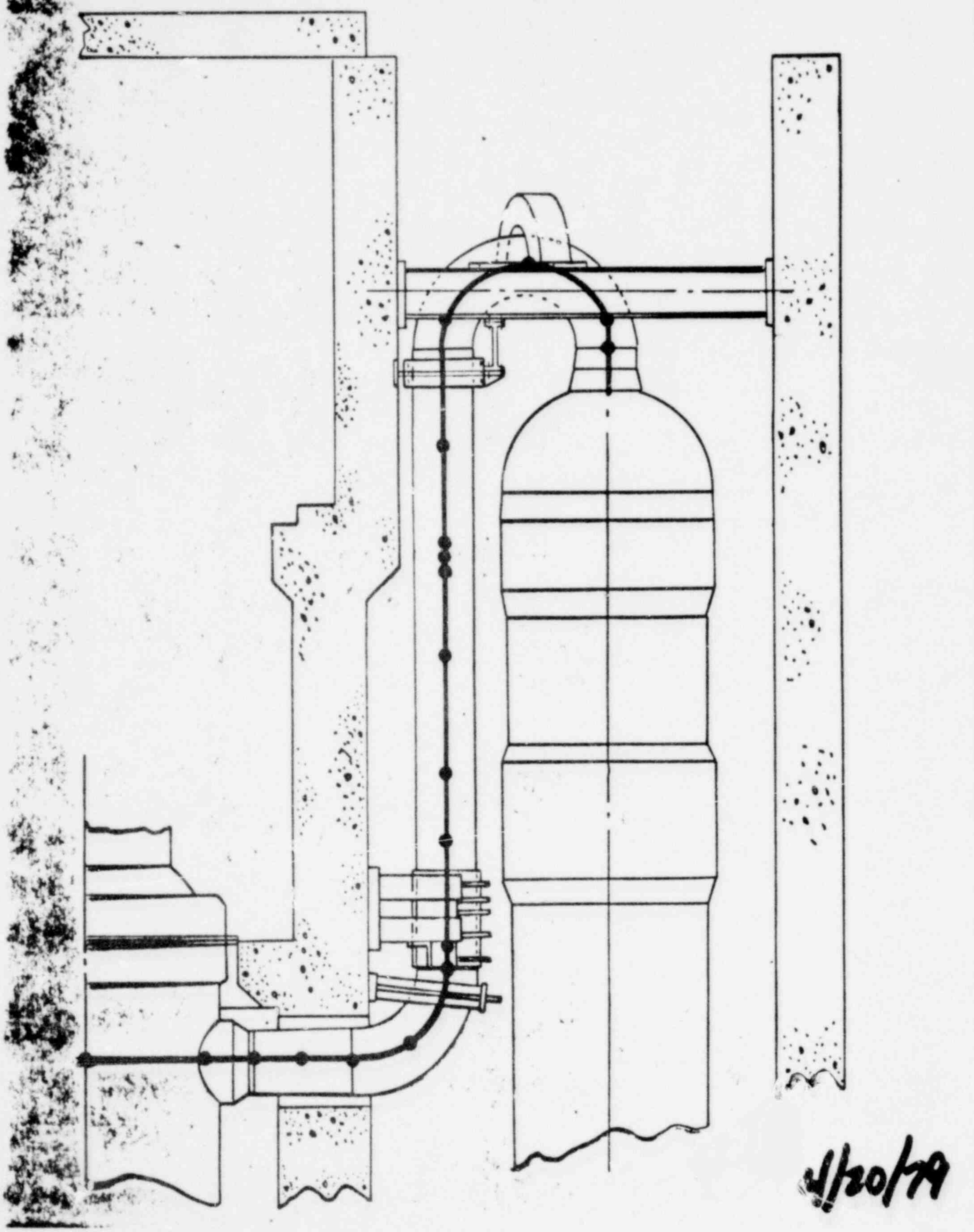
Guillotine break at pump suction nozzle



Math model



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POOR