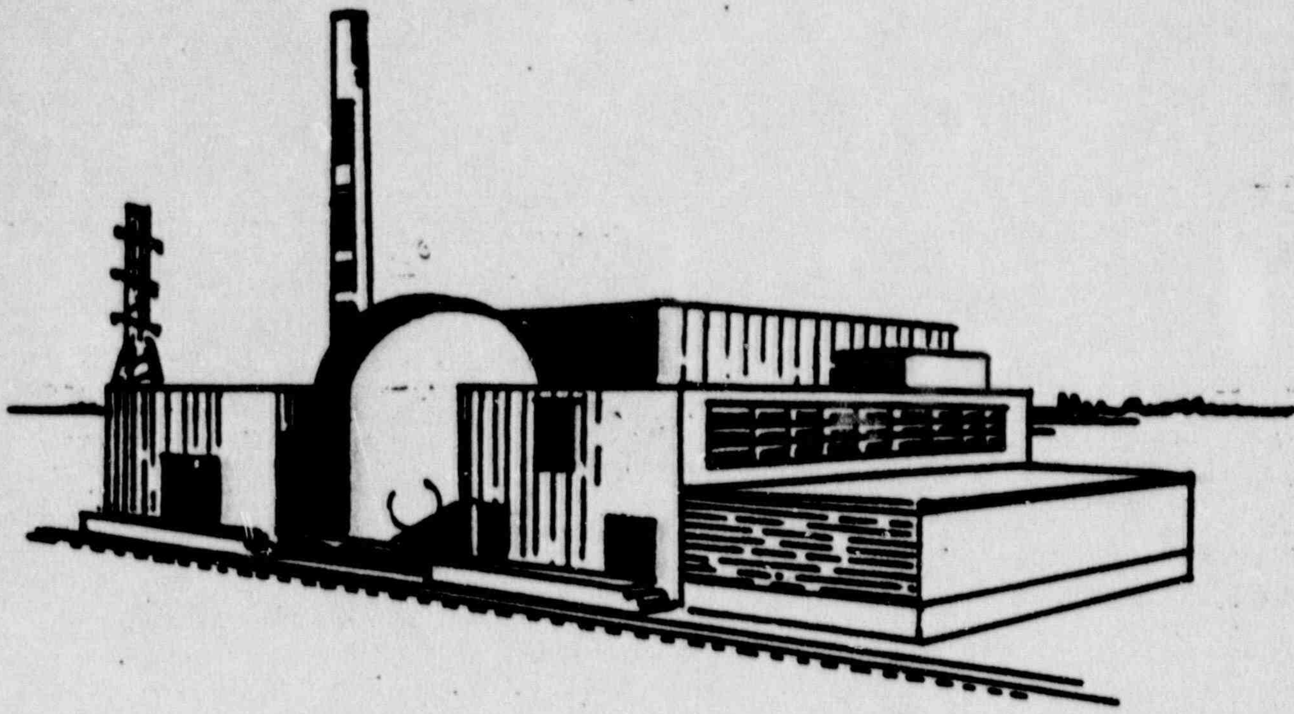
NSP 680



Pathfinder Atomic Power Plant

INCORPORATING INTEGRAL NUCLEAR SUPERHEAT

SIOUX FALLS, SOUTH DAKOTA

REPORT TO UNITED STATES ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

LICENSE NO. DPR-11

SIX-MONTH OPERATING REPORT NO. 4

PLANT OPERATING EXPERIENCE

NOV. 19, 1967 TO MAY 19, 1968

NORTHERN STATES POWER COMPANY

PATHFINDER ATOMIC POWER PLANT

Six Month Report No. 4

November 19, 1967 to May 19, 1968

Prepared By:

Northern States Power Company

June 17, 1968

TABLE OF CONTENTS

I. Introduction

Northern States Power Company's Provisional Operating License No. DPR-11 for the Pathfinder Atomic Power Plant requires that:

> "Within 30 days after the completion of six months of operation of the reactor (calculated from the date of completion of Phase II of the Power Operation Test Program), and at the end of each six-month period thereafter Northern States shall submit a written report to the Commission which summarizes the following:
>
> (a) Total number of hours of operation and total energy generated by the reactor;
>
> (b) Number of shutdowns of the reactor with a brief explanation of the cause of each shutdown;
>
> (c) Operating experience including levels of radioactivity in principle systems; routine releases, discharges, and shipments of radioactive materials; a description of tests performed in the reactor; and the results of any test analyses completed during the period including results of tests required by the Technical Specifications; a summary of experiments conducted; number of malfunctions in the control and safety systems with brief explanations of each; and a discussion of data obtained relating to superheater operation;
>
> (d) Principle maintenance performed and replacements made in the reactor and associated systems including a report on various tests performed on components of the reactor and associated systems;
>
> (e) A description of the leak tests performed pursuant to the Technical Specifications and the results of such tests including a description of any necessary corrective measures taken to meet the requirements of the Technical Specifications for assuring the specified containment leak tightness;
>
> (f) Significant changes made in operating procedures and in plant organization;
>
> (g) Radiation levels recorded at both on-site and off-site monitoring stations."

Section 50.59, 10 CFR 50, permits the holder of a license authorizing operation of a nuclear reactor to make changes in the facility, changes in procedures, and conduct tests or experiments, provided that the change, test or experiment does not involve a change in the technical specifications or an unreviewed safety question. The licensee is required to report such changes to the Commission.

This report is intended to summarize plant and reactor operations for the six month period from November 19, 1967, through May 19, 1968, thereby satisfying the reporting requirements of the Provisional Operating License DPR-11 and Section 50.59 of 10 CFR 50.

This report is a compilation of many documents and reports. The Pathfinder Safeguards Report, ACNP 5905, and the last semi-annual report to the Atomic Energy Commission, dated December 15, 1967 are the source of much of the descriptive material. The results of inspections of reactor components are taken from inspection reports written by the Pathfinder engineering staff.

A section of descriptive material has been added to this report to provide a background for readers who are unfamiliar with Pathfinder.

There are references in this report to "rust" found on various stainless steel components. The "rust" was probably the result of the stainless steel coming in contact with items made of steel such as cables or tools. The term "rust" is not intended to imply that the stainless steel was oxidizing.

II. Report Summary

The Pathfinder reactor remained shutdown during this report period. The reactor fuel, control rods, and vessel internal components were removed for inspection.

A general failure by cracking of the boiler control rods was found. The blades are 2% boron in stainless steel plates. The superheater control rods were not inspected during this report period.

The steam separator discharge nozzles and lower baffle of the boiler fuel holddown mechanism had failed. The wear on some of the boiler control rod extensions and their guides in the steam dryer assembly was higher than expected.

The reactor components were extremely radioactive as would be expected. There were no personnel radiation exposures above legal limits. The releases to the environment were minimal. There were no shipments of radioactive waste during this period.

Two "Class C" leak tests were conducted during this report period. Minor repairs were made at each test.

There were several new engineers added to the staff. By the end of this report period the new employees and some of the original staff were reassigned.

III. Reactor Description

The Pathfinder reactor has a two region heterogenous boiling water core. The superheater region is approximately a right circular cylinder in the center of the boiler region. The superheater fuel contains highly enriched uranium dioxide in a stainless steel cermet clad with stainless steel. The boiler fuel contains a slightly enriched uranium dioxide clad in Zircaloy-2. Light water is used as the coolant and moderator and is recirculated through the boiler region by pumps located external to the reactor vessel. Saturated steam at 489F is generated in the boiler region and passed to the superheater. The bulk exit temperature from the superheater region is 725F.

The reactor vessel is fabricated of stainless steel clad carbon steel. Three recirculation pumps force water through the boiler region of the core. Feedwater entering the reactor is mixed with the water entering the recirculation pumps to provide subcooling. Each of the three recirculation pumps is connected by a closed loop of 22 in OD stainless steel clad pipe. The rated flow in each loop is 21,600 gpm.

A total of 44, 10 inch diameter, centrifugal type steam separators are located inside the reactor vessel around the boiler region in the dowcomer space. The separators are designed to remove entrained steam from the recirculation water and prevent loss of pump head due to entrained steam and loss of pump NPSH.

A mesh type steam dryer is located above the steam-water interface. Saturated steam generated in the boiler region is passed through the steam dryer to remove moisture particles entrained in the steam.

A holddown structure is located above the boiler region of the core. The holddown aligns the boiler fuel and prevents movement. The actual weight of the holddown is supported by the core shroud (baffle).

The core shroud is a circular baffle fabricated of stainless steel. It completely surrounds the boiler fuel to prevent bypass of the coolant flow, positions the boiler fuel elements, and serves as a mounting surface for the holddown structure.

The boiler and superheater fuel elements are supported by structural members of stainless steel. These are rigid members to maintain position and alignment of the fuel elements with respect to each other and with respect to other components such as control rods and the holddown structure.

The steam separators and steam dryer are supported on stainless steel shelves and by brackets welded to the wall of the reactor vessel. Mountings are designed to facilitate removal of components if necessary and to prevent flow-induced vibrations.

The feedwater is returned to the reactor vessel and injected into the recirculation water through a feedwater distribution ring located in the vessel just above the suction openings of the recirculation pumps. The feedwater is injected in this manner to provide adequate subcooling of the recirculation water.

IV. Operating History

The Pathfinder Plant achieved initial criticality on a fifteen element slab core on March 24, 1964. The full core loading and zero power testing on the 2.2 w/o reference core was completed late in the fall of 1965. At that time additional excess reactivity was gained by loading thirty-two 3.2 w/o boiler elements in place of 2.2 w/o elements. The reference core for power operation included 64 2.2 w/o elements, 32 3.2 elements, 409 highly enriched superheater elements, and 40 poison shims. Additional zero power tests were completed on this core early in 1966.

The second phase of operation was limited to tests at power levels up to 8 MWT. This phase was completed in May of 1966.

The third phase of operation was an escalation to 100% power. Power was increased in 20% increments. At each new power level a series of tests were run. The full power recirculation flow had been expected to be 60,000 gpm. During low power testing the full power recirculation flow was extrapolated to be about 55,000 gpm. The maximum allowable power was limited during the testing early in 1967 by a Technical Specification requirement of 348 gpm of recirculation flow for each thermal megawatt. A Technical Specification change was approved in May 1967 which allows full power operation with a reduced recirculation flow capability.

During the summer of 1967, difficulties were encountered with reactor recirculation flow. Substantial power generation did occur during this period and the initial rise to 100% power was completed on September 12, 1967. Additional 100% power was scheduled prior to a planned extended shutdown. A main condenser tube failure forced an early shutdown on September 16. The plant remained in the shutdown condition for a scheduled poison shim removal from the boiler core. At the time of the shutdown the reactor was capable of about 75% power with an equilibrium xenon condition. The reactor had a total fuel burnup of 16,635 megawatt days.

Early in the summer of 1967 a change in the recirculation flow rate was noted during a routine heat balance. An investigation of the possible causes was begun.

In June a rapid drop of 1500 gpm at the center pump (#12) was noted. Persons thoroughly familiar with manufacture and installation were contacted. Changes in the impeller or fluid bearing were considered the most probable cause. A series of pump tests were proposed to pinpoint the source of the difficulty.

Flow changes occurred again in July. A suggested cause at this time was pump cavitation as a function of subcooling. A test was proposed to evaluate the effect of operating at various feedwater temperatures. The feedwater temperatures were varied to change the amount of subcooling. The results were inconclusive. The test proved that the lower feedwater temperature was a desirable operating condition but gave no information about the cause of the flow changes.

During the latter part of July other flow changes were again experienced. It was generally agreed that pump cavitation was the probable cause. A representative of the manufacturer's pump department was on-site at the end of July to collect pump vibration and balance data to prove or disprove the cavitation theory. The tests and measurements at this time disclosed that the resonant frequency of the pump loops was close to the rotational frequency of the pumps. Wood braces were installed and permanent pipe hangers were designed to change the natural frequency of the loops.

Other flow changes were observed during August. The situation was being studied by representatives of design, manufacture, construction, and operation groups associated with Pathfinder.

Since the September shutdown would be for core alterations it was extremely desirable to operate at 100% power before the shutdown. The Pathfinder Safety Committee was polled for their comments. The pump manufacturer was contacted for their comments. Since there were no objections an escalation to full power was made.

During the startup for the last reactor operation prior to the scheduled shutdown a tube in the main condenser failed. The contamination of the reactor water with cooling tower water required an immediate shutdown. A considerable amount of time and effort was spent cleaning up the reactor and associated systems.

Late in September plans were formulated for inspections to be completed during the shutdown. At this time any mechanical failures were considered doubtful but a complete inspection was planned.

On October 18 number 12 recirculation pump was pulled for inspection and maintenance. After the pump motor and impeller were raised one vane of a steam separator discharge nozzle was found lodged against the discharge guide vane of the pump. It was apparent that at least one steam separator had failed and an extensive dismantling and inspection of the reactor internals was in order.

V. Reactor Component Inspections

A. Boiler Fuel Elements

As part of the shim removal shutdown inspection eight boiler fuel elements were removed from the core and inspected. After the steam separator failures were discovered, the entire boiler core was unloaded and inspected. In this report the findings of these inspections are reported and an analysis of the effects of crud buildup is presented.

The original intent of the boiler fuel inspection was to look at only eight elements from the core. These eight would consist of four 3.2% assemblies and four 2.2% assemblies. These eight assemblies were inspected before pulling No. 12 recirculation pump.

After discovering that the separators had failed, the entire boiler core was unloaded. Because it was thought that possibly a small piece of a broken separator nozzle vane may have hung up in a fuel element nozzle assembly, each fuel element was inspected for nozzle blockage as it was removed from the core.

When the boiler core was unloaded and put in the storage pool, the question was raised in the December 12 Safety Committee meeting as to the thickness of crud buildup, if any, on the fuel pins. Two high burnup elements were selected and an attempt was made to measure this thickness.

The first phase of the boiler fuel inspection consisted of inspecting eight preselected elements. The results of this inspection are reported in a letter from M H Clarity to the file which is attached to this report. This inspection revealed no abnormalities in the eight elements.

This inspection consisted of looking at the grid bars between all of the fuel pins and up into the element inlet nozzle. The actual inspection took place as each element was removed from the core. The element was positioned over a 45 degree mirror and the interior of the inlet nozzle was viewed with the periscope and the right angle viewing head. A small light mounted on a "U" shaped tool was inserted into the nozzle and used for illumination. Using the binoculars, the spaces between the fuel pins, the grid areas and the upper area of the nozzle interior were checked for any foreign objects. With the light shining up from below, the area between the fuel pins and the area below the lower grid was easily and completely inspected. Using the combination of looking with the periscope from below and the binoculars from above, a complete inspection for foreign objects in the fuel element was possible.

The significant finding of this inspection was the piece of separator vane in the nozzle of element number 223 from core location R-BB-9. The exact shape and the location of the piece in the nozzle of the fuel element is reported in the separator vane accountability report (page 50). The piece measured 2 inches X 2-1/8 inches with a thickness of about 1/8 inch. This is about the maximum size piece that could fit into the bottom nozzle. Mr Pearce, A-C Co., at the December 12, 1967, Safety Committee meeting presented calculations showing that the largest piece of metal that could enter the nozzle could reduce the total flow through a fuel element by 4.3%. The construction of the lower nozzle and guide vanes prevents serious blockage of flow to the fuel elements by preventing loose material from covering the inlet nozzles.

Element 223 was given a complete visual inspection and the spacing of the fuel pins was checked with a gauge that would show any pins that exceeded the original manufacturing tolerances. The inspection revealed no abnormalities in the fuel element.

To satisfy a Safety Committee inquiry, an inspection of two high burnup boiler fuel elements (#213 and #48) for thickness of deposit on the fuel pin 3 was made.

This inspection consisted of scraping the elements with a wooden paddle and measuring the diameter of the corner fuel pins. Also an attempt was made to determine the thickness of the deposit scraped off when measuring the corner pins.

After scraping the fuel pins with the wooden paddle no marks were visible when looking through the periscope. A slight cloud of suspended crud in the water was visible after scraping with the paddle. This light suspended material was also visible when each fuel element was pulled from its boiler box. This material is so lightly attached to the fuel elements it is thought that it would easily wash off in the core. The source of this material is thought to be natural settling of crud in the vessel from components removed. Because of the looseness and small amount of this material it was not considered significant.

The measurement of the corner fuel pins was done using a micrometer put on the hook tool handle. This was set at the nominal diameter of the pins (.367 inches for the upper and .409 inches for the lower) and slipped on the corner pins. The micrometer went on all pins except one upper pin of element No. 48. The micrometer was opened to .370 inches to get it on. This is .003 inches over the nominal .367 diameter but within the manufacturing tolerances of ± .006 inches for the pins.

The micrometer marks were left on the pins. Other areas on pins have been marked during handling. These marks are believed to be where the oxide coat has been removed by light contact. By observing these areas where the oxide has been removed it is estimated that this thickness is no more than 1 or 2 mils. It was impossible to measure accurately the thickness of the oxide because of a lack of "feel" on the micrometer. The use of the micrometer showed that there was no swelling of the tubes and gave an indication of the thickness of the oxide buildup.

Inspection of the boiler fuel indicates very little crud buildup. The deposit on the fuel as observed appeared to be of an oxide nature. If the buildup is taken as mainly oxide, the buildup permitted to keep below the 600°F limit of oxide corrosion is then raised to approximately 2.5 mils of oxide.

General Electric studies indicate that the buildup on Zircaloy II fuel cladding reaches a maximum of about 2 mils. Since the deposit on the fuel shows signs of approaching 2 mils, it is recommended that at the next outage after an approximately equal amount of burnup that the fuel be inspected for an increase in buildup.

In conclusion there have been no indications of a fuel cladding failure. The fuel cladding appears to be sound. Since the core has operated only two hours at full power it is difficult to say at this time what effect many hours of full power operation will have on the fuel cladding. No abnormalities could be found on any of the fuel elements inspected except for the piece of separator vane found in the nozzle of element No. 223.

There is a slight coating of loose crud that should wash off when the fuel is replaced in the core. The oxide coating on the fuel pins is about what is to be expected on Zircaloy II cladding. The total thickness of deposit on the fuel cladding is no more than 1 or 2 mils thick. This deposit is thought to consist mainly of an oxide coating.

C O P Y

October 18, 1967

FROM: M H Clarity

TO: File

SUBJECT: Irradiated Boiler Fuel Inspection

General

During the week of October 8-14, 1967, a thorough visual inspection was made of the eight Core I boiler fuel assemblies around control rods No. 1 and No. 3. Total core burnup at this time was 16,600 MWD. The assemblies around rod No. 1 were 3.2% "spike" assemblies, No.s 203, 204, 210 and 228. Those around rod No. 8 were 2.2% "Quad box" assemblies, No.s 88, 89, 90, and 91. The general condition of all assemblies was quite good. No mechanical defects or changes were detected. The assemblies were generally clean and crud free. Some small pieces of foreign material were found on the assemblies as noted in the more detailed description below.

Upper End Fittings

Slight rust pitting was noted on the upper end fittings -- slightly less on the 3.2% assemblies which have chrome plated upper fittings. The nickel plugs on the quad box assemblies are dark gray and show up quite well.

Upper Grids

All upper grids were clean and crud free, all studs and tack welds were found in good condition. One stud nut on assembly 210 was bright and clean while all others were found rusty colored. No cracks, warpage, or other abnormalities were noted.

Assembly No. 90

Assembly No. 90 was inspected in detail. The close tolerance sections of the nozzle were inspected for scratches, gall marks, etc., and found in very good condition. This, along with the ease with which the assemblies were lifted from the core, indicates there is no problem with galling or sticking of the nozzle in the grid plate. A small piece of rusty crud (about 1/8" diameter) was seen on the spring. On two sides of the assembly there was a light coating of gray, powdery crud on the spring. On the outside of the round-to-square transition section just above the spring there were "splotches" or red deposit. The fuel pins were light gray colored -- not as black as a new assembly. The intermediate grids were clean and there were very few marks on the side "bumpers". All screws and welds on the side plates at both ends were found in good condition. A mirror was used to look at the bottom grid which appeared to have no crud buildup -- although it was impossible to see the bottom grid as well as the upper grid, it appeared to be very similar (clean, with light rust color).

Individual tube straightness, rod spacing (measured with gages), fuel tube surface condition and overall straightness of the assembly were judged to be quite good.

Other Assemblies

The remaining assemblies were inspected and found similar in appearance to No. 90 with the following exceptions:

- No. 89 Not as much red deposit. Slight gray deposit on corner tubes at intermediate grid plates.
- No. 88 More rust spots than No. 90. Small piece of crud on spring.
- No. 91 Very few rust spots. Slight deposit at intermediate grids.
- No. 210 Upper side plates have slightly thicker accumulation.
- No. 204 Very few rust spots.
- No. 205 Small piece of metal ~1/16" x 1/16" x 3/8" found on lower grid. Spectrum analysis indicates high chrome content. Activity about 1 R/hr at contact.

Pictures

Color and black and white pictures showing the upper grid on assembly No. 210 and portions of the lower nozzle, side plates, and fuel tubes on assembly No. 90 are in the plant picture file.

M H Clarity
Plant Results Engineer

MHC/sge/daw

Pages 10 and 11 of this report have been deleted.

B. Boiler Boxes

The boiler boxes are highly radioactive and therefore were kept submerged in water throughout the inspection. As each box was taken out of the vessel, the outside was given a cursory inspection using binoculars. The boxes were moved to storage racks in the spent fuel storage pool where they were given a more complete inspection.

There are two types of boiler boxes. One box holds a single fuel assembly while the other holds four assemblies separated by an envelope that guides a cruciform control rod. Five single element boxes and four of the four element cluster boxes were chosen for a thorough inspection. These nine boxes contained about 22% of the boiler fuel assemblies.

The inner surfaces, outer surfaces, and the bottom side of the base plates were visually inspected using the underwater periscope. The inside of the control rod guide envelopes were inspected using the borescope attachment. By this method, any cracks, missing screws, rub marks, and deposits of crud could be found, if they existed.

Thirty-two of the 96 boiler fuel assemblies of the Pathfinder reactor are in single element boxes. The single element box has walls of Zircaloy-II with a stainless steel base. A sample of five of the 32 single element boxes was chosen for a more complete inspection. The following were chosen as the most likely to show any abnormalities:

BH-4 This box is in a high flux area.

BL-10 This box is on the side of the four element cluster box, CR-2, where an unusual mark was found.

BK-5 and BL-3 These two boxes are adjacent to CR-1, the location where a control rod had a wear mark on the extension.

BA-6 On two sides of this box there is more clearance than most other boxes.

Each quad box supports four fuel assemblies which are clustered around a control rod. Sixteen of these boxes contain 64 of the 96 boiler fuel assemblies. A cruciform envelope separates the four assemblies to serve as a control rod guide. At the base of each guide are 20 coolant passages for cooling the control rod. The quad boxes have Zircaloy-II walls with a stainless steel base.

As with the single element boxes, only a sample of the quad boxes was given a closer inspection. Four of the 16 boxes were chosen on the following criteria.

CR-2 (Group V rod) When this box was removed from the vessel, an unusual mark was observed on the outer surface.

CR-9 (Group II rod) This box is in a high flux area. Also, the control rod in this location had many cracks at the stitch welds and in the nose of the rod.

CR-13 (Group III rod) There were small wear marks on the extension of the control rod in this location.

CR-1 (Group IV rod) The extension of the rod in this location had a deep wear mark. Also, the rod was at one time dropped without dashpot action.

No cracks were found in the boiler boxes. A mark on the east side of CR-2 appeared to be a crack at first. However, it lacks the depth and is straighter than a crack normally is. A similar mark can be found on the north side of CR-9. This mark starts in the wall material and continues in the base plate. These two materia's are bolted together so a crack would not normally propagate from one to another. Both marks originate from screws; it is felt that the marks are scratches left from a screwdriver. Rust bleeding from the scratch gives the appearance of a crack.

Each screw observed in the boiler boxes is securely in place.

In the fuel assemblies there is a grid every 18 inches which keeps the fuel from touching the sides of the boiler box. This grid rubs on the side of the box but no serious wear was apparent.

There is no cracking in the control rod cooling tubes over a 3-1/2 inch diameter field of view. Any gross cracking beyond this area could have been seen from the inside of the fuel assembly locations or the bottom of the base plate. No such cracking was observed.

The upper 4 inches of the control rod guides show bright metal from normal rubbing but no appreciable wear marks.

There are no significant rub marks. (Some marks remain due to fuel assembly removal, shim removal, etc.)

Near the top of the south side of CR-2 (box number 407) there is a mark about 3 inches long by 1/8 inch wide. Using a 3/32 inch welding rod on a 10 foot section tool, no depth of the mark was detected. There is no sign of this mark on the inside of the box. The mark was apparently made either by some slight movement of a shim or by some tool used in removing the core components.

The outside surfaces of the boxes are coated with a powdery, red deposit. The inside surfaces have a somewhat heavier deposit of scale.

When tapping the base plate, holes were drilled completely through the beveled seat of the base plate in the quad boxes. In most cases, the hole is filled or partially filled with crud.

C. Boiler Control Rods

Shortly after the reactor was shutdown a spot check of the boiler control rods showed that cracks propagated from about 80% of the stitch welds of the 2% natural boron stainless steel cruciform rods. At that time it was felt that the rods could not be used for power operation again but should be replaced by the new B_4C. A thorough inspection of all 16 boiler control rods followed to (1) study the extent and cause of cracking in the 304ss-B rods and (2) note any other abnormalities in the rods which might point to other problems in the system.

There is a large number of cracks in the borated blades, particularly in the high burnup group.

None of the rods are bowed, as such, but five rods are bent or misaligned slightly.

There are wear marks in the control rod extensions where the fully withdrawn rod rubs against the reduced area of the steam dryer assembly.

There is no swelling of the rods.

The nose piece of the control rod is deformed in the shape of the control rod coolant inlet channels of the quad box.

There are rub marks on the lower 4 inches of the blade. This area remains in the cruciform guide of the quad box when the rod is fully withdrawn from the core.

The Pathfinder boiler control rods are cruciform shaped rods made of borated stainless steel. The borated section is 10-7/16 inches wide by 1/4 inch thick by 6 feet long. On the lower end is welded a 1/2 inch nose piece of 304ss. On the top of the rod a transition piece reduces the 10 inch blade to a 3 inch cruciform extension made of 304L ss which is 9 feet long. A complete description of the control rods is given in ACNP Report No. 63001, "Pathfinder Atomic Power Plant Boron Stainless Steel Control Rods for the Pathfinder Reactor".

The neutron absorber section of the control rod is 2 w/o natural boron (1.95 a/o B^{10}) alloyed with type 304 stainless steel having a high nickel content (10.65%). Two narrow blades were butt-welded against a full width blade. Stitch welds were made in an alternate pattern to circumvent unusual stresses. Following welding, the rods underwent a high temperature stress relief procedure. All welds were given a dye penetrant check; the welds connecting the nose piece were also radiographed.

The properties of metals are known to change under irradiation. In particular, the yield and tensile strength increases while the per cent elongation decreases. Irradiation damage is of particular importance in the 304ss-B alloy, since the B^{10} (n, alpha) $L_1{}^7$ reaction generates helium gas in the steel lattice structure as the rod absorbs neutrons. From the analysis of data concerning irradiation effects, the maximum local burnup of boiler control rods was set at 1.0 a/o. The rods were to be removed due to loss of reactivity before this burnup was accomplished. In fact, the maximum point burnup of the rods at the end of core life was calculated to be 0.91 a/o; this is five times the average burnup. The minimum burnup occurs in the cruciform weld area.

Pathfinder reactor has 16 boiler control rod locations. (Note: There are two ways to identify the control rods, by their core location and serial number. The numbers used in this report refer to core locations.) Table I below lists the five control rod groups.

TABLE I

Group Number	Name	Rod Number
I	Superheater	17, 18, 19, 20
II	Flats	5, 8, 9, 12
III	Diagonals	4, 6, 11, 13
IV	Outer	1, 7, 10, 16
V	Outer	2, 3, 14, 15

In the past the rod programming has called for groups IV and V withdrawn to reach critical. Depending on the poison in the core, group III may or may not be withdrawn to become critical. At power, group III must be fully withdrawn. This means the outer rods are in the fully withdrawn position during almost all reactor operation. On the other hand, group II has a high burnup since it is partially in the core during operation and also because it is located in a high flux region of the core.

When the Pathfinder boiler core was unloaded, the boiler control rods were placed in their storage racks in the storage pool. Since the borated blade was highly radioactive, it had to remain underwater at all times. Each rod was removed from the storage rack and positioned adjacent to the underwater periscope. By raising and lowering the control rod, the full length of each quadrant was inspected. Any unusual marks were noted for further investigation and study. Lengths of cracks, welds, etc., were determined by comparing the length to the 10 inch or 3 inch blades; measurements made underwater were such comparative estimates.

Each rod was given a line-of-sight test for bowing while it was suspended from the bridge crane. For bowing or bending in the extension, the rod was taken partially out of the water using the dry loading tool. The rod could be safely raised so that the entire extension was out of the water, with the transition piece just breaking the water surface. For bending, over a short length, a straight edge was laid on the rod and for entire rod, a plumb was suspended with the string touching the top and bottom of the rod. The offset was again measured with a tape.

Unusual wear marks were found on the extensions of some control rods during the underwater inspection. The size and location of each wear mark was then carefully measured.

Rods 4 and 5 were chosen from groups III and II respectively to be checked for swelling. A micrometer with a 5 inch throat was bolted onto a 10 foot section of aluminum tubing. With the rod extension out of the pool the micrometer was fitted onto the blade and moved over the length of each blade several times. The tool was taken out of the water and re-adjusted until the micrometer would not slip over the blade. By this method, the difference between 250 mils and 255 mils was readily noticed. Therefore 5 mils was the maximum uncertainty in this go/no go measurement.

Numerous cracks were found in each of the 16 boiler control rods. All cracks were confined to the borated section of the blade. The size and number of cracks in each rod is tabulated in Table II, page 17. There were two types of cracks found. One type started at the outer edge of the blade about 1 to 2 inches from the bottom of the rod. They propagate toward the center of the blade and curve toward the nose piece. The second type of crack propagates from the stitch welds. In most cases these cracks start at the ends of the welds and propagate about 1 inch. Some of these cracks extend from one end of the weld around to the other end to form a "football-shaped" crack. Cracks are visible on both sides of the blades, that is, the crack is apparent through the full 1/4 inch metal thickness. At some welds there is a crack on both sides of the weld so as to completely surround the weld.

Of the total cracks found in the boiler control rods, 54% were in group II, 9% in group III, 23% in group IV, and 14% in group V. Of the boiler control rod burnup calculated for the core at 36,000 MWD, 67% of the burnup occurs in group II and 11% occurs in each of groups III, IV, and V. Therefore the number of cracks appears to be a direct function of irradiation exposure. As additional evidence of this fact, it was noted that more cracks occur in the lower half of the group II rods (where burn-up is higher) than in the upper half.

TABLE II

roup	Rod Number (Core Location)	Rod Serial Number	Offset Construction of Transition Piece	Longer than Specified Welds	Cracks from welds less than 1 inch	Cracks from welds larger than 1 inch	Cracks from the edge of blade	Rods Bent or Bowed	Sides of Extensions Having Wear Marks	Sides of Blades Showing Rub Marks	Comments
II	5	3			53	9	0	X	S (Small)	0	
II	8	17	X		39	14	0			0	4 cracks g from the top to the bottom of the weld
II	9	1		X	39	16	2	X		2 (Small)	
II	12	7	X	X	32	5	0	X		5 (Small)	
III	4	9			5	0	0	X	N	6	
III	6	14			15	3	0		(Small)	4 (Small)	direction wear mark was not recorded
III	11	5		X	2	0	0			2 (Small)	
III	13	12	X		7	2	0		W-S	5	
IV	1	8		X	14	11	0		N-E	8	2 cracks g from top t bottom of the weld
IV	7	10	X		9	5	0			0	
IV	10	16	X		13	2	2		E	8	
IV	16	13			26	10	3		S-E	8	2 cracks g from top t bottom of the weld
V	2	4		X	9	4	0		N-W	8	
V	3	15	X		3	3	0	X		8	1 crack is about 6" long
V	14	11	X		11	6	0			7	
V	15	6	X	X	15	3	2		S-E	5	

No cracks were found in the extensions. There were machine marks which on first sight appeared to be cracks. One such mark was investigated visually and with dye penetrant when the extension was taken out of the water. No cracks were found in any of the 16 control rod extensions.

There was no bowing, as such, observed in the boiler control rods. Five of the 16 rods appeared bent during the line-of-sight test. The bends were sharp and over only a few inches, not gradual bowing effects as expected from a differential thermal stress or any internal stress caused by the differential neutron absorption and swelling. The bent rods followed three separate patterns.

Rod numbers 3, 9, and 12 had a slight angle where the extension joined the transition piece. The acute angles were toward west, east, and west for the three rods, respectively. By holding the string for a plumb against the top and bottom of control rod number 3, the offset at the bend was measured to be 1/8 inch. The 1/4 inch control rod blades fit into a 17/32 inch cruciform channel in the quad box. This allows room for an offset of 9/32 of an inch before the rod binds in the guide channel. If it were interferring with withdrawal and insertion of the control rod, there would have been marks on the blades. This is not the case so it was assumed that there was no interference with rod movement.

On the south side of rod number 4 there is an "S-shaped" bend pattern in the extension over a region from about 8 to 20 inches above the transition piece. By laying a 24 inch straight edge on this region, the control rod offset was measured to be 3/32 to 1/8 of an inch.

About 11 inches below the latch on control rod number 5 is a bend. The solid 3 inch plate which forms the east and west blades forms an acute angle on the south. The north blade is acute on the west and the south blade is acute on the east. Each bend has an offset of about 1/16 inch when measured with a 24 inch straight edge. Judging from 3 of the 4 blades, this appears to be a rotational type bend.

The bends could be a result of (1) some tangential force (such forces do not exist in the reactor), (2) improper handling (lifting the rod by the latch while it is in a horizontal position), or (3) excessive axial loading (and simultaneous rotation as in latching in the case of rod number 5).

As the boiler control rods are withdrawn from the core, the 3 inch cruciform extension is guided through the steam dryer in a 3-3/4 inch I.D. tube. On the bottom of that tube is a reduced area (3-7/16 inch I.D.) approximately 3/4 inch long. There is wear mark on some control rods at the elevation of that reduced area corresponding to the rod in its fully withdrawn position. Corresponding wear marks were found in the steam dryer.

A deep mark was worn in the east edge of rod number 10 and another in the north edge of rod number 1. Fourteen such marks were observed, 13 of them being in groups III, IV, and V which are generally fully withdrawn at power. On all rods except numbers 1 and 10, there was no appreciable metal depletion. The wear mark on rod number 1 was 1/8 inch deep over a length of 3/4 inch; the metal depletion was confined to a 1-1/4 inch length. On rod number 10, the wear was 1/8 inch deep over a length of 3/4 inch but the metal depletion extended over a length of 1-3/4 inches. It appeared that a limit point had been changed and another wear mark was being created about 3/4 to 1 inch above the first one.

No swelling was observed in the rods gauged with the micrometer. With the micrometer set at exactly 1/4 inch, the gauge was slid over the blades with no binding.

When the boiler control rod is fully inserted in the core, it bottoms on the base plate of the quad box. This base plate has 20 coolant channels which are 1/4 inch diameter with about a 1/2 inch diameter bevel where the control rod bottoms. Small deformations were found on the base of some control rods corresponding to the size and spacing of the coolant channels. It appears that the rod bottoms so hard that the metal is deformed. This problem was not observed immediately; only rod numbers 1, 3, 4, 9, 11, and 16 were checked closely. Deformations or discolorations were found on all rods. Rod number 4 was the worst with about 1/32 to 1/16 inch deformation. By scraping this area with a metal tool the bumps on the rod were determined to be metal and not merely a crud or scale buildup.

During a scram of control rod number 1 in March of 1965 the dashpot action was questionable. During the ensuing 1965 inspection, a photograph was taken to show a discoloration of the nose piece. A photograph of the same rod was taken during this inspection; deformations of the metal are now more pronounced.

When the control rods are in the fully withdrawn position, they extend about 4 inches into the quad box cruciform guide. The lower 4 inches of group III, IV, and V rods shows shiny metal where the fully withdrawn rods contact the quad box. The amount of wear is slight and could not be measured when gauging the blades with a micrometer. The fact that cracks developed near the bottom of the rod could have been a result of either vibrations or irradiation damage. Since similar cracks also developed in group II rods which were not usually fully withdrawn, it is more likely that these cracks were due to irradiation damage and stresses created by the welded nose piece rather than by the rod striking the quad box.

D. Steam Dryer

During the investigation of reactor internals, wear marks were found on the control rod extensions and in the collar sections at the lower end of the control rod guide tubes. The rod extension wear and the guide tube wear correspond when the control rods are fully withdrawn.

The function of the steam dryer is to remove moisture droplets from the steam, to steady the upper portion of the superheater assembly, to support the upper control rod guide tube, and to line up the control rods to facilitate latching.

There are 16 control rod guide tubes extending through the steam dryer. The tubes are 3-3/4 inch I.D. by 76-3/16 inches long with reducer sections at both ends. See Figure 1, page 21. The lower reducer forms a collar section in the tubes 3/4 inch long and 3-7/16 inches in diameter.

With the control rods fully inserted, the control rod latch head is centered in the lower collar to facilitate latching. The control rod latch head has an effective diameter of 3-3/16 inches which limits the horizontal movement in the collar to 1/4 inch. The control rod extensions have an effective diameter of 3 inches allowing a horizontal movement of 7/16 inch in the collar when the control rods are withdrawn. See Figure 2, page 22.

The dryer structure and control rod guide tubes are 304L stainless steel.

During the investigation of reactor internals, a notch was found on the north blade of the No. 1 control rod extension. Subsequent periscope inspection of the control rod guide tubes indicated wear marks in the lower collar section of all guide tubes in one or more quadrants. The control rod extension and guide tube wear correspond when the control rods are fully withdrawn.