ORIGINAL

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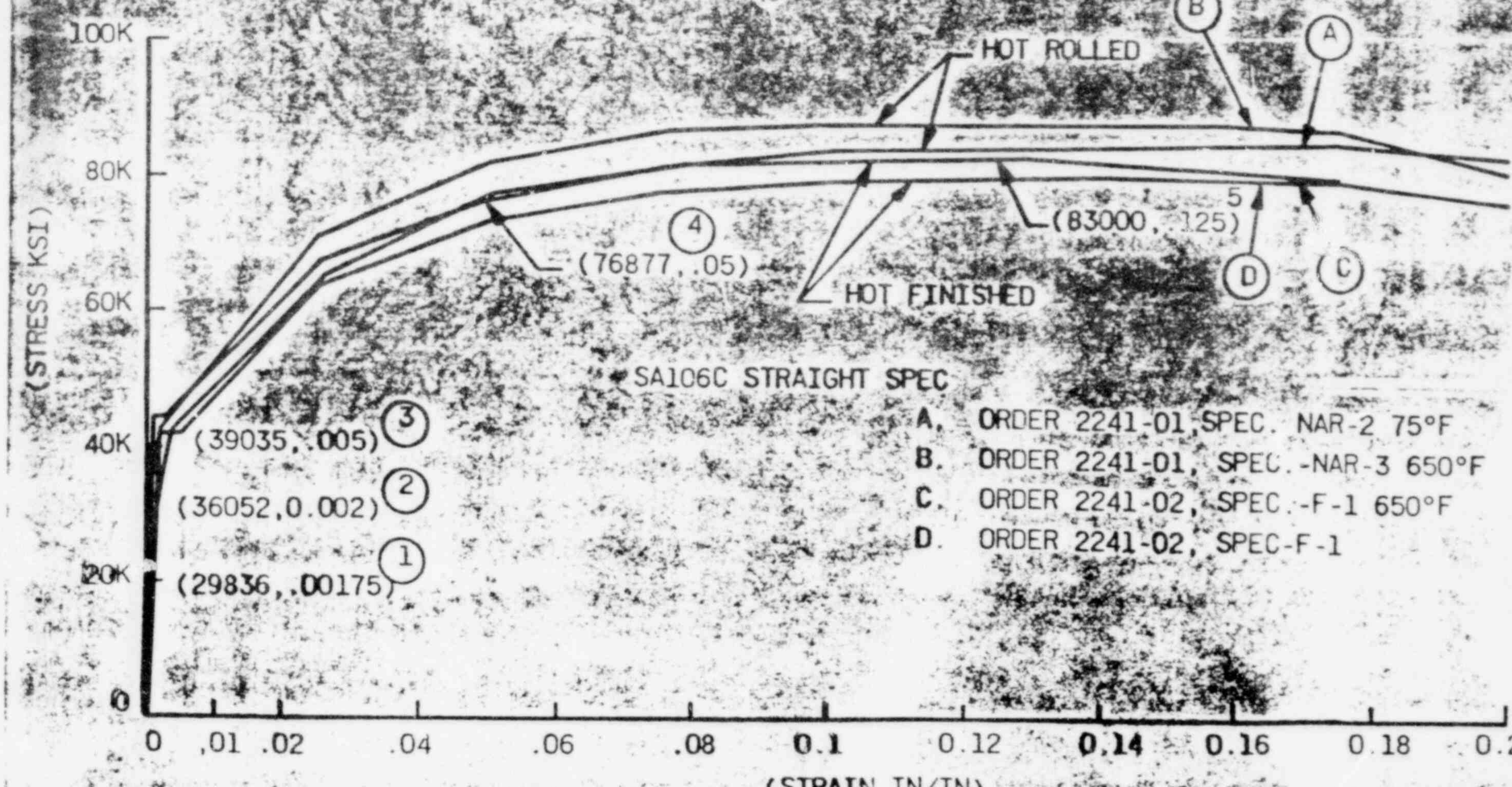
POOR ORIGINAL