A general inspection was made with the steam dryer removed from the shield pool. With the exception of guide tube wear, the steam dryer assembly was in good condition. Measurements were taken to determine the amount of metal loss from the collar sections of the guide tubes. Seven tubes were found to have measureable metal loss in at least one quadrant with the remaining tubes showing evidence of rubbing in one or more quadrants. See Table 1, page 23. The design thickness of the collar sections is 5/32 inch. The most severe wear is in the east quadrant of the No. 10 tube collar with a groove 1/2 inch wide, 3/4 inch long, and 0.070 inch deep. This groove represents approximately 45% metal loss in the collar section. Rub marks were also noted on the inside surface of the tubes. These marks extend varying distances up from the bottom of the tubes and show no evidence of metal loss. Slight rubbing of the control rod drive latch head with the guide tubes is the apparent cause of these marks.

The alignment of the guide tubes with the dryer structure was suspected as a possible cause of the evidenced wear. Positioning plates were fabricated to facilitate an alignment check of the tubes. A top plate was turned to fit the inside diameter of the top reducer and a hole

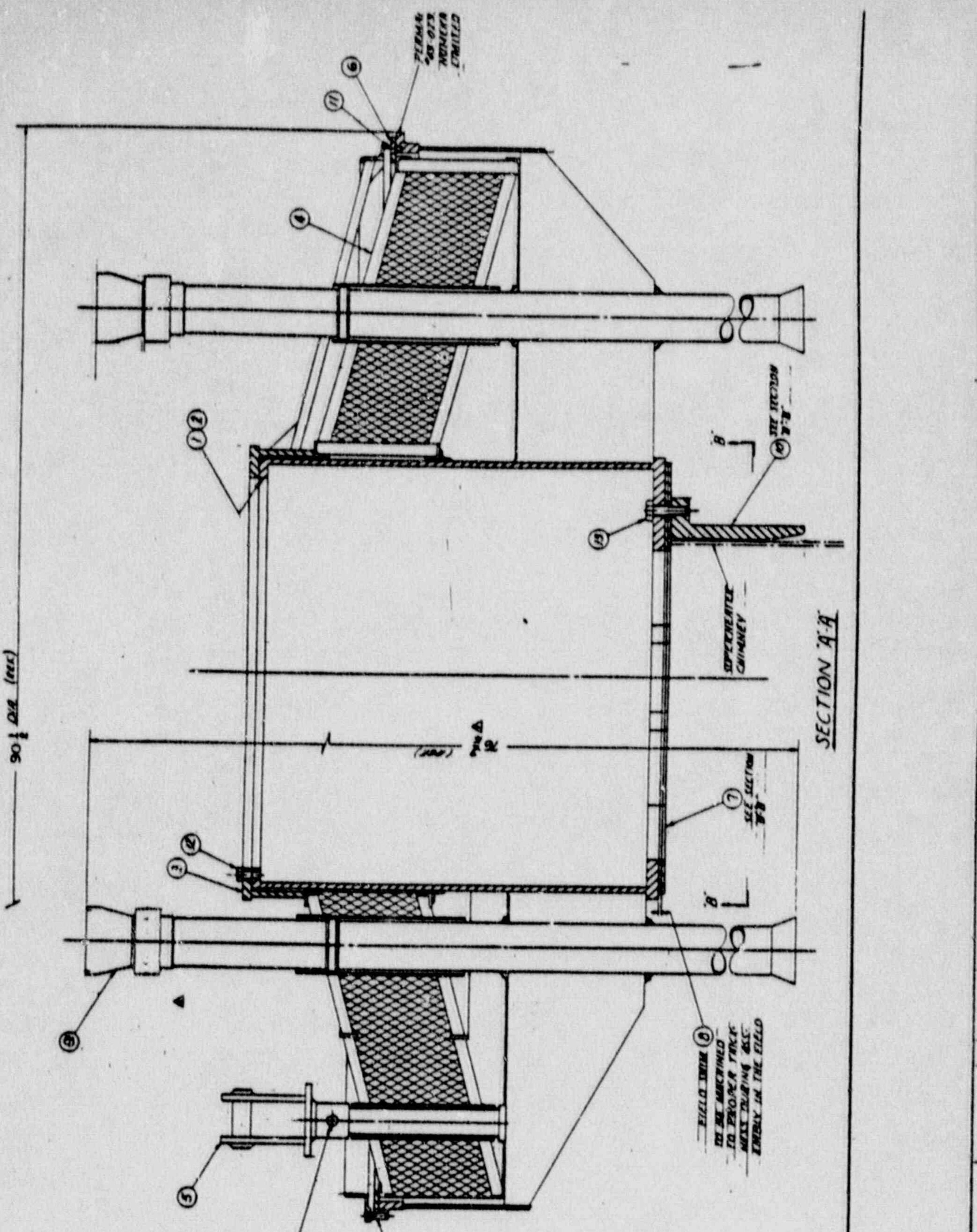
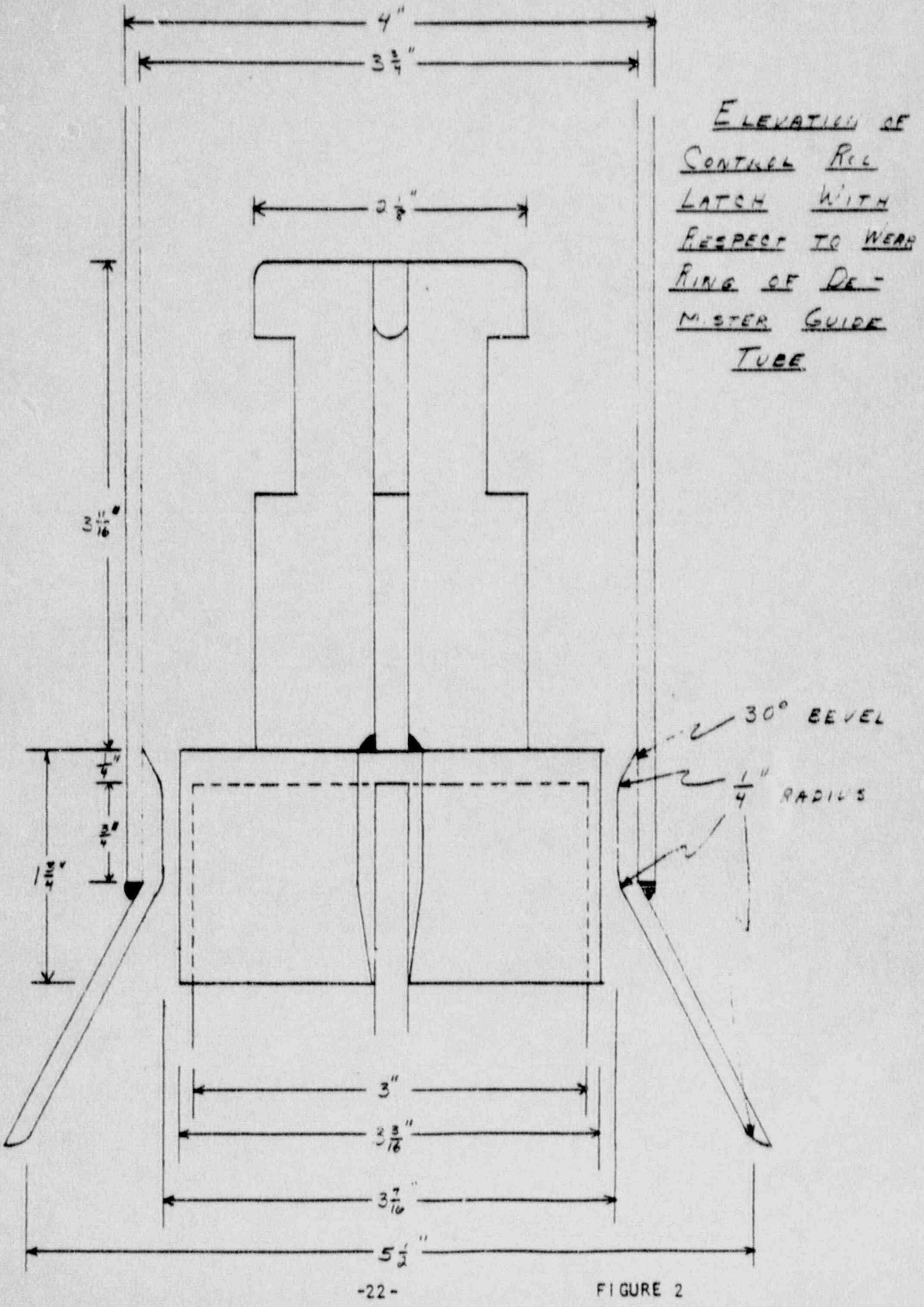

SECTION 'A-A'
SEE SECTION 'B-B'
SUPERHEATER CHIMNEY
FIELD TRIM
TO BE MACHINED TO PROPER THICKNESS DURING ASSEMBLY IN THE FIELD
90 1/2 DIA (REF)

FIGURE 1



4"
3 3/4"
Elevation of Control Rod Latch With Respect to Wear Ring of De-mister Guide Tube
2 1/8"
3 11/16"
30° BEVEL
1/4"
3/4"
1 3/4"
1/4" RADIUS
3"
3 3/16"
3 7/16"
5 1/2"

FIGURE 2

TABLE I

INSPECTION OF STEAM DRYER GUIDE TUBES

January 16, 1968

Guide Tube No.	Collar Wear-in *. (Inches) N	E	S	W	Tube Rub ** (Feet) N	E	S	W
1	.051	S	S	S	2/3			1/3
2	.028			S				3/4
3	S		S	S				
4	S			S	1			
5	.003		S	S	1			
6		S	S	S				1/3
7	.011		S	S				3/4
8		S	S	S			1	
9	S	S	S	.010				3/4
10		.070	S			3/4		
11	S	S	S	S				
12			S	S			Full	
13	S		S	S				1/2
14			S	S				1/3
15		.030	.018				1/2	
16	S	S	S	S				2/3

* S indicates slight.

** Figures given represent approximate distance rub marks extend from bottom of tube.

drilled in the center for a plumb line. A lower plate was turned to fit the outside diameter of the lower reducer section and to hold a 3 inch filter paper centered on the plate. Using these plates, a plumb was lowered through each tube, steadied, and dropped a short distance on the filter paper to mark the location of the top reducer center. Several tubes were rechecked with good repeatability on all tubes. By using the positioning plates on the reducer sections for checking alignment, it is assumed that reducers are exactly centered in the guide tubes. This is considered an accurate assumption and any significant misalignment should be detected.

Difficulty was experienced in properly leveling the dryer structure. The initial data showed the bottom centers of the tubes off predominately in one quadrant indicating the dryer was not exactly leveled prior to plumbing the tubes. The maximum measured deviation was 0.36 inch. See Table 2, page 25. Releveling and checking several tubes showed no significant improvement in leveling the dryer.

The length of the guide tubes (76-3/16 inches) requires very exact leveling of the dryer structure to obtain the magnitude and direction of any misalignment. Considerable care was taken in leveling the dryer and it was decided that the improvement was not significant enough to warrant additional personnel radiation exposure. The data taken was analyzed for relative alignment between tubes.

The alignment of the guide tubes in the steam dryer does not appear to be the direct cause of the collar and rod extension wear. While some misalignment is apparent, a maximum of 0.33 inch between tube centerlines, no consistent correlation with the collar wear marks was found. There was also no apparent correlation between the tube rub marks and the collar wear.

The evidenced wear in more than one quadrant on all tube collars suggests the control rods are being forced from side to side in the collars by the turbulance in the reactor. All the wear marks are approximately 1/2 inch wide and the control rod extension blades are 1/4 inch wide. This also suggests the rod extensions were oscillating in the collars.

The wear on the control rod extensions is in the area which corresponds with the collar sections when the control rods are fully withdrawn. There is no evidence of rub marks along the rod extension blades as would be expected if the rods were being forced against the collars when the rods are inserted and withdrawn.

The present core I control rods with 1/4 inch extension blades will not be used in future reactor power operation. New core II rods with 1/2 inch extension blades will replace the core I rods. The increased width of the new control rod extension blades will increase the effective contact area with the guide tube collars and, thus, reduce the rate of any wear on the new rod extensions.

TABLE 2

STEAM DRYER GUIDE TUBE ALIGNMENT

February 23, 1968

Guide Tube No.	Bottom Center Relative to Top Center: Inches	Bottom Center Relative to Top Center: Degrees (Ref. East)
1	0.11	-10
2	0.14	-55
3	0.20	-20
4	0.25	0
5	0.13	-20
6	0.03	0
7	0.10	64
8	0.29	-11
9	0.25	-42
10	0.22	0
11	0.17	-30
12	0.24	30
13	0.15	-20
14	0.25	32
15	0.36	-13
16	0.16	-25

NOTE: The steam dryer was leveled with respect to the dryer structure prior to plumbing the tubes.

With the pendulum construction of the control rods and the clearance required for latching, some rubbing between the rod extensions and the guide tube collars can be expected. The wear in the collar sections does not by itself present any operational problems as long as the control rods can be properly latched. Of primary concern is the possibility of a notch forming on a control rod extension and catching on a guide tube collar during a reactor scram. While this is considered an extremely remote possibility, a regular inspection of the rod extensions and guide tubes will be performed during refueling operations.

To insure proper operation of the control rod drive system, the rod drives are exercised at least once a week during power operation. Rod drop tests are also performed to check the control rod scram action. These tests monitor control rod operation and would be expected to indicate any abnormal operation.

E. Boiler Shroud

The boiler core shroud supports the fuel element holddown structure, and separates the annular downcomer region from the boiler core.

The boiler core shroud is cylindrical and completely encloses the outer periphery of the core. The top and bottom plates are machined on the inside to fit the core configuration. The boiler core shroud is 89 inches high and [illegible] inches diameter. Rigidity of the structure is maintained by suitable structural angles and gusset plates.

The boiler core shroud was inspected underwater while it was stored on the floor of the shield pool. Underwater radiation measurements showed activities near the shroud of 3500 to 8000 R/hr. A general binocular inspection was made along with a close inspection using the underwater periscope. The general condition of the shroud appears to be very good. No evidence of any warping was seen. The outside of the shroud casing was seen to have widespread small "rusty" blotches. The portion of the outside of the shroud casing which is above the steam separator shelf in the vessel has a slightly reddish tinge, while the portion below the separator shelf has a grayish tinge. At the separator shelf level the casing has a dark gray band.

The shroud casing has one vertical welded seam. The weld is in good condition. The circumferential welds at the top and bottom plates appeared to be sound.

Several superficial scratches were seen on the outside of the shroud casing. These are attributed to steel cables touching the shroud while handling.

A wide superficial scrape was seen on the outside of the shroud casing starting at the steam separator shelf line and extending downward a few inches. The shroud casing apparently rubbed the separator shelf when the shroud was removed from the vessel.

Parallel superficial scrapes were seen on the outside of the shroud casing in two places starting a short distance above the separator shelf line and extending up for several inches. These were caused by the spring-loaded shoe on a neutron window when it was removed and reinstalled.

All the welds on the inside support and gusset plates were in good condition. The holddown studs appeared to be in good condition. The two locking screws used to secure each nut on the holddown studs appeared to be in good condition.

F. Feedwater Ring

The feedwater inlet to the reactor vessel is at the northwest side of the vessel. The 8 inch inlet pipe "tees" into the distribution ring which is a nominal 5 inch pipe formed into an elliptical pipe of about 5 x 6 inches cross section. The ring has a 105 inch diameter. There are bolted flanges on the north and south and three support blocks on the northeast, southeast, and southwest. The ring is supported on the northwest by the inlet pipe and nozzle.

There are 127, 3/8 inch holes in the feedwater ring. They are near the bottom of the ring positioned to direct the feedwater away from the vessel wall.

A general binocular inspection of the feedwater ring was made by viewing the top of the ring through separator holes in the steam separator support shelf. In this manner about half of the ring could be seen. The underwater periscope was used to complete the inspection. The "inside" half of the ring could be seen with the periscope. Most of the "outside" of the ring was not inspected.

The feedwater ring, inlet tee and support blocks have a general brownish appearance. The inside rims of the distribution holes have a rusty orange appearance.

On the east-southeast side of the ring there is a foreign object sticking out of a distribution hole. The object could be part of a thermocouple well which was broken off on the suction side of No. 11 feedwater pump. Part of the well (2 inches of the original 6 inches) was found in the pump; the rest was never found. The foreign object could be a piece of welding rod left in the pipe during construction. An attempt to retrieve the foreign object will be made, but the possibility of success is considered unlikely.

At the point where the feedwater ring is welded to the inlet tee from the north the weld looks undercut at one location.

All other welds appear to be in good condition. All bolts and nuts and tack welds were found in position and in good condition.

G. Holddown

The holddown was inspected entirely under water due to radiation levels ranging from 6 R/hr on the top to 350 R/hr at the bottom on contact. The upper area of the holddown was inspected using the binoculars and the underwater periscope was used for the rest of the assembly. All areas that were accessible were visually inspected. This included:

1. Upper bracing of guide tubes
2. Lifting lugs
3. "C" washer assemblies
4. Base ring
5. Rod guide tubes
6. Base plate
7. Upper and middle baffle
8. Lower baffle

Mr John Haines, Allis-Chalmers Mfg. Co., Bethesda, Maryland, participated in the inspection of the lower, middle and upper baffles.

It was concluded that:

1. The holddown assembly has been subjected to forces in the reactor that have torn the lower flow deflector baffle loose from its supports. The baffle is bent and broken into four separate quadrants and is out of its original position.

2. A few of the rod guide channels have slight wear marks on the insides of the upper ends of the channels.

3. All of the other areas inspected were found to be free of any abnormalities.

The fuel element holddown assembly retains the boiler fuel elements against the hydraulic forces developed by the water flowing upward through the core. Through the use of the control rod guide channels, which surround each control rod in the boiler core, the holddown assembly prevents the coolant flow from exerting lateral forces on the control rods as it flows across the core to enter the downcomer region. A flow deflector (three baffles) is attached to the guide tubes to break up steam channels that may be formed at the exit of the boiler fuel elements. (Reference to ACNP Report No. 62031 "Reactor Internal Components", page 28.)

As part of the planned inspection of reactor internals the holddown was removed, rotated 180 degrees and placed on the pad in the shield pool. No difficulties were experienced in unlocking the "C" washers or removing the holddown. After removal of the boiler fuel and boiler core shroud an initial inspection of the vessel with the binoculars was completed. Two triangular pieces of perforated plate were observed lying on the steam separator shelf. The piece lying on the inner area of the shelf was retrieved but radiation levels prohibited bringing the piece to the surface of the water. By observing the shape and size of the piece it was determined that they are part of the lower baffle of the holddown.

The lower baffle is constructed of 11 gauge (.125 inch) perforated plate having 1/2 inch diameter holes on 1 inch centers. The sheet appears to have been made in one piece with reinforcing straps welded on at areas where perforations and narrow sections coincide. The area where the quadrants are connected is an area approximately 1/4 inch wide. These areas have no reinforcing straps. See Figures II, III, IV, and V, pages 33 through 36, for location of the reinforcing straps. The holddown assembly was fabricated by slipping the baffle down over the rod channels and welding it to 44, 1 inch x 1 inch angle supports which are welded to the rod channels.

The lower baffle was found separated from the angle supports except for one area. Also, the baffle was broken into four separate quadrants. The quadrants are at random levels with indications on the guide channels that the baffle parts have been moving up and down. One quadrant has a 3-1/2 inch tear in the baffle and there are random bends in the baffle where the elevation of the plate changes in each quadrant.

Three pieces are missing from the baffle. Two of the pieces are those found on the separator shelf. These two pieces are triangular in shape whose sides measure approximately 5 x 5 x 7 inches. The third piece is a rectangle which should measure approximately 2 x 1/4 inches. This piece has not been found at this time.

The middle baffle is made in two pieces of 11 gauge (.125 inches) perforated plate (1/2 inch holes on 1 inch centers). The plates are welded together on an east-west seam to form the middle baffle. This baffle is 13 inches above the lower baffle and welded to 64, 1 inch x 1 inch angle supports that are welded to the rod guide channels. The upper baffle is constructed identically and fastened to the middle baffle with 72, 1/4 inch bolts using 1 inch spacers to space the baffles.

The upper and middle baffles were inspected and found to be in good condition. No distortions, warping, cracks, or loose bracket welds were observed. The seam welds on the two baffles were sound and all of the spacer bolts were tight with sound tack welds. Some of the spacer bolt nuts have a rust colored deposit similar to that seen on other areas of the reactor internals.

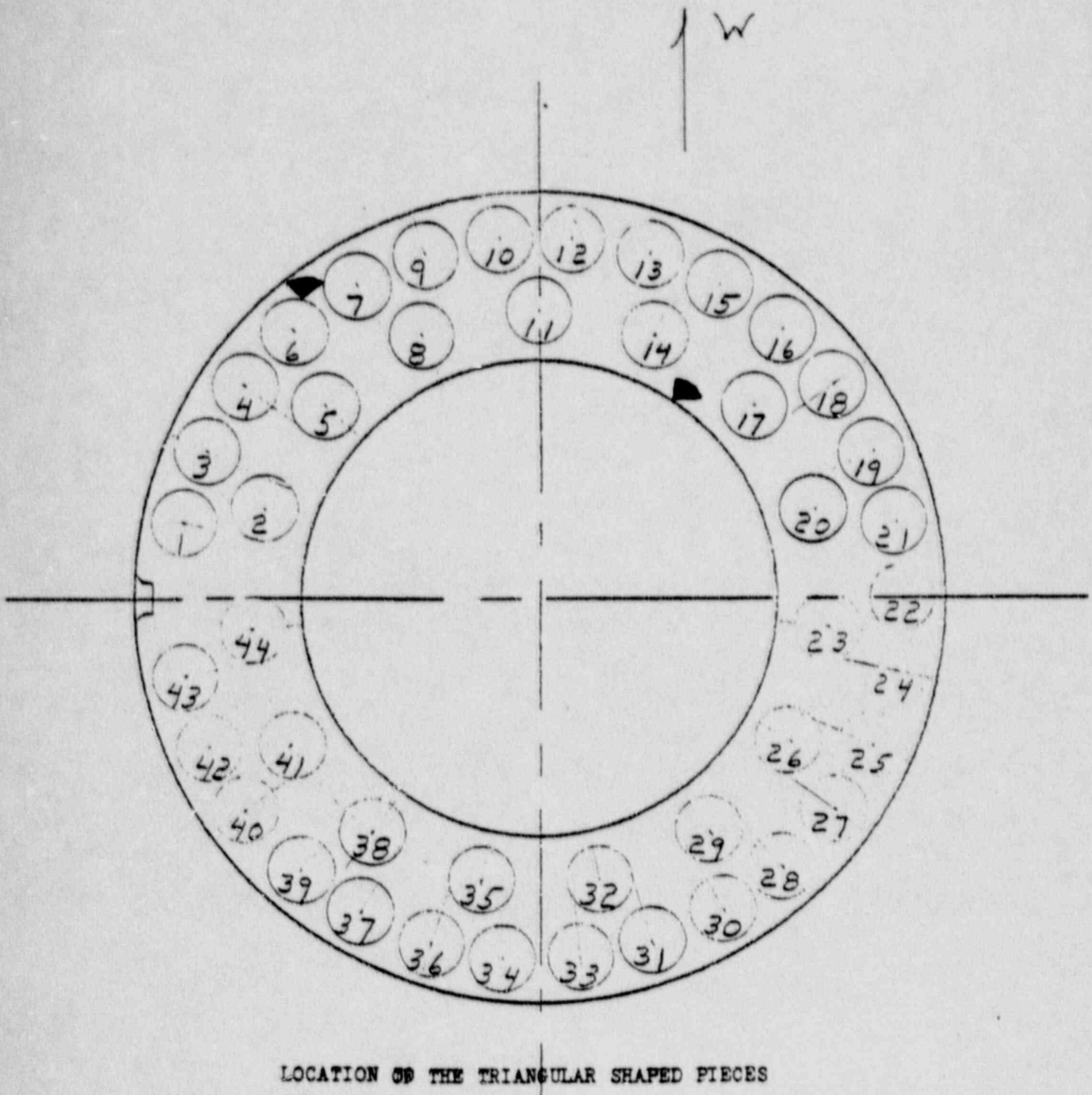
Inspection of the rod guide channels revealed no cracking, warping, or loose fasteners. Light scratching has occurred on the outisde of some of the channels where the lower baffle has moved up and down. Also, the inside of some of the channels have rub marks that have been caused by contact with the control rods. These marks were viewed with the forward viewing head on the periscope and were very light rub marks.

The holddown was supported from the polar crane and rotated past the periscope for the inspection of the base ring and bottom plate. The bottom area of the base plate was observed to have very little crud buildup. There was no evidence of cracking or warping in the plate or base ring. By looking through the cut-outs the interior cap screws holding the guide channels to the support lattice were observed. All cap screws and welds that were observed are in good condition.

The upper guide channel bracing was inspected with the binoculars for warpage, loose cap screws and cracking. A cap screw holding an angle brace to No. 6 rod guide channel was missing. The screw appeared to be twisted off. A letter from R Scanlan to R Michel dated April 6, 1965, describes this screw and location. This screw was twisted off in an attempt to tighten it before installation of the holddown. The letter also states that the remainder of the screw is sufficiently welded into the guide tube. No other abnormalities were observed on the bracing.

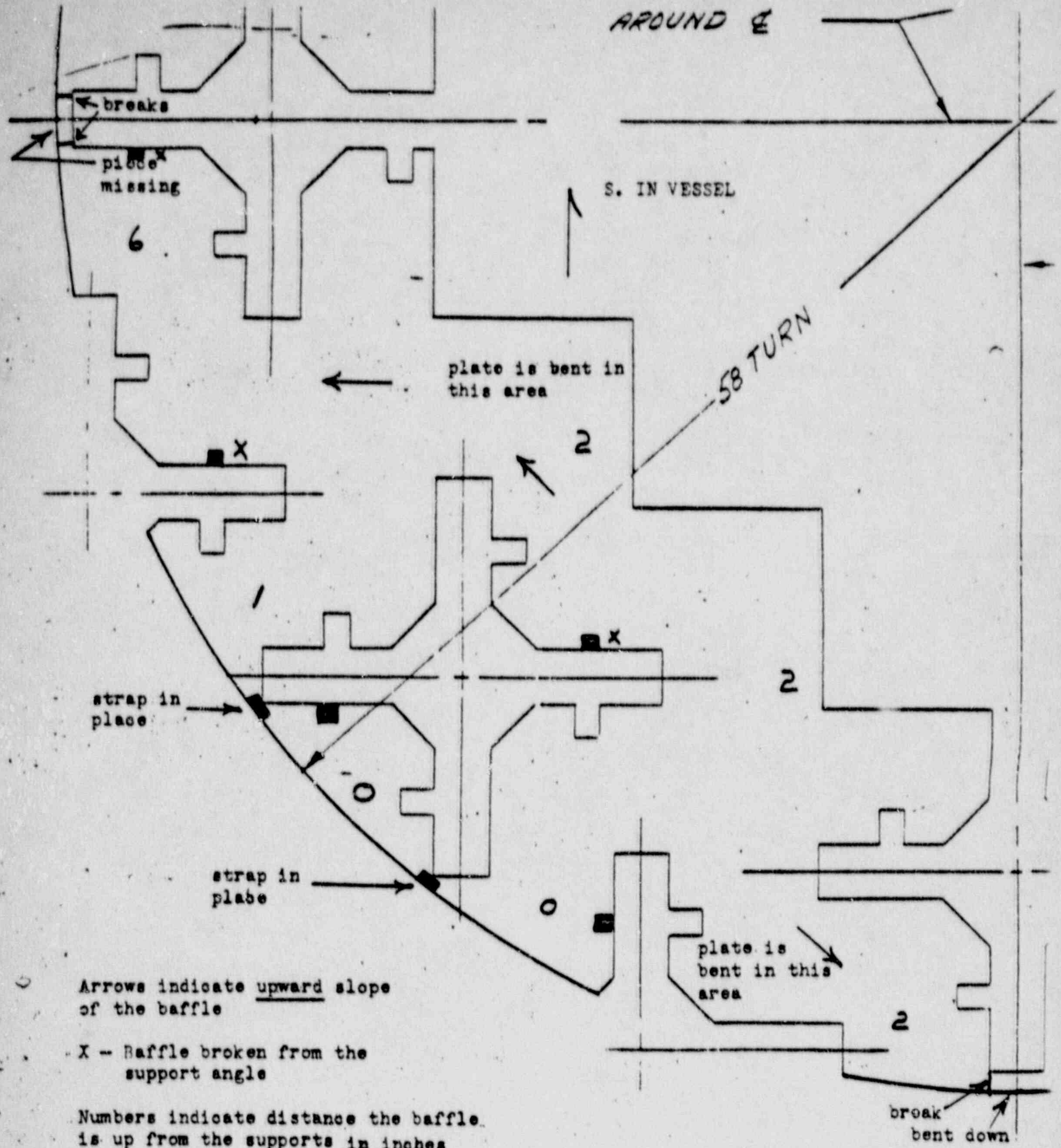
The lugs and washers were inspected for cracks or breakage. No evidence of any damage or movement of the lugs was observed. The "C" washer locking pins are in place and no difficuity was experienced in unlocking the washers.

Map of Steam Separators



LOCATION OF THE TRIANGULAR SHAPED PIECES
FROM THE LOWER BAFFLE FOUND
ON THE SEPARATOR SHELF

FIGURE I



Arrows indicate <u>upward</u> slope of the baffle

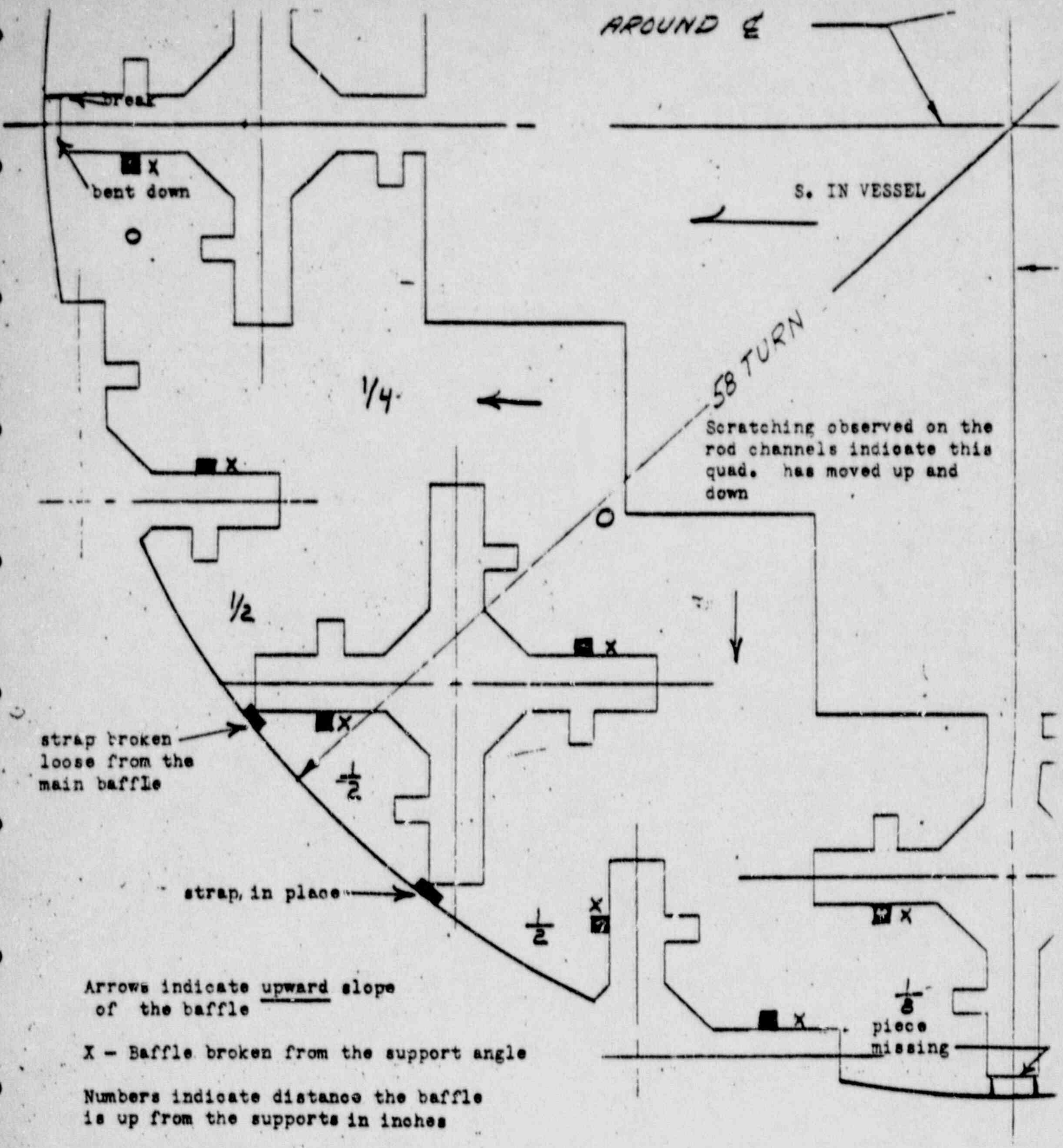
X – Baffle broken from the support angle

Numbers indicate distance the baffle is up from the supports in inches

Scale – 3" = 12"