Typical stress-strain curve for SA 106G1C



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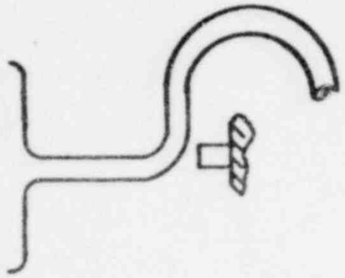
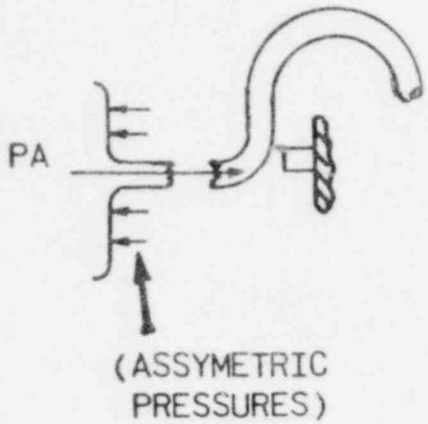
POOR ORIGINAL

Iteration procedure



INITIAL CONDITIONS

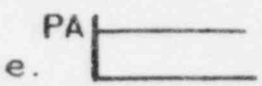
- PIPE INTACT
- PLANT AT STEADY STATE FLOW, TEMPERATURE, AND PRESSURE



INITIAL BREAK

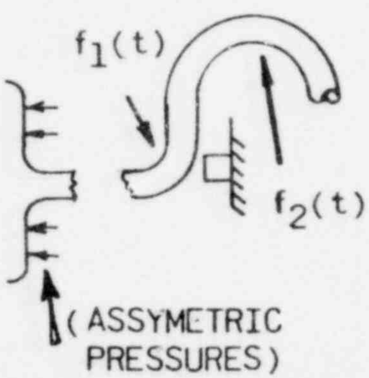
- TRANSIENT EVENT STARTS FROM ABOVE STEADY STATE INITIAL CONDITIONS
- INSTANTANEOUS PIPE SEVERANCE
- ZERO OPENING AREA ASSUMED WHICH GIVES CONSTANT PA PIPE REACTION FORCES

i.e.



- PIPE TIME-HISTORY DISPLACEMENTS & AREA DETERMINED
- PIPE REACTION FORCES CALCULATED USING DETERMINED AREA

(CRAFT CODE



ITERATION PHASE

- INSTANTANEOUS PIPE SEVERANCE FROM STEADY STATE CONDITION
- CALCULATED PIPE REACTION FORCES APPLIED
- NEW PIPE TIME HISTORY DISPLACEMENTS & AREAS DETERMINED
- NEW PIPE REACTION FORCES DETERMINED
- PROCESS REPEATED UNTIL AREA CONVERGENCE IS REACHED