<u>N.E. Quadrent of Lower Baffle</u>

<u>FIGURE II</u>



S.E. Quadrent of Lower Baffle

FIGURE III

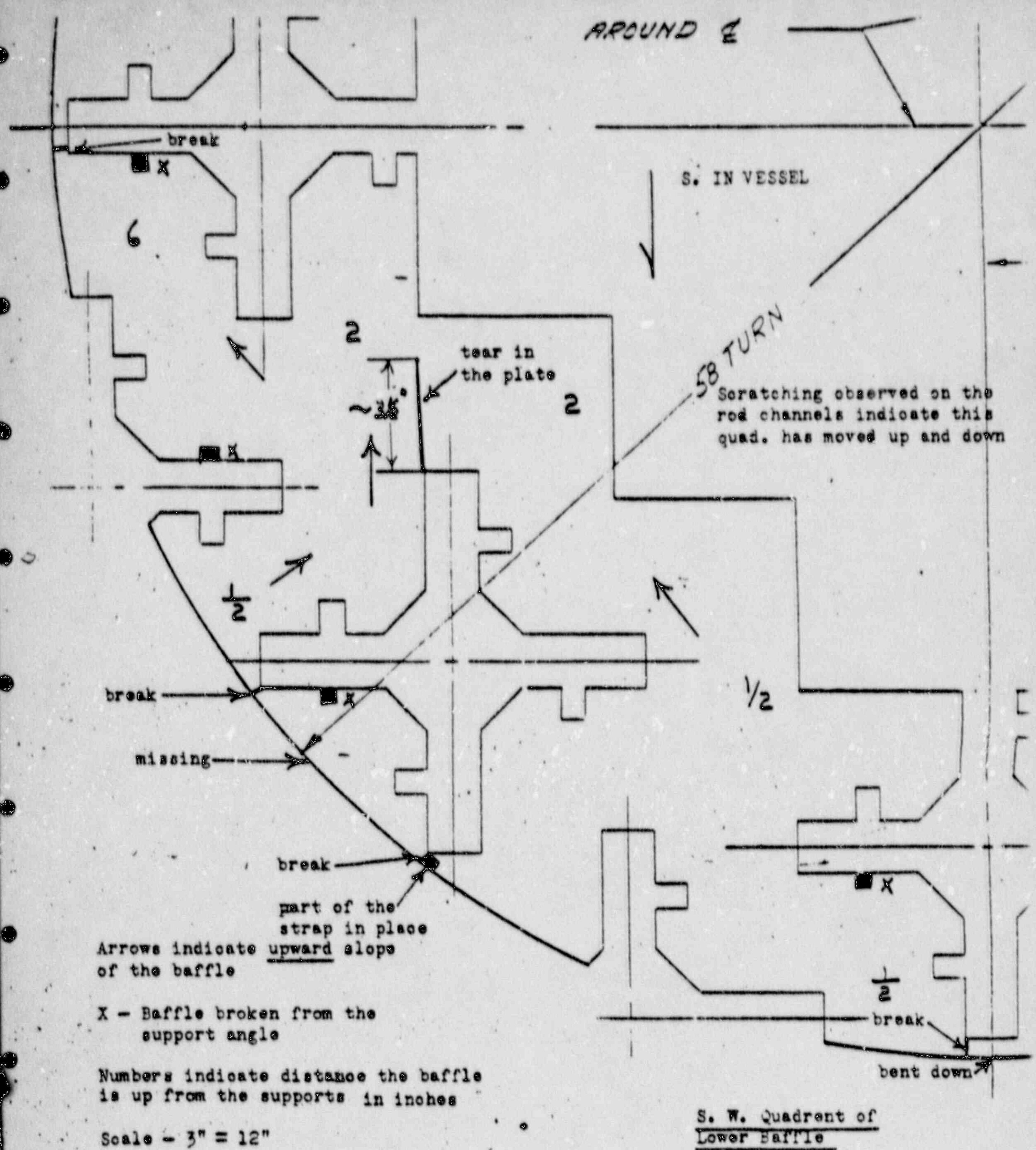
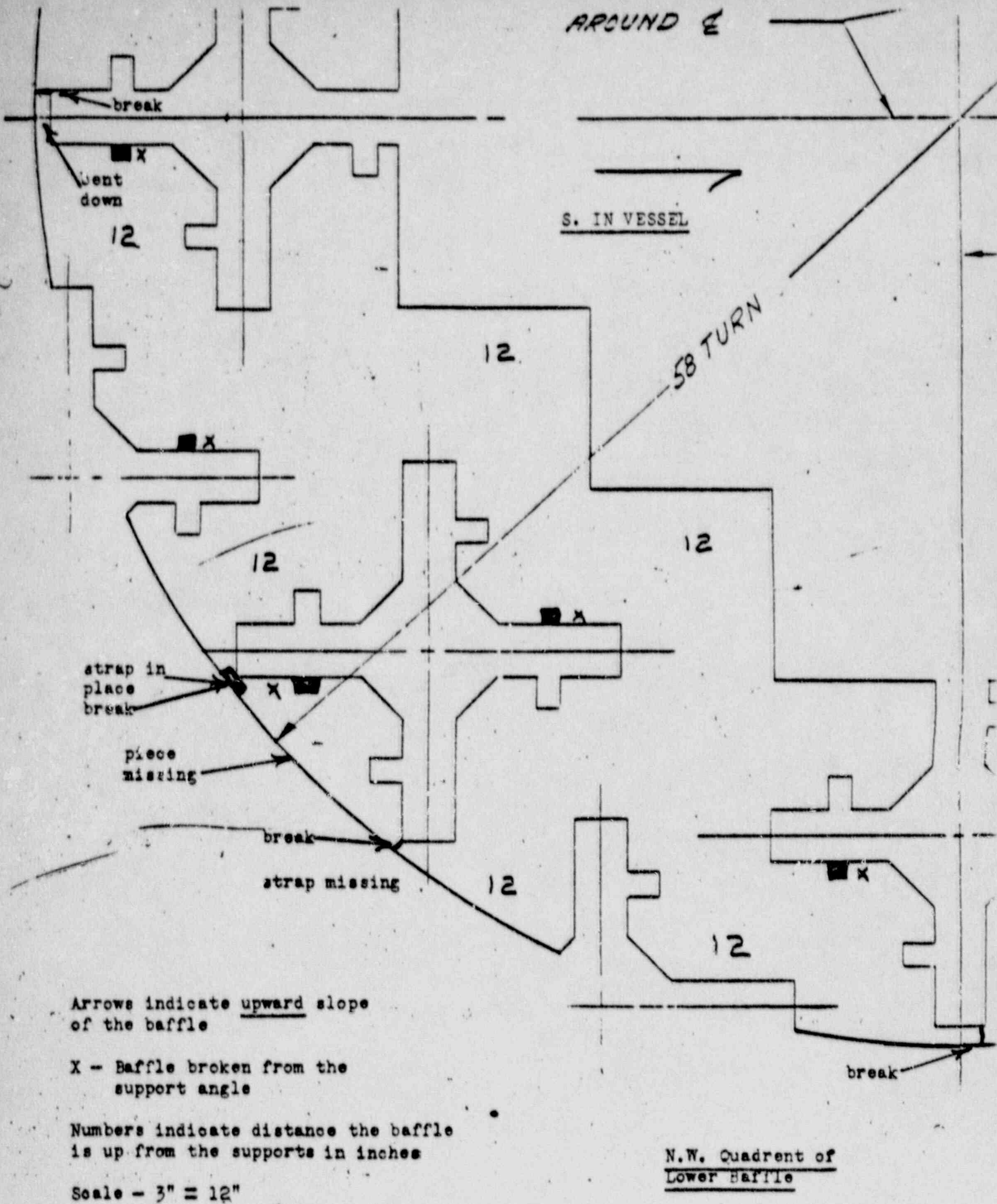
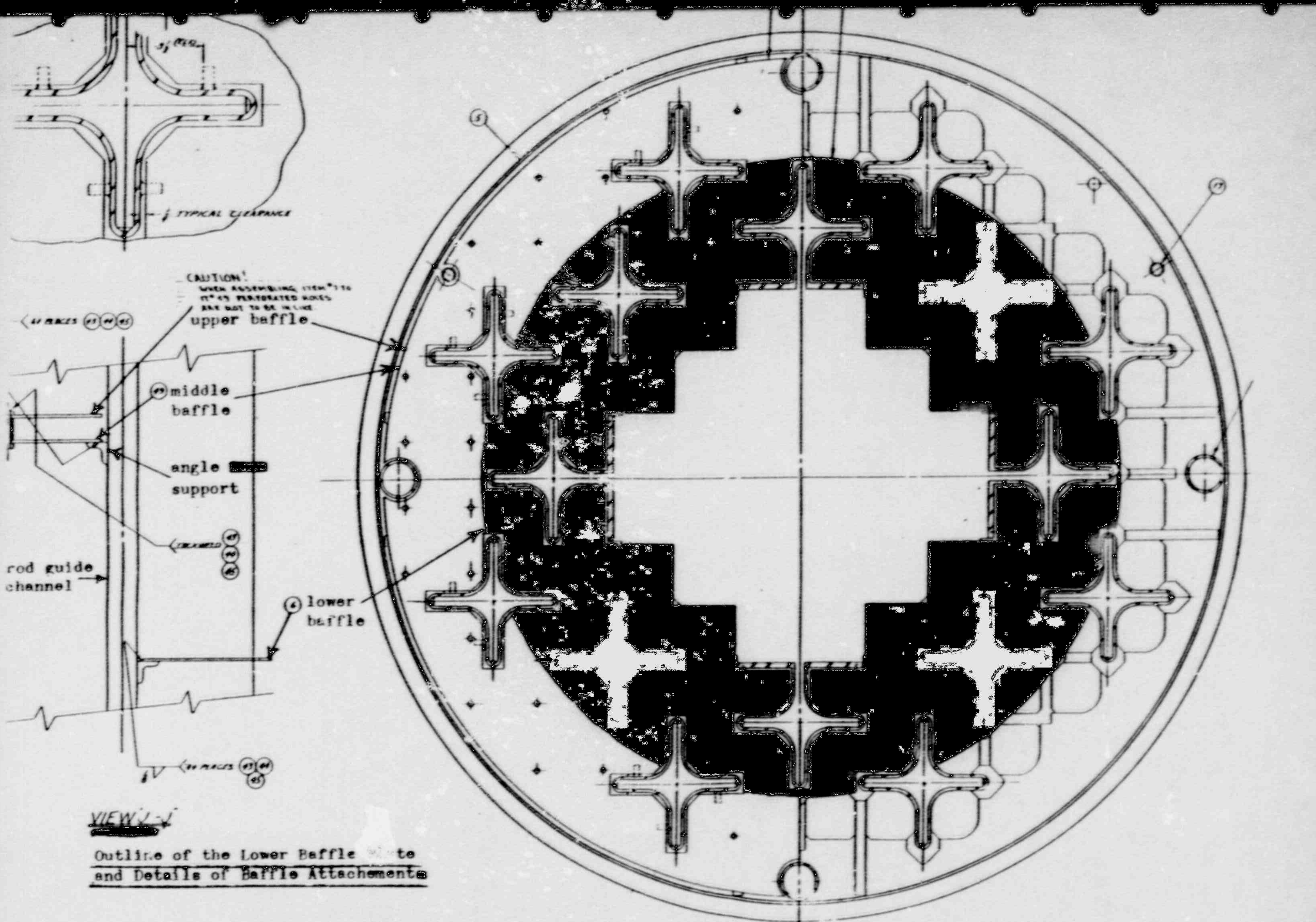

AROUND ℄
break
S. IN VESSEL
6
2
tear in the plate
2
Scratching observed on the rod channels indicate this quad. has moved up and down
1/2
break
1/2
missing
break
part of the strap in place
1/2
break
bent down
Arrows indicate upward slope of the baffle
X - Baffle broken from the support angle
Numbers indicate distance the baffle is up from the supports in inches
Scale - 3" = 12"
S. W. Quadrent of Lower Baffle

FIGURE IV



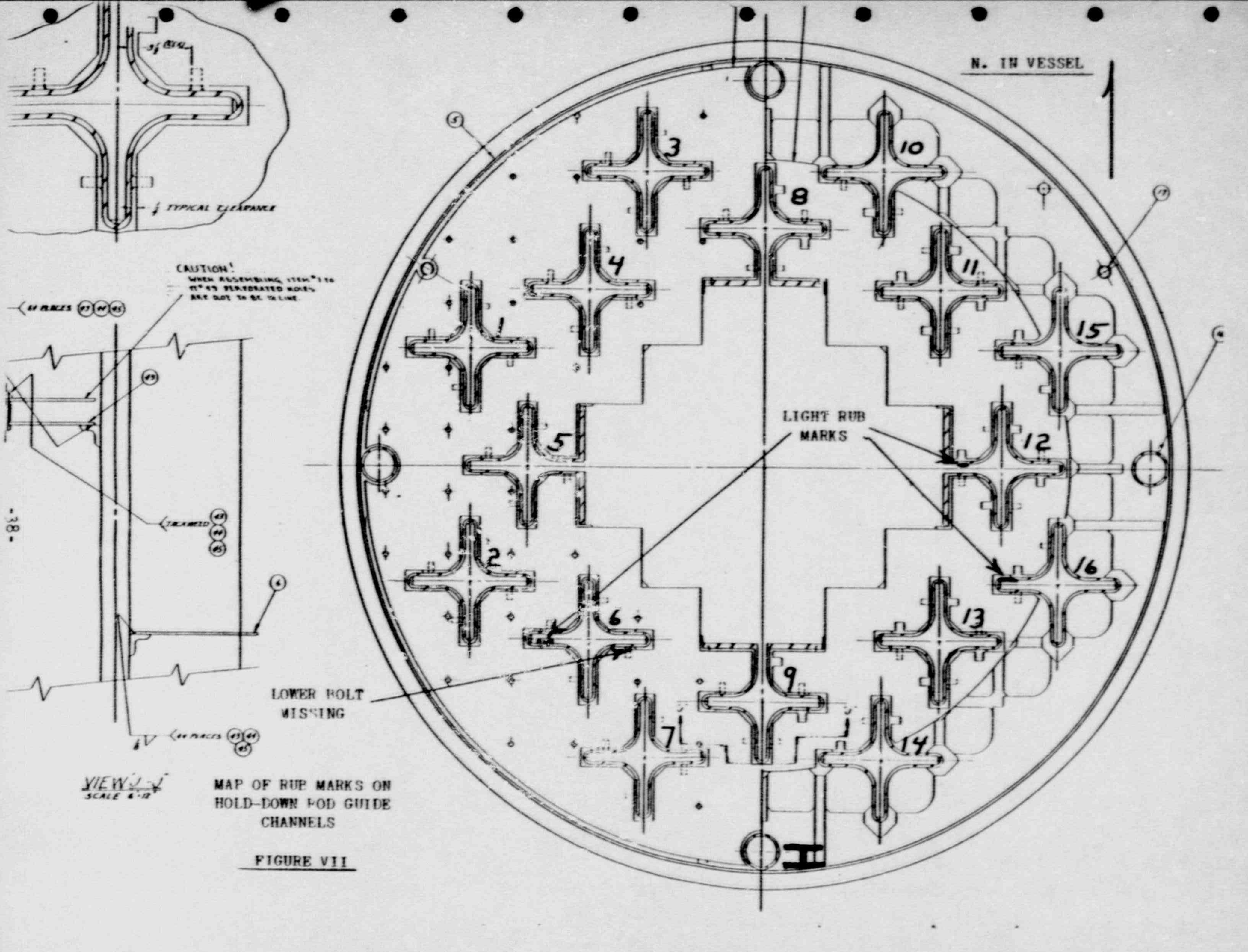
N.W. Quadrent of Lower Baffle

FIGURE V



Outline of the Lower Baffle Plate
and Details of Baffle Attachements

FIGURE VI



MAP OF RUB MARKS ON HOLD-DOWN ROD GUIDE CHANNELS

FIGURE VII

H. Process Tubes

The outer row of tubes was inspected around the entire perimeter of the superheater, excluding the north, south, east, and west flats using the underwater periscope. The steam separator pieces found laying in the reactor vessel had come in contact with the process tubes. The effect of this contact on the thin process tubes, .026 inch wall, could have a direct bearing on the overall operability of the reactor.

The purpose of the process tube inspection was to determine the condition of the outer surface of the process tubes. The inspection involved 76 tubes, 19 each in the NE, SE, SW, and NW corners of the superheater fuel array. It was not possible to inspect more than the outer row of tubes in each corner because of the size and maneuverability of the periscope and attached lights and mirror. The curved dish supporting the process tubes restricted the vertical distance over which the tubes could be inspected. Also, it was not possible to inspect tubes on the north, south, east, or west sides because of a protective support plate outside of these tubes. With the periscope viewing perpendicular to the tube axes, inspection was possible along 13 inches of each tube, from the bottom of the superheater lower baffle to the superheater lower grid plate flange. Then, with a mirror attached below the viewing lens of the periscope, it was possible to view the remainder of the process tubes' lengths; i.e., from the lower grid plate flange to the junction of the tubes and the dish.

Small scratches were found on some of the 19 tubes in each corner of the superheater. Near the top, i.e., immediately below the lower baffle, a scratch about 2 inches x 1/32 inch was found on tube W-9. Scratches were found about 9 inches down from the lower baffle on tubes Z-1, Z-2, Z-3, and Z-4. Scratches about 2 inches long were seen on tubes N-1 and N-10, flush with the bolts on the flange of the lower superheater grid plate. Scratches were also seen on Z-16 and Z-18, approximately 8 to 10 inches below the lower superheater baffle, the scrape on Z-18 extending down to the dish.

On Z-17 a deeper indentation was found 2 to 4 inches above the lower grid plate flange. Tube N-1 had a similar dent about 6 inches below the lower baffle with a brownish dark spot in the center.

About 2 inches above the flange, tube S-1 showed a deep indentation which was black in the center, (identified as such because three dimensions were observed when the lighting caused a shadow of one "wall" of the indentation on the other). It extended about 1 inch in length and was approximately 0.1 inches wide.

In the NE corner tubes A-1 and A-2 had dull, silvery spots about 0.1 to 0.2 inches in size on the east side 1 inch above the flange.

The longest scratches seen extending about 4 to 6 inches above the lower grid plate. All others were comparatively shorter and no deeper or wider. The largest number of scratch marks was found to be in the NW corner. The largest number of metal pieces that collected in the dish and grid plate were in that corner. In general, the scratches appeared silvery and about 0.05 inches wide. They appeared to have been caused by a single, slight contact with another metallic object. Apparently, the contact occurred with little force.

The indentations observed on tubes Z-17, N-1, A-1, and A-2 were about 1/8 x 1/8 inch and looked deeper than the scratches described above. They were dents rather than scratches, and were apparently the result of contact with the point of some metallic object.

The mark on tube S-1 appeared to have depth because three dimensions were observed when the lighting caused a shadow of one "wall" of the identation on the other.

It should be emphasized that the alignment of the periscope, mirror, and lights vastly affected the view afforded by this equipment. By turning 10 degrees in either direction from the perpendicular to a certain tube surface, one lost sight of or saw far more or less of the particular deformation. Consequently, it is possible that marks exist which were not viewed. However, it is improbable that any such marks would be deep and/or long, or else they would have been seen from some angle. Also, the tube configuration prohibits the inspection of the majority of tubes. Metallic pieces of major size could not fit between the tubes. The tube spacing is 0.43 inch between tubes in the middle of the superheater. Pieces of approximately 8 to 12 in^2 were removed from the dish four rows west of the east support plate and four rows north of the south support plate.

The size of these pieces makes it improbable that they could have caused any damage in the area where they were found. Further, because the flow of coolant water decreases drastically as it approaches the middle of the superheater, it is not likely that pieces could have worked in any further toward the center.

1. <u>Grid Plate</u>

The inspection included a close-up view of the top and bottom surfaces, the outer edge, the inside chimney, and the fuel and quad box coolant flow channels. A gross inspection was first performed from the bridge using binoculars with the grid near the shield pool floor. Then the underwater periscope was lowered into position to scan the grid plate surfaces. The borescope was used to look inside 10 of the 96 fuel element flow channels and 3 of the quad box flow holes. A mirror and light arrangement facilitated inspection of the lower surfaces.

The upper surfaces showed some minor scratches around several element flow channels. The lower surface had rust spots and/or scratches on approximately 10 channels, with other scrape marks between holes. One group of four slight scrape marks was found on the edge of the grid. All eight grid plate coolant channels around the perimeter were fully open and unmarked.

The inside of flow channels No. 28, 88, 89, 90, and 228 revealed nothing unusual. Small nicks were seen on the top edge of the lower machined surface inside these thimbles. No. 210 showed several small rust spots on the lower machined surface. No. 20 had a small, silvery-looking nick in the middle section between the machined surfaces. No. 91 and 203 each had a pock mark in the middle section which appeared rusted out about 10 - 20 mils deep with a blackish center. Inspection of No. 204 showed a definite dent roughly 1/2 x 1/16 x 1/16 inch on the upper machined surface. The lower machined surface was partially discolored as if it were burned or brazed.

Quad box coolant channels for control rods No. 8, 9, and 12 were rusted but not to the point of restricting coolant flow.

The overall inspection revealed nothing that was unexpected. Scratching and resultant rusting on the outer surfaces was a consequence of pulling the grid plate out of the reactor and moving it around. Significantly, no cracking was detected. Nicks and dents inside the element flow channels can be traced to loading and unloading the fuel and changing the position of the various elements. The gray discoloration of the lower surface is probably a heat effect. The visible rusty scratches probably were caused by pieces of the steam separators bumping against the lower side of the grid plate. There is no other way that this portion of the grid plate comes in contact with any metal. Ten of the 96 fuel element channels were inspected, these were in good condition and exhibited no buildup of crud.

Quad box flow channels exhibited corrosion and apparent scaling when viewed from above. These holes have a smaller diameter than the borescope head, prohibiting a closer view of the inside of the channels. This rusting is apparently a result of a small flow rate compared to that of a fuel element hole. The quad box flow channel for outer boiler control rod No. 8. There is no sleeve or thimble through the quad box flow channel as there is in each fuel element flow channel. Each fuel element nozzle projects below the bottom of the grid plate, whereas the quad box coolant channel nozzle goes only part way through the grid plate, leaving a cavity between the nozzle and the lower surface of the grid plate. Also, there is a cavity below the 20 flow holes that direct coolant around the cruciform control rods. Thus, there are two places in which flow would be slower than average and in which crud could deposit.

J. Steam Separators

As part of the study of recirculation flow changes during the shutdown of September 16, 1967, No. 12 recirculation pump was pulled for inspection. A large piece of metal was found wrapped around a diffuser vane of the pump. This piece was identified as a vane from a steam separator discharge nozzle. This report covers the subsequent inspection and investigation of the steam separators.

When the piece of separator was found it was decided to remove the boiler core and the shroud to facilitate a complete inspection of the separator discharge nozzles. This inspection revealed a general failure of the nozzle vanes. An investigation into the cause of the failure was initiated by Allis-Chalmers and resulted in the nozzles of separator group No. 12 being sawed off and sent to Battelle Memorial Institute for hot cell investigation.

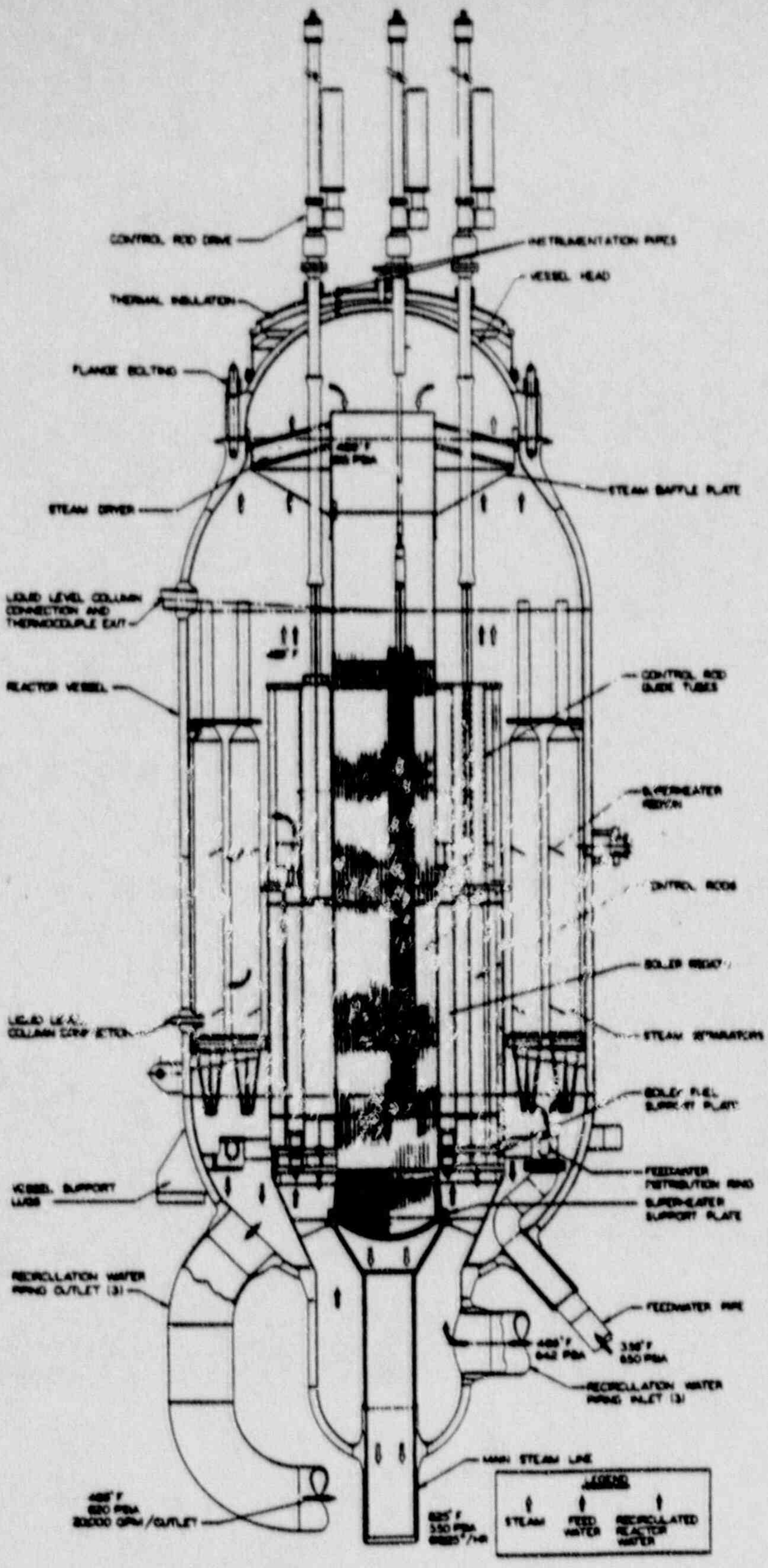
The remaining 14 separator groups were then removed from the reactor using the special lifting rig developed on site. As each group was removed it was brought to within approximately 4 feet of the surface of the shield pool for inspection and measurements of the areas on the nozzles where the vanes are missing. After being inspected the separators were stored in the shield pool.

The 44 separators are of the centrifugal type and are arranged around the outside of the boiler region in the downcomer space. They are designed to remove entrained steam from the recirculation water and prevent loss of pump N.P.S.H. The separators are constructed of .109 inch type 304 stainless steel. The body is 10 inches in diameter and 8-1/2 feet long. The steam-water mixture enters the cylinder through a tangential inlet measuring 1.5 inches wide and 4-1/2 feet long. A conical lower outlet nozzle contains the vortex formed and a 4 inch diameter pipe exhausts steam upward to the reactor steam dome. These separators are placed in the reactor in groups of three except for one two-element group near the water column connection. The groups are held in the separator shelf by their own weight and steadied by pins that attach to brackets on the vessel wall. Figures I and II, page 44 and page 45 show the locations of the separators and the numbering system used.

Because of the long period of time covered in this report, the following sections are listed with their dates in the order they occurred.

June 1967 - September 16, 1967 Recirculation Flow Changes

During this period recirculation flow changes were noticed during operations. Considerable efforts were expended to explain these flow changes. Because of these unexplained flow changes it was decided to pull number 12 recirculation pump for inspection during the scheduled September shutdown for shim removal.



Controlled recirculation boiling reactor (CRBR) with nuclear superheater for Path‑inder Atomic Power Plant. (206530)

FIGURE 1

MAP OF THE STEAM SEPARATORS

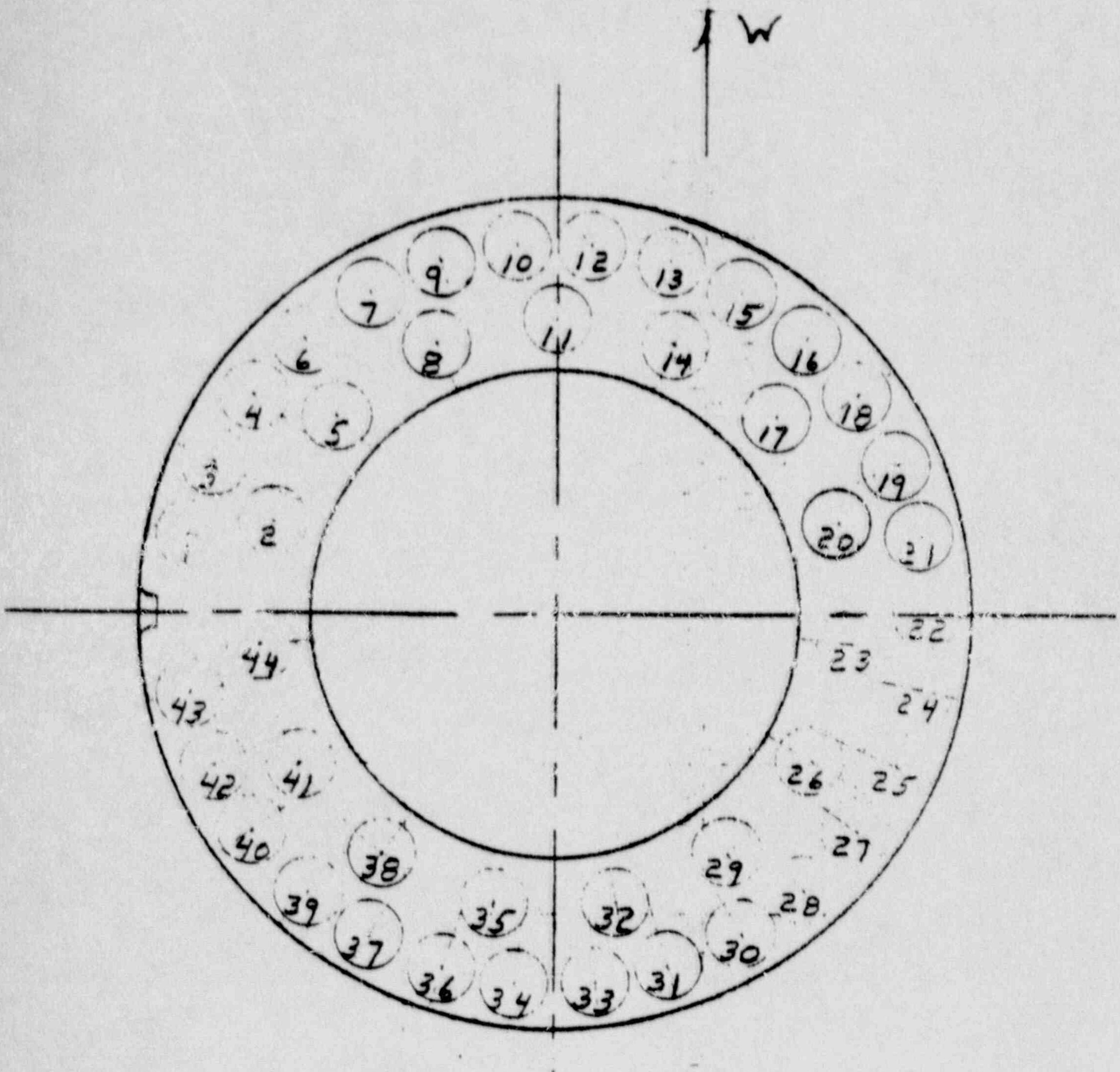


FIGURE II

October 18 - No. 12 Recirc Pump Pulled

Recirc pump #12 was pulled and a 12 inch long trapezoidal sheet of metal 5 inches wide at one end and 2.5 inches at the other was found bent around a vane in the diffuser section of the pump. This piece was identified as part of a vane from the outlet nozzle of a steam separator and was the first indication of a failure of the separators.

The Operations Committee decided all of the steam separator nozzles must be inspected. To inspect these nozzles the boiler fuel, poison shims, boiler control rods, boiler fuel boxes, and core shroud had to be removed.

November 8 - Piece of Separator Vane Found in Fuel Element

When boiler fuel element No. 223 from core location R-BB-9 was inspected, a 2-1/2 x 2 inch piece of sheet metal was found in the nozzle of the element. This piece was retrieved and was found to coincide with part of a steam separator vane. A more complete discussion of this separator piece can be found in the boiler fuel inspection report.

November 21 - Periscope Inspection Under the Grid Plate

After the last of the boiler boxes were removed, an inspection was conducted below the grid plate. Five large pieces of separator nozzles were observed lodged near the lower support plate of the superheater structure. This was the first positive indication that there was a gross failure of the nozzles and that the piece in the pump was not just an isolated case of vane failure.

December 2 - Shroud Removal and Initial Nozzle Inspection

The shroud was removed and the separator nozzles inspected with the periscope. This inspection revealed that of the 44 nozzles, 10 had vanes missing, one had a piece of a vane missing, and 8 other nozzles, had fractures at the bottom of the vanes. (A later inspection after removal of the separators revealed that the number of failures was greater than this initial inspection revealed.)

December 7 - Removal of the First Separator Group and Inspection

Steam separator group #12 consisting of separators No. 34, 35, and 36 was removed from the vessel and placed in a horizontal position in the shield pool for inspection. Group #12 was selected because it had one nozzle that appeared intact, one nozzle with pieces of vanes missing and one nozzle with visible cracks in the vanes.

December 11 - Site Meeting with Battelle Columbus

At this meeting it was decided to saw off the three nozzles of the group #12 separators and send them to Battelle's hot cell lab for examination and, if possible, determination of the cause of failure.

January 11, 1968 - Shipment of Nozzles

Battelle's cask arrived and the nozzles were loaded after having been cut off with a modified control rod extension cut-off saw. The upper parts of the group 12 separators were inspected and no abnormalities were found. The 2 inch area between the support ring and the attachment weld on the nozzles had no indication of cracking. The nozzles were cut off 3/8 inch above the weld.

March 6 - Additional Separator Samples

The triangular pieces of the separators that are used to make up the inlet were thought to be from the same sheet stock as the nozzles. It was therefore decided to take samples of these triangular areas from one separator. Three samples were taken out of No. 34 separator with a 3/4 inch hole saw, one from the upper triangular piece, one from the side of the inlet, and on a piece out of the cone for comparison. These samples were loaded in a cask and sent to Battelle Labs for analysis.

March 19 - Steam Separator Removal and Inspection

The remaining 14 groups of separators were removed from the vessel and stored in the shield pool. As each group of separators was removed it was inspected and the area of missing vanes measured. The inspection of the separators consisted of the following:

1. Determination of the total number of vanes and pieces of vanes that are missing.

2. Inspection of the area above the nozzle welds for cracking or any discolorations.

3. Inspection of the pins that hold the separator groups to the vessel wall for wear or cracking.

4. Inspection of the triangular area above and below the inlet opening of the separator.

5. Inspection of the main part of the separators for any conditions that would make them unsuitable for further use.

Table I, page 49, gives the findings of the inspection of the nozzles. The "separator intact" column means that there were no pieces missing from the nozzle. The "vane missing" and "piece missing" column mean that an entire vane was missing or a piece was missing from a vane. The vanes that remained on the separators were generally cracked with many vanes being loose on the bottom. Most of the lower breaks occurred approximately 1 inch up from the bottom of the vane. On the vanes that were missing the upper breaks are in the area between the stress-relief holes and the attachment weld. This area fits through the separator shelf. There was no indication of separator movement vertically or horizontally. The report on the accounting of the missing pieces gives more detailed findings as to the location of the breaks.

Above the nozzle attachment welds there was no visual indication of cracking or surface attack. The pins that hold the separators to the vessel wall were all inspected and found in good condition. The pins are shiny but have no metal missing or shoulders formed. The pins and the plates that the pins are attached to show no indications of cracking. Also there was no indication on the pins of vertical separator movement.

The upper body of the separators were in generally good condition. The only abnormalities found were marks from the neutron windows. For a detailed discussion of these wear marks see the Inspection Report on the Neutron Windows.

Other than the wear marks from the neutron windows there were no abnormalities found on the separators above the nozzle attachment welds.

The Battelle investigation has been completed but the final report has not been received. The findings of this investigation are summarized in the following statements from the March 27, 1968, Safety Committee Meeting Minutes.

> "Mr Pearce stated that Battelle has found evidence of general intergranular corrosion on the nozzle vanes. There is no evidence of corrosion anywhere else on the separators. A search of the separator fabrication records revealed that a change was made that allowed hot working of the nozzle vanes. It is believed that this heat treatment caused sensitization of the nozzle vanes to corrosion. However, it cannot be stated that the vanes would not have failed without the previous corrosion sensitization. Mr Haines said that probably a combination of fatigue, and stress corrosion all played a part in the failure of the nozzle vanes."

TABLE I

TABULATION OF SEPARATOR INSPECTION RESULTS

Sep. No.	Separator Intact	Vane Missing	Piece Missing	Sep. No.	Separator Intact	Vane Missing	Piece Missing
1		X		23	X		
2	X			24	X		
3		X		25		X	
4	X			26			
5				27		X	
6				28	X		
7	X			29			
8	X			30		X	X
9	X			31		XX	
10		X		32	X		
11	X			33		X	
12		X		34		X	X
13	X			35			
14	X			36	X		
15				37		X	
16		X		38			
17	X			39	X		
18	X			40	X		
19	X			41	X		
20				42	X		
21	X			43	X		
22	X			44		X	
				TOTAL	23	14	2

K. Piece Accounting

This report is divided into three sections. The first deals with the location and retrieval of the separator pieces; the second describes the identifying and measuring of the found pieces; and the third section describes the actual accountability effort carried out on the separator vanes. This report only covers the accounting efforts and results that have taken place prior to the end of the six month reporting date - May 19, 1968. The accountability has not been completed and no conclusion has been included for this report.

Location and Retrieval of the Steam Separator Pieces

As of May 19th, 46 pieces have been located and retrieved. Their actual locations were as follows:

1 piece	No. 1 was found wrapped around a stay vane in No. 12 recirc pump diffuser section. (Pieces numbered in order of recovery.)
1 piece	No. 2 was resting in a boiler fuel element inlet nozzle (Fig. 4, page 60).
1 piece	No. 20 was found in the vessel in the dead region of the recirc pump inlet plenum (Fig. 2, page 58).
2 pieces	No. 45 and No. 46 were resting between the superheater outer insulating tubes.
2 pieces	No. 14 and No. 39 were on the supporting struts between the superheater supporting dish and the boiler fuel grid plate support ring (Fig. 1, page 57).
39 pieces	All except the above, were lying in the superheater supporting dish.

A detailed account of the finding and retrieving of the pieces is given in the following paragraphs.

The piece from the recirc pump:

The first separator vane piece was unexpectedly found when No. 12 recirc pump was pulled for inspection. It was the first indication of any separator nozzle failure. The piece was retrieved by hand with no difficulty.

The piece from the boiler fuel element:

The second piece was found while removing the boiler fuel assemblies. The position of the piece in the nozzle is illustrated in Figure 4, page 60. The piece was knocked out of the nozzle into a bucket with the "hook" tool. No other pieces were located during fuel removal.

The pieces on the superheater chimney supporting struts and the lower dish:

After the boiler fuel and boiler boxes were removed, the periscope was used to view below the grid plate. The grid plate was removed and after doing so a large aggregate of vane pieces were found lying in the superheater support dish.

Two pieces were also found resting on the supporting struts between the dish and the grid plate support ring. The largest piece appeared to be hooked under the edge of the dish and resting edgewise across one of the struts. It was decided to remove this piece with the Ameray tool.

The other piece on a support strut and most of the pieces in the dish were either too small or positioned so that the Ameray tool could not be used to grasp them. A rubber suction nozzle was made for the underwater vacuum cleaner. A special basket was also made to fit snuggly just below the dish and between the struts. Both the pieces retrieved by the Ameray tool and by the suction head on the vacuum were then released into this basket and brought up out of the vessel. They were then transferred to plastic pails hanging on the side of the shield pool.

Some difficulty was encountered in trying to retrieve some of the pieces lying in the dish because they were partially between the outer superheater insulating tubes. Care was taken so as not to damage any of the tubes.

The pieces between the superheater outer insulating tubes:

After removing all the pieces from the dish, a thorough inspection was made to locate any pieces that might be resting between the superheater insulating tubes. A special mirror attachment was fastened to the periscope to permit viewing between the tubes running down into the dish. Several suspected pieces were spotted, but because the gaps between the tubes were only about 1/2 of an inch wide, a special tool had to be devised in order to retrieve them.

The tool was similar to a crevice cleaning attachment for a vacuum cleaner. This was attached to the discharge of the underwater vacuum and was used to "blow" the pieces from between the tubes out into the dish. The suction tool was then used to retrieve them. Two pieces were recovered by this method - No. 45 and No. 46. They were the only two pieces found between the tubes.

The piece from the recirc pump inlet plenum:

The initial periscope inspection revealed a piece - No. 20, on the bottom of the recirc pump inlet plenum (Fig. 2, page 58).

This piece was removed by the use of the vacuum suction head. A thorough inspection of the region found that piece to be the only one located there.

The possibility of other pieces:

A complete inspection of the rest of the vessel internals found no other pieces. It is believed that all the pieces that accumulated in the vessel have been removed.

The Identifying of the Pieces and the Missing Areas from the Separator Vanes

The identification of the pieces was done by taking accurate size and radiation measurements of each piece. The identification of the missing areas from the nozzles involved only the measurement of the amounts missing. The identification process is described in the following paragraphs.

The condition of the pieces:

Most of the pieces were badly deformed and would require some straightening in order to allow the quick measurement necessitated by the possible high radiation exposure.