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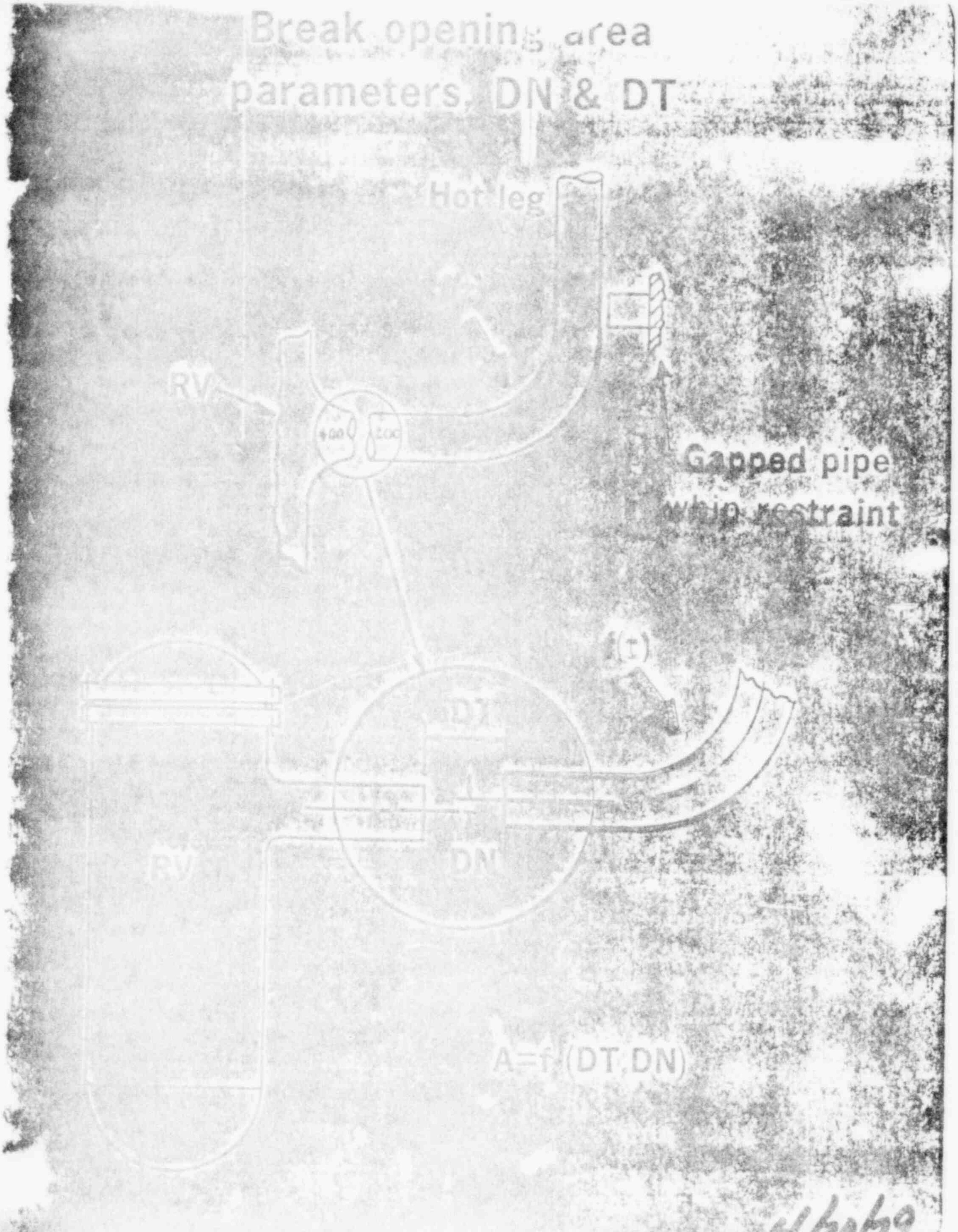
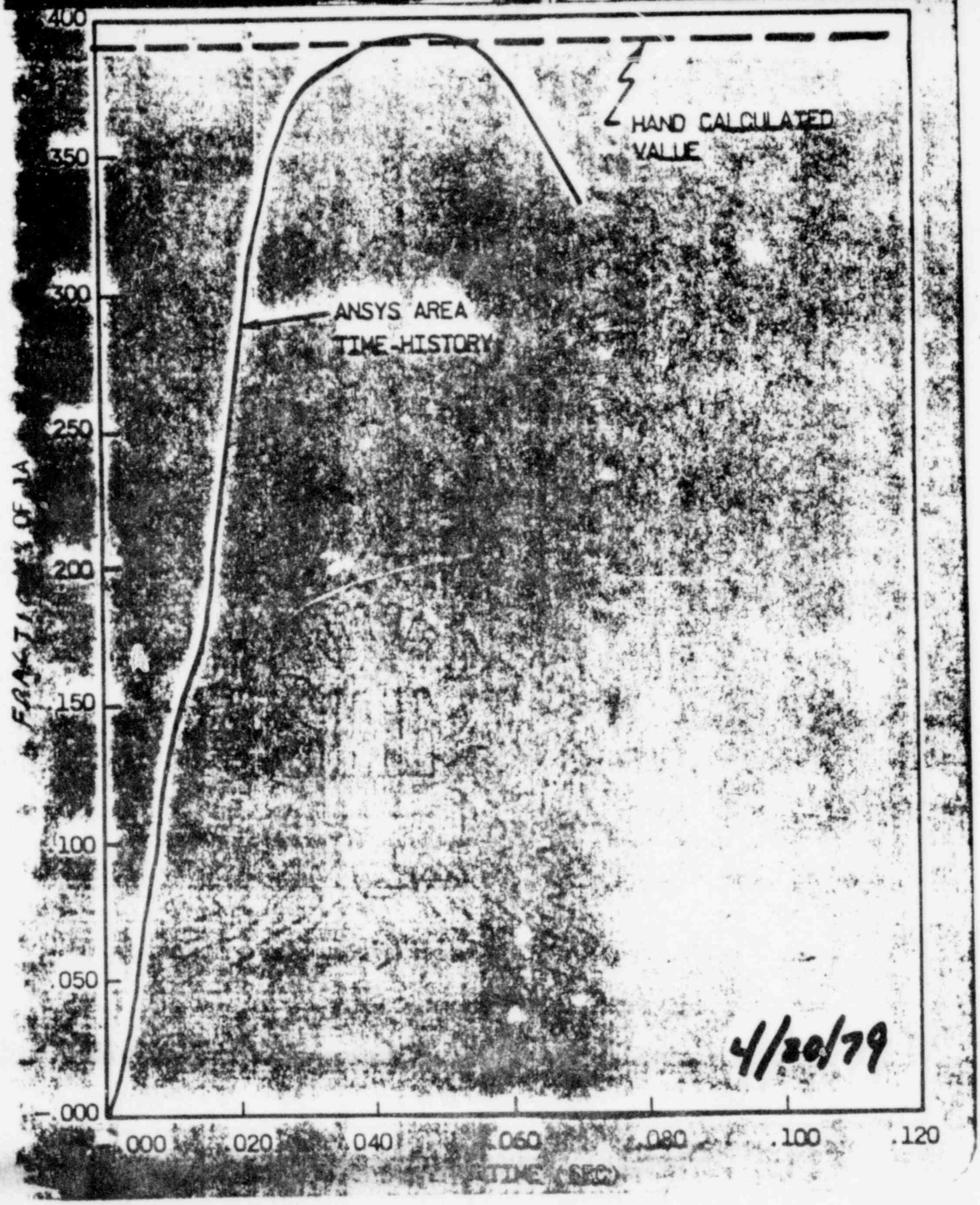