The straightening of the pieces:

The pieces were straightened on a piece of steel plate to which angle was welded on two sides to "hook" the pieces. The piece could either be pounded flat with a steel maul or by using a crowbar type tool "pried" flat. The plate was placed in a truncated 55 gallon drum under 24 inches of water for the pieces with high radiation levels, and on the floor for pieces not so radioactively "hot". The time required to straighten each piece ran about 10 to 30 seconds.

The radiation level and numbering of the pieces:

A radiation reading for each piece was taken at contact as the piece was taken from the pail before straightening. After straightening each piece was numbered with a crayon. The number of each piece along with its radiation level is given in Figures 6 through 21, pages 62 through 71.

The measuring of each piece:

The radiation levels permitted most of the pieces to be traced directly onto a piece of paper; only five pieces, Nos. 2, 8, 14, 43, and 44 had to be measured underwater. The detail of the measured pieces is not as accurate as that of the traced pieces.

The accuracy of piece measurements:

The straightening of the pieces was not complete. It would have been very impracticable, if not impossible, to pound the pieces perfectly flat; consequently, error was introduced both in tracing and measuring due to surface irregularities. For the most part these errors would be negative, e.g., resulting in smaller measurements than actually existed. In the tracings, a "pencil parallax error" was also introduced by trying to follow the wavy edge. This could be an either positive or negative error.

Some of the smaller and badly twisted pieces might have been deformed slightly during the straightening process. This might result in a slight mismatch between some of the adjoining pieces.

All errors considered, the measurements and tracings can be considered reliable up to ± 1/8 inch. However, the detail on the five non-traced pieces might exceed this limit slightly.

The measurement of the separators:

The exact amount of material missing from the separators was determined by direct measurement. Due to the radiation levels encountered, the separators could only be brought to within a few feet of the shield pool surface. A special measuring tool was therefore constructed out of a piece of metal tape mounted on a frame perpendicular to a six foot handle. The scale could be read accurately to 1/16 of an inch.

The details of the cracked off edges could only be approximated and sketched with the aid of some reference measurements.

The accuracy of the separator measurements:

Initially in the measurements of the separators, the distances from the flange to the weld, the weld to the hole, and the hole to the nozzle end were taken to be constant and equal, to 2, 4/8 and 23-5/8 inches, respectively. (Fig. 3, page 61.) Only after some of the separators were measured was it discovered that two of those dimensions were not constant. The distance from the weld to the hole varied from 3/4 to 1-1/8 inches and the distance from the hole to the nozzle end varied from 23-3/16 to 23-9/16 inches. This variance could result in an error of up to 3/8 of an inch in the measured vane length. The distance from the weld to the hole was not constant for some vanes.

The largest loss of accuracy in the separator vane measurements, however, was associated with the approximation of the detail at the cracked off edges. The natural curvature in the vanes, along with the lack of good depth perception caused by the viewing through the water, makes these approximations quite difficult. It is possible that in some cases these approximations could be off by as much as 1/4 inch.

When trying to match the top of the vane piece with the intact part of the vane, the detail of the piece was considered to be more reliable than that of the intact portion.

The Accounting for the Separator Vanes

The matching of the pieces with each other and with the missing areas from nozzles:

Once all the separator nozzles had been inspected and the missing portions measured, the actual accounting of the pieces could take place. The measurements taken of the missing nozzle portions were used to make full size drawings of the areas missing. The measurements of the pieces were used to make full size cutouts of each of them. The cutouts were then matched together and with the missing section of the nozzles. Figures 6 through 21, pages 62 through 77, show the missing pieces placed within the missing nozzle areas.

The darkened portions represent areas or parts missing. The lined pieces off to the right of these areas are their actual sizes taken from the full sized diagrams. No measuring error was taken into account when drawing these pieces. They represent the maximum amounts that could be missing. The lined area on the vane reproduction itself is the intact portion of the vane. The pieces retrieved are outlined and numbered.

For the most part the pieces fit together quite well. The major mismatches occur where the pieces are to meet the intact areas of the nozzles. This could be due to the difficulty encountered in measuring the detail of the broken edges on the intact portion of the nozzles.

There is also evidence to indicate that the larger pieces were "roughed up" more than the smaller ones while going through the recirc pumps. It could be expected then, that on the larger ends of the vane pieces a few missing corners or chunks would be found. The general condition of the pieces when found indicate they took quite a "beating". Where only small pieces with easily defined borders were involved (Figures 14 and 19, pages 70 and 75) the match was very good.

Accounting for the darkened "ground off" area of the diagrams:

Some of the darkened areas are large enough to suspect that they might have been broken off as significant pieces. The vessel internal inspection however found very few small pieces, and it is doubtful if the rest of these missing areas will be found in piece form. Substantial amounts of scaled corrosion products were found on the vessel bottom and evidence in the form of "shiny" areas on the vessel wall, was found to indicate that many of these chunks might have been eroded away in the turbulant waters present in the recirc pump inlet plenum.

There are however, four large areas still missing that can be expected to be found as significant pieces. They are labeled as missing pieces on diagrams 8, 9, 10, and 12.

The possible locations of the pieces still missing:

It is believed that all the pieces have been removed from the vessel. The only other probable locations for the four pieces still missing are the recirculation loops of the pumps or the pumps themselves (Fig. 3). No. 12 recirc pump has been pulled and inspected. No. 11 pump has been pulled and has been partially inspected. If the pieces are not found in No. 11 or No. 13 recirc pump, it will be assumed they are in the pipe loops.

From Figure 1, one can see that the still missing pieces come from separator located over No. 11 and No. 13 recirc pump suctions. (The separator numbers with "m" suffixes are the ones with a piece still missing). One might expect to find them in those pumps if as they were broken off, they fell into the closest pump suction.

This thinking however, it not supported if some recognizable pieces are picked out of the picture of the superheater dish "as found" and compared with the separator from which they came. This is done on Figure 1. The numbers with the "p" suffixes are located where the pieces were found, and the number is that of the mother separator. It can be seen that some pieces are on the opposite side of the core from where they might be expected. It is possible however that this "mixing" took place after the pieces went through the recirc pumps.

If any of the pieces are still unaccounted for after the three recirc pumps are inspected, the loops will have to be checked in an attempt to retrieve pieces. The tentative plan is to run a pump for a short period of time and then inspect the vessel for any pieces that might have been flushed out. The order of recirc pump shutdown after the September 16th scram gives some indication as to which loop the pieces might be found in.

While any recirc pump is running the other recirc pump discharge and inlet valves are open about 6%. With one pump running, there is backflow through the other loops. The recirc pump shutdown starting with September 16th was as follows:

Sept 16th	Before scram	All 3 pumps on
	19:17	No. 11 and 12 OFF
	19:35	No. 12 pump ON and No. 13 pump OFF
Sept 17th	06:16	No. 12 pump OFF

No. 12 pump was run intermittently throughout the day.

It can be seen then that No. 12 pump was the last recirc pump running after the shutdown with No. 13 pump the next to last. If it is assumed that while a pump is running enough flow exists in the loop to wash the pieces out, No. 12 loop should be void of pieces whereas the other loops having some backflow may of had some pieces washed into them. The backflow velocity perhaps was not sufficient to carry them through the loop.

Relation of the Separators to the Recirculation Pump Suctions

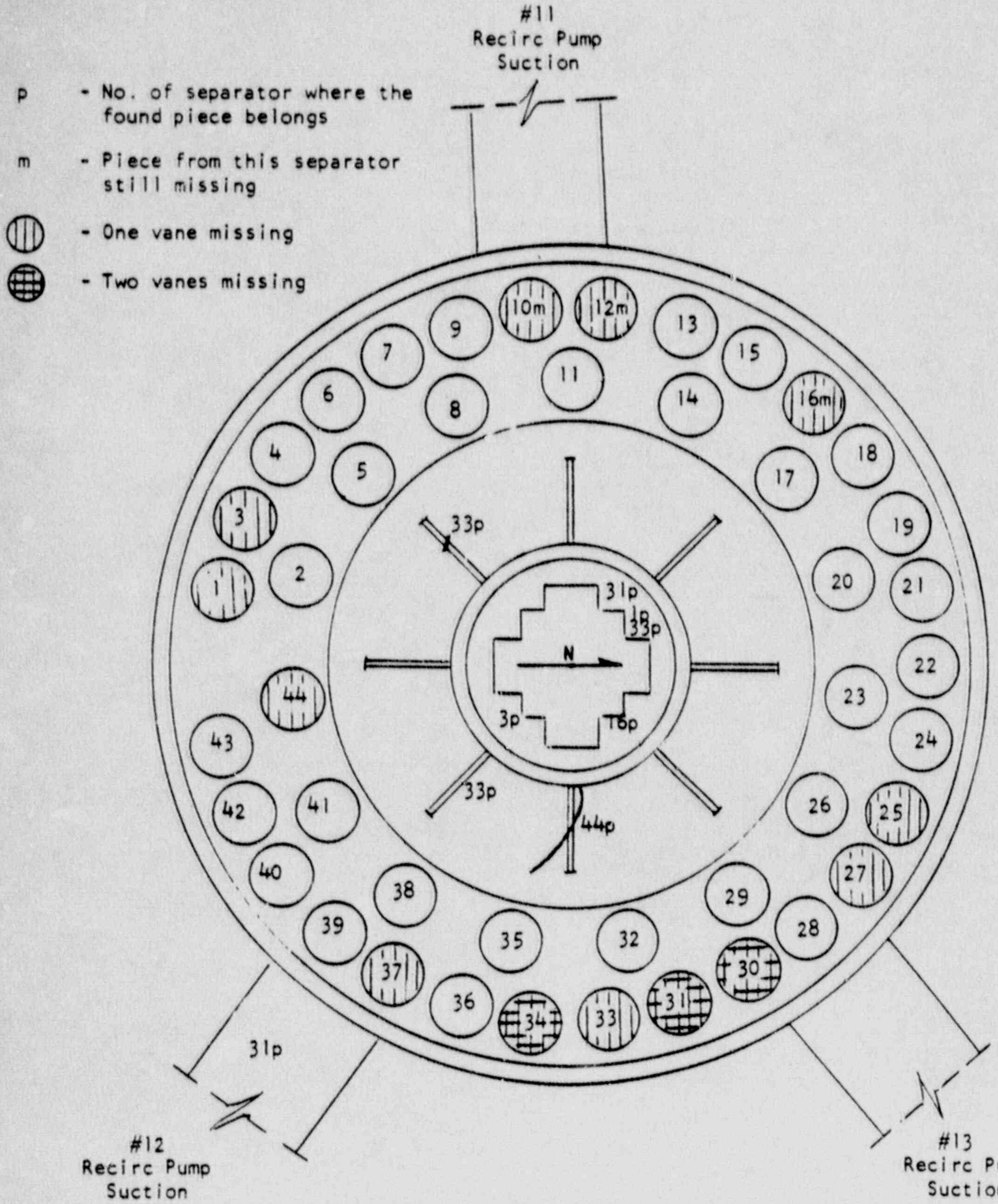


FIGURE I

Side View of Reactor Core

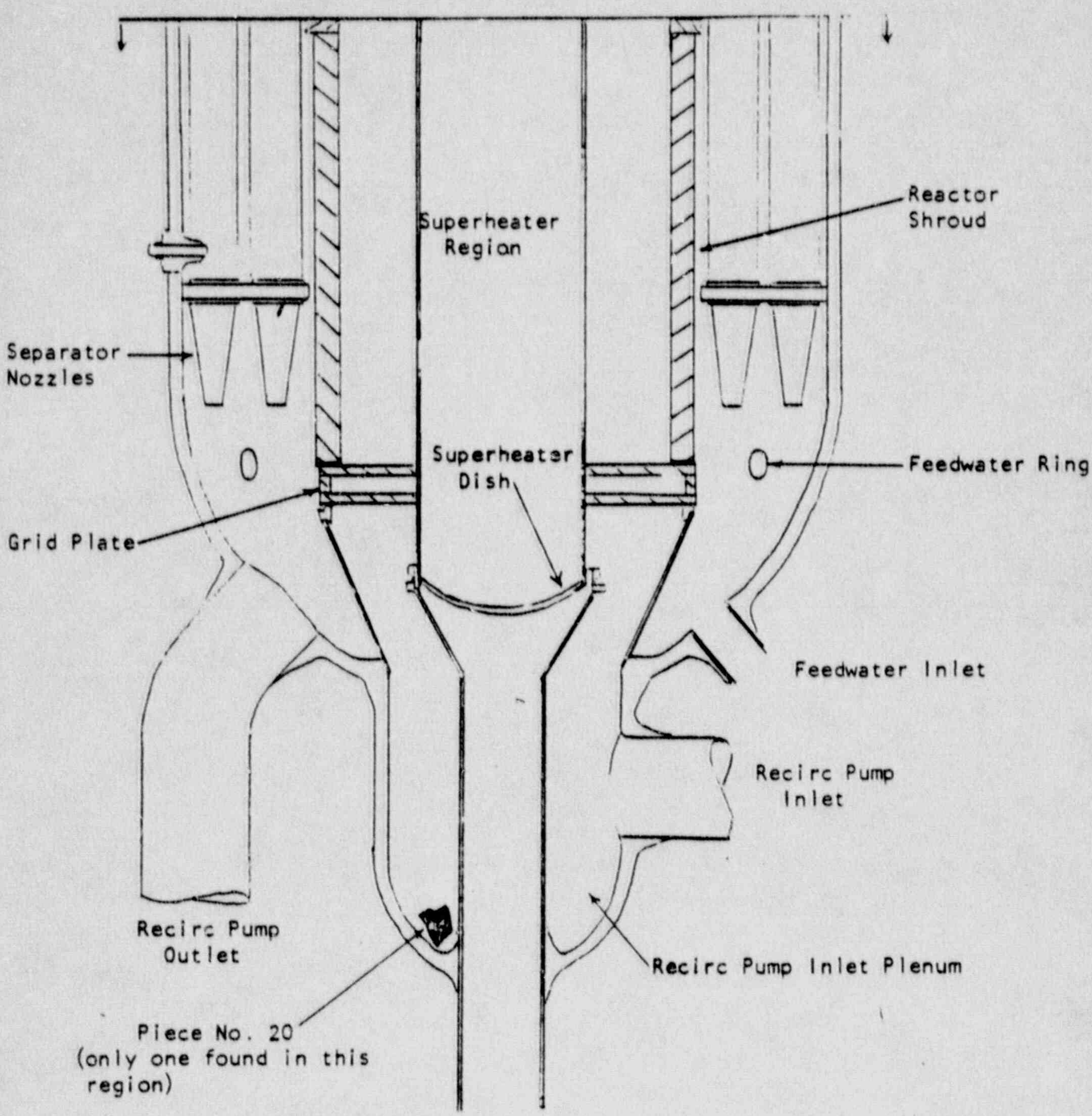


FIGURE 2

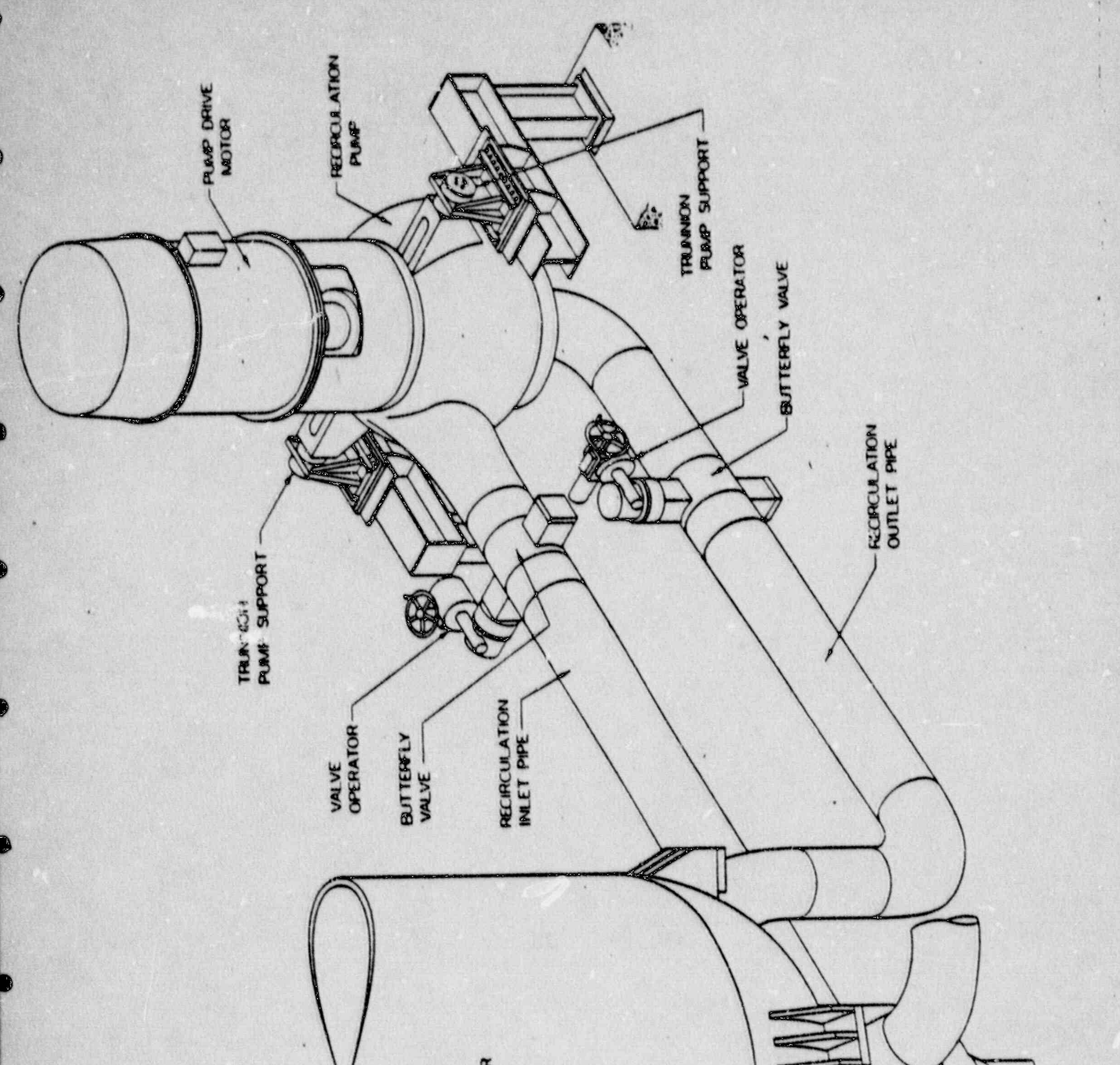
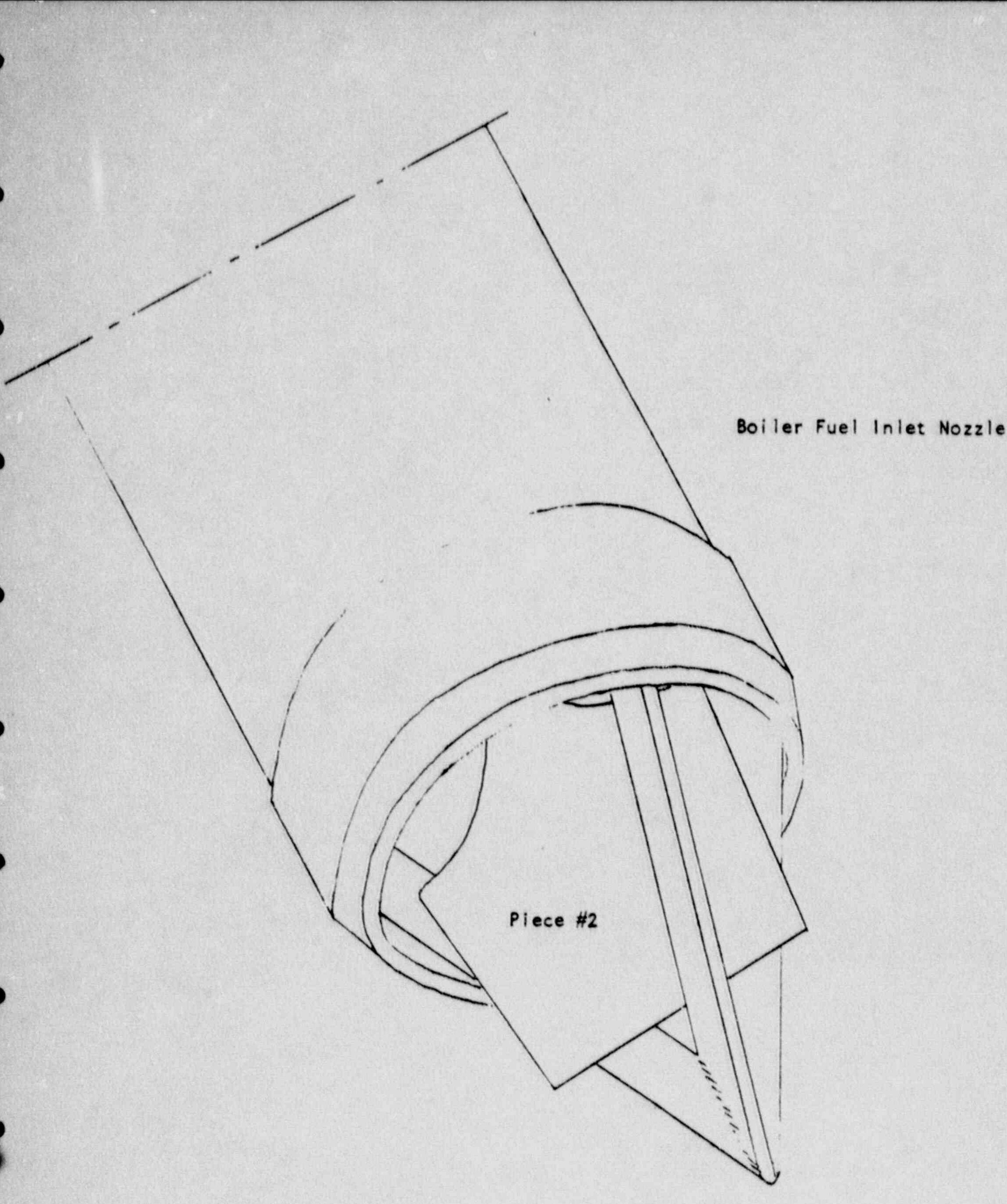