POOR ORIGINAL

Calculated break opening area at hot leg



POOR ORIGINAL

Comparison of break opening areas

LOCATION	CALCULATED BY ANSYS	HAND CALCULATIONS BASED ON GEOMETRY
• HOT LEG RV NOZZLE	0.39A	0.388A
• COLD LEG STEAM GENERATOR NOZZLE	0.15A	0.144A
• COLD LEG PUMP SUCTION NOZZLE	0.53A	0 [illegible]4A

EXAMPLE HAND CALCULATION ASSUMPTIONS (FOR PUMP SUCTION GUILLOTINE)

1. PIPE HINGED AT OTSG NOZZLE.
2. MOTION LIMITED BY GAPS.
3. NO CONSIDERATION OF PIPE YIELDING.
4. NO RESTRAINT COMPRESSION CONSIDERED.
5. ONLY PLANAR MOTION CONSIDERED.
6. NO PUMP MOTION CONSIDERED.

ANSYS CONSIDERATIONS

1. PIPE STRESS IS MONITORED THROUGHOUT FOR YIELDIN[illegible]
2. RESTRAINT COMPRESSION IS CONSIDERED.
3. EXACT GEOMETRY CONSIDERED.
4. MOTION IN ALL DIRECTIONS CONSIDERED.

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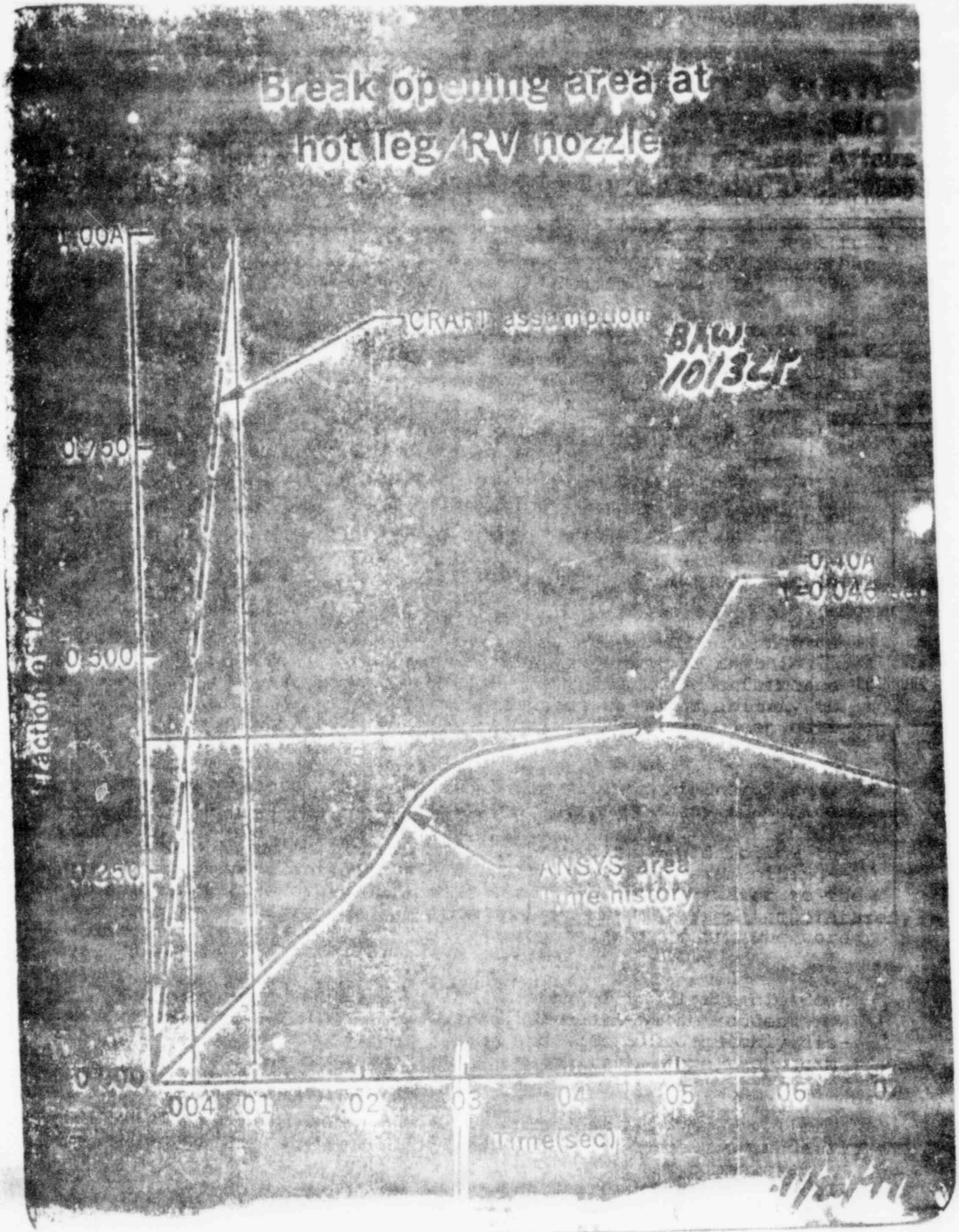

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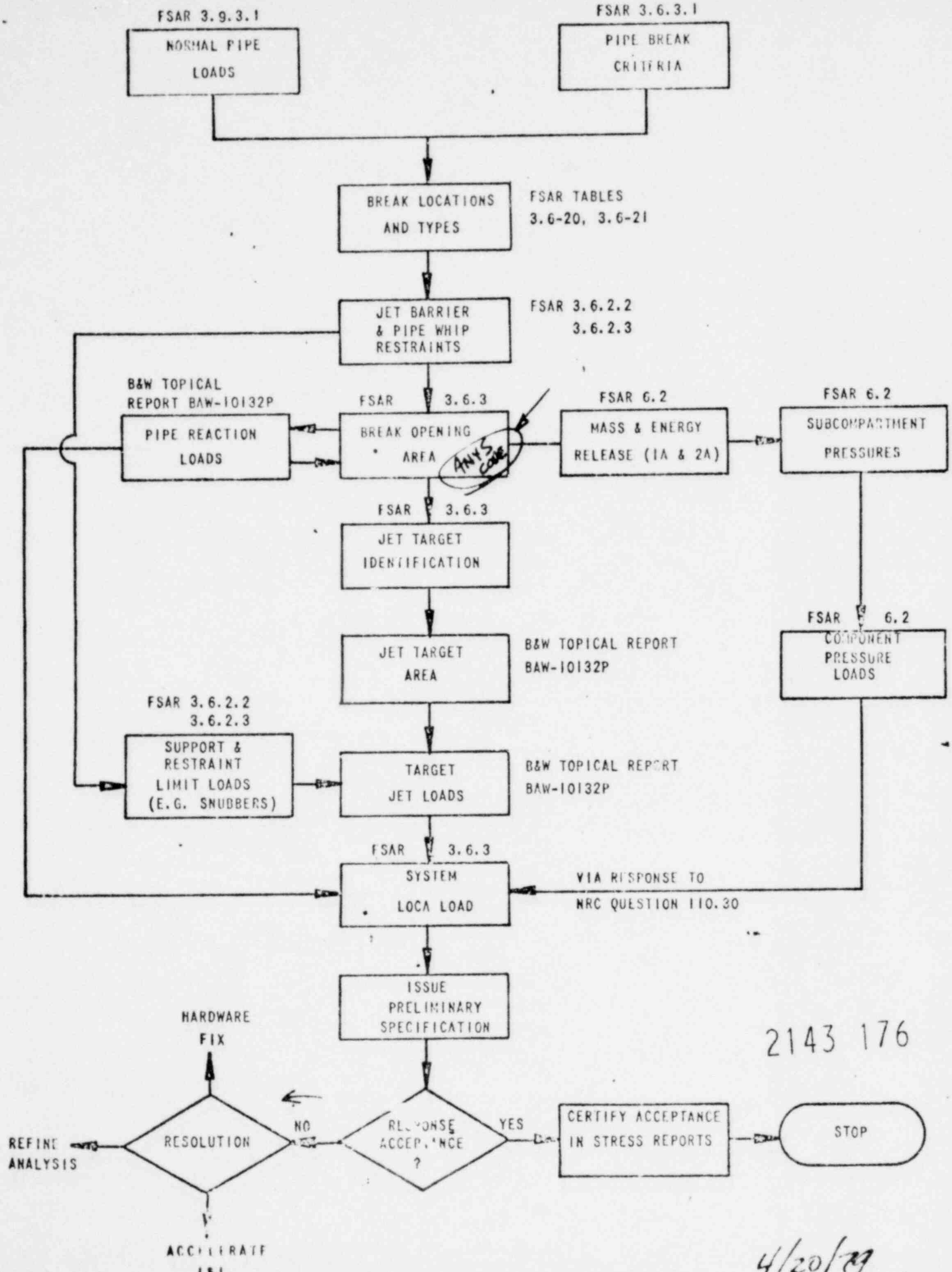

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MEETING SUMMARY DISTRIBUTION

MAY 15 1979

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Branch File #4
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J. Shapaker
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