FIGURE 3 **RECIRCULATION PUMP LOOP NO. 2**



Location of Piece Found in Fuel Element

FIGURE 4

Steam Separator Nozzle

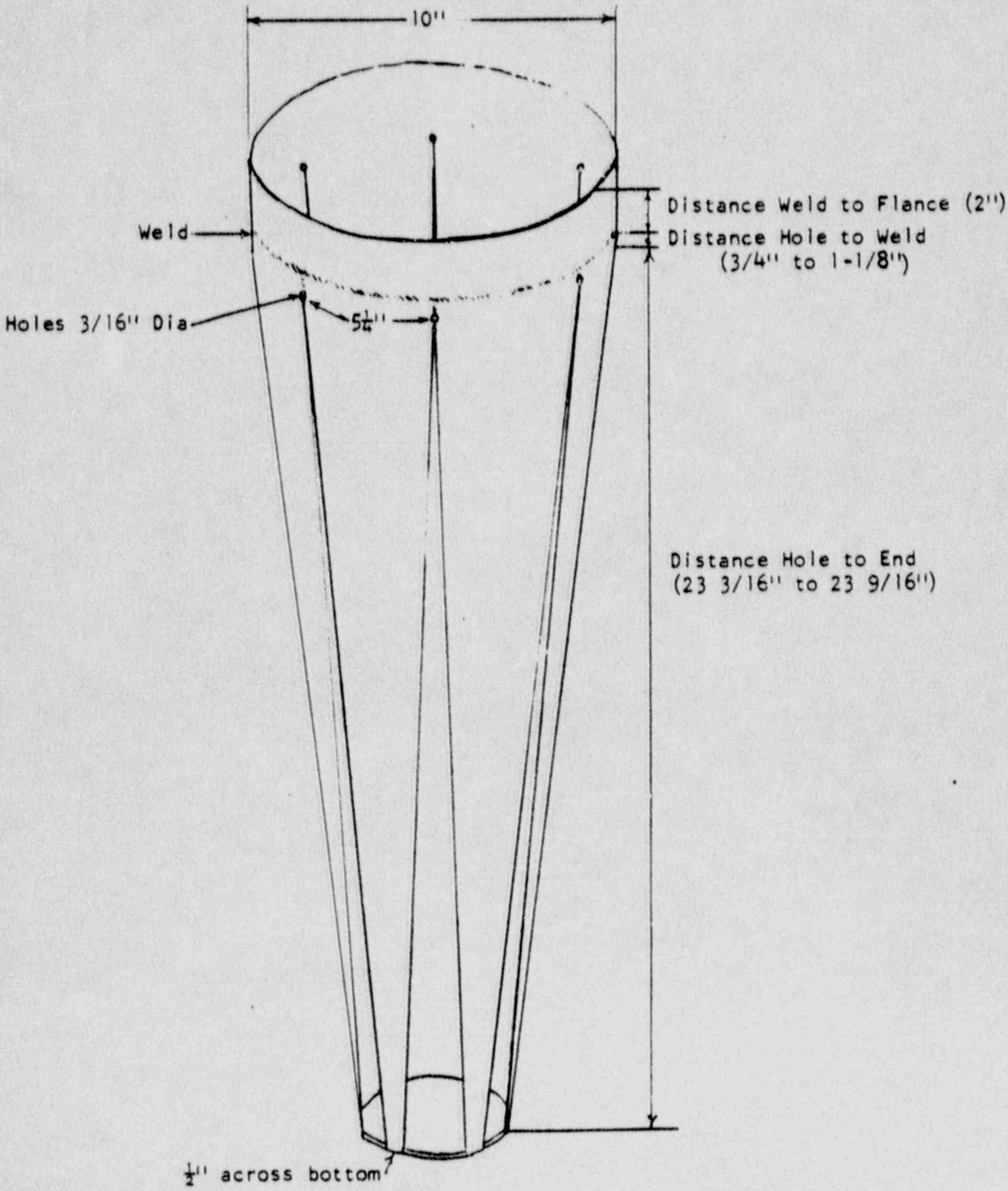


FIGURE 5

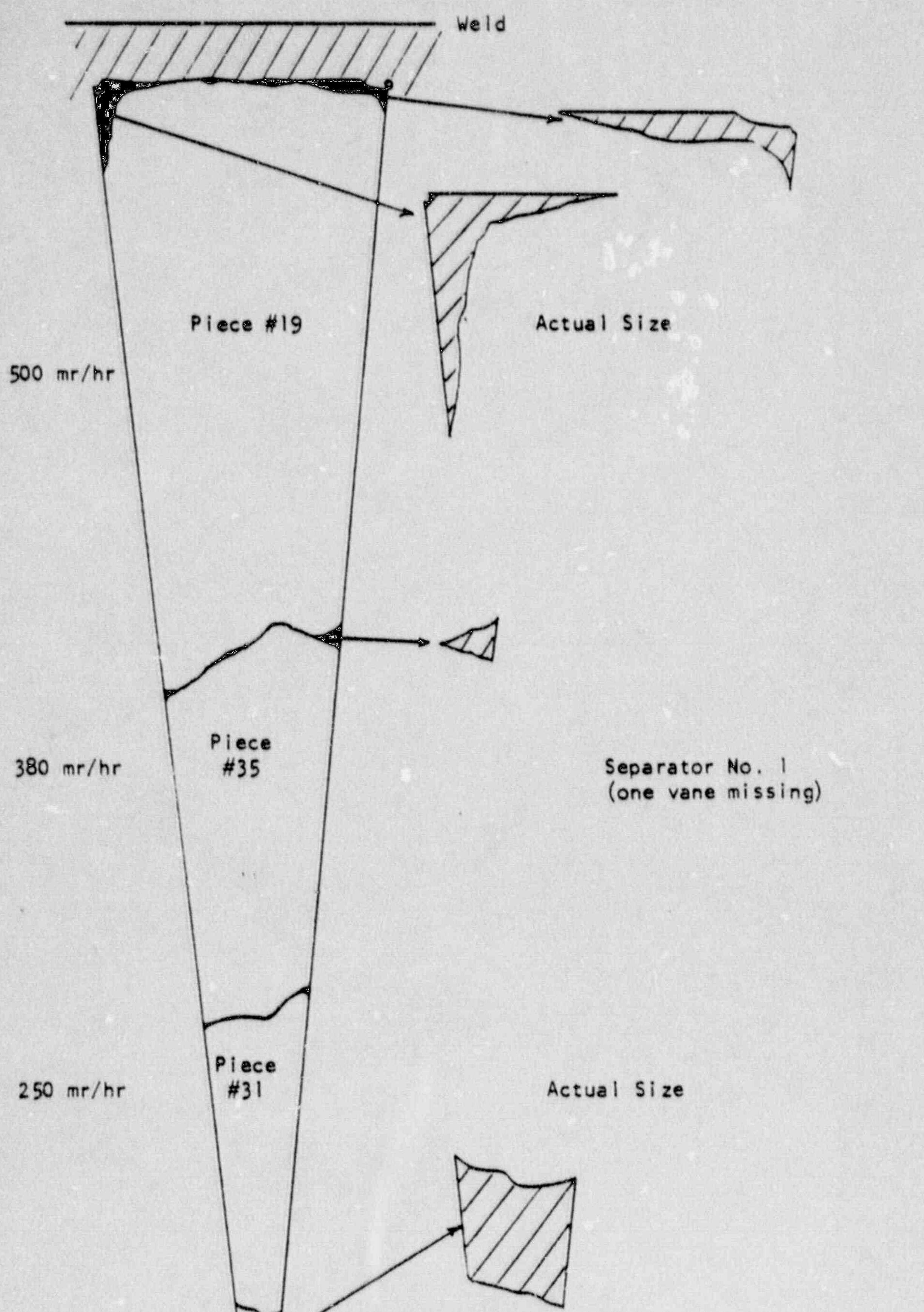

Weld
Piece #19
Actual Size
500 mr/hr
Piece #35
380 mr/hr
Separator No. 1
(one vane missing)
Piece #31
250 mr/hr
Actual Size
Scale: 1 = 3/8

FIGURE 6

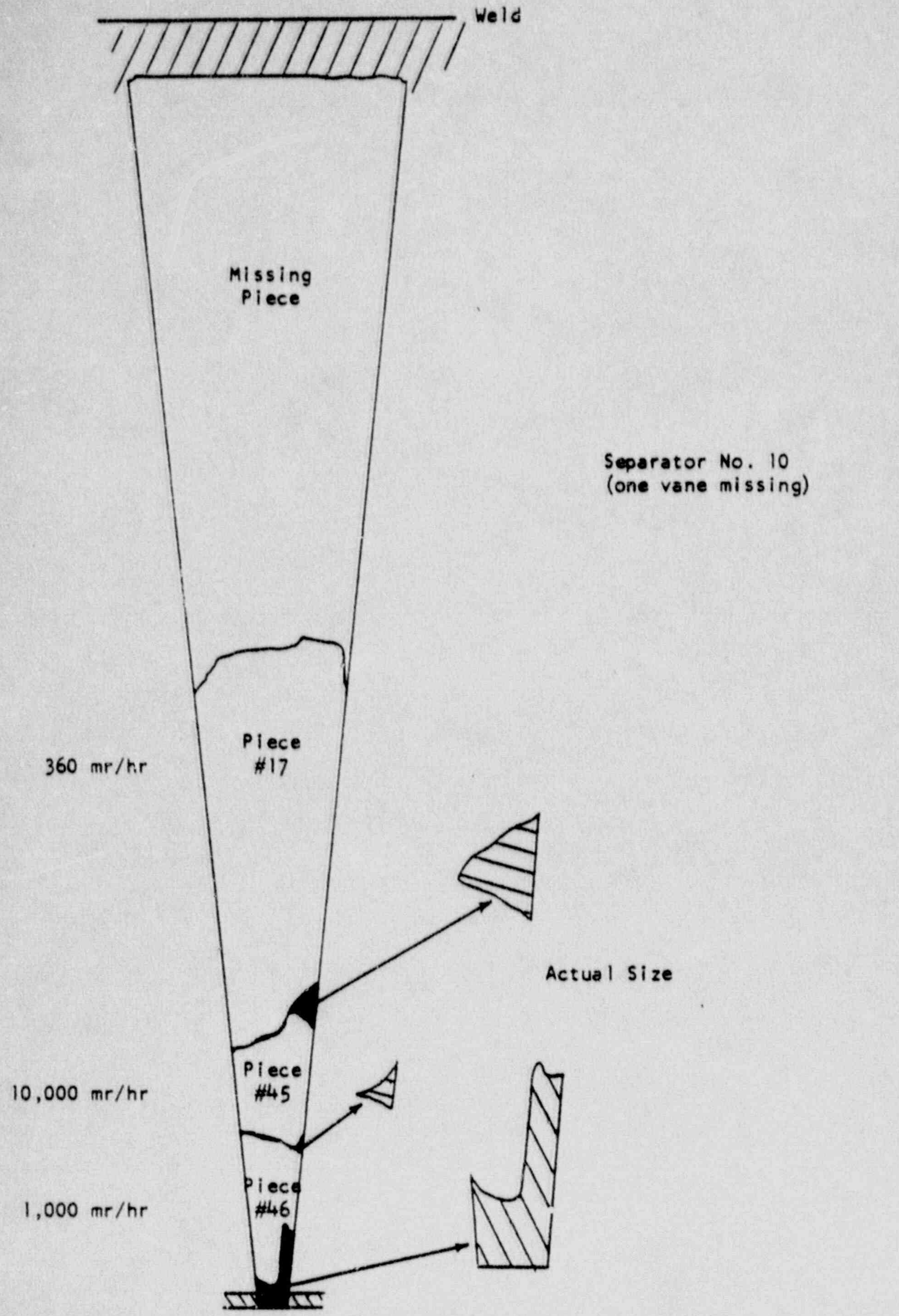

Weld
Missing Piece
Separator No. 10
(one vane missing)
Piece #17
360 mr/hr
Actual Size
Piece #45
10,000 mr/hr
Piece #46
1,000 mr/hr
Scale: 1 = 3/8

FIGURE 8

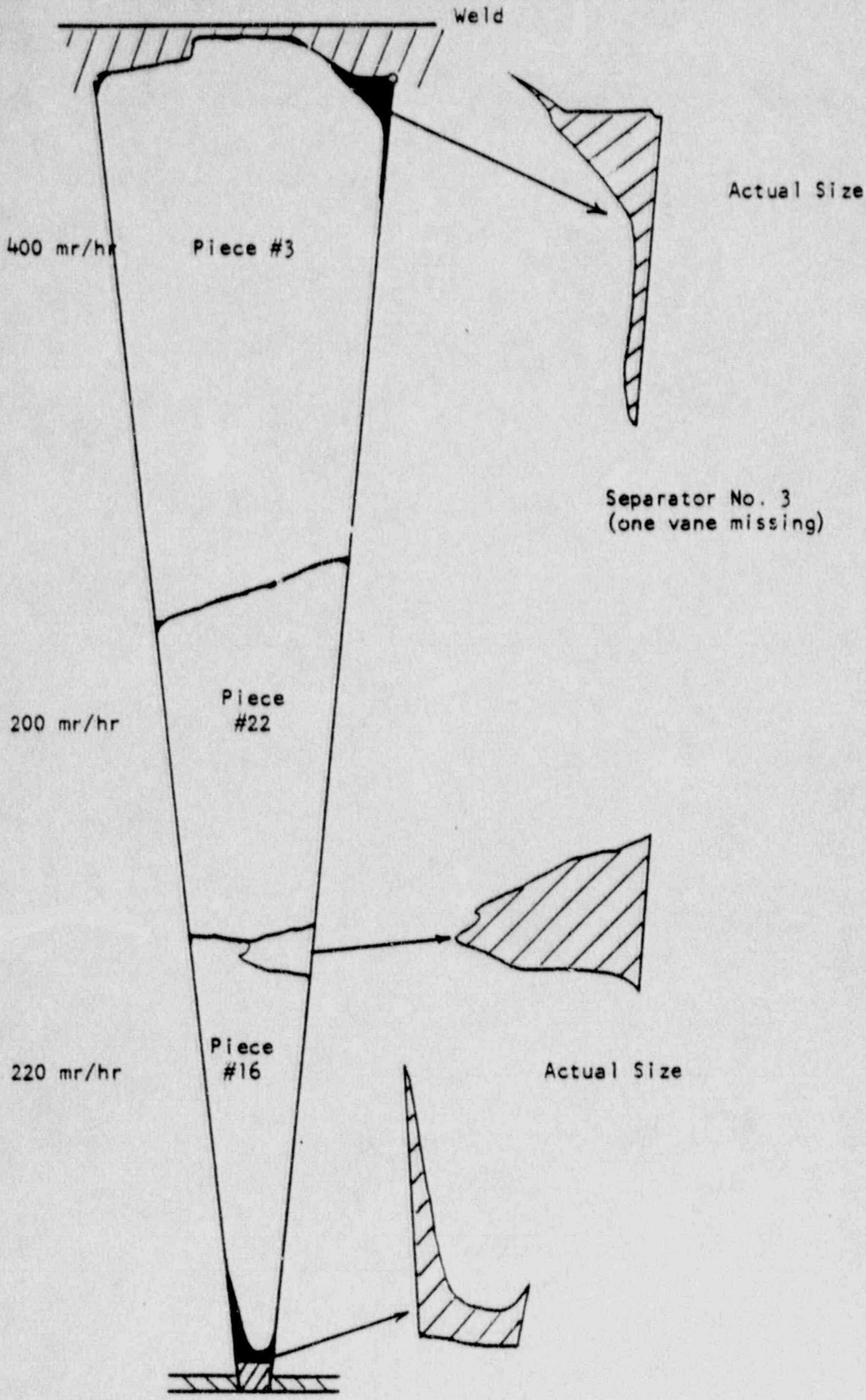


FIGURE 7

Scale: 1 = 3/8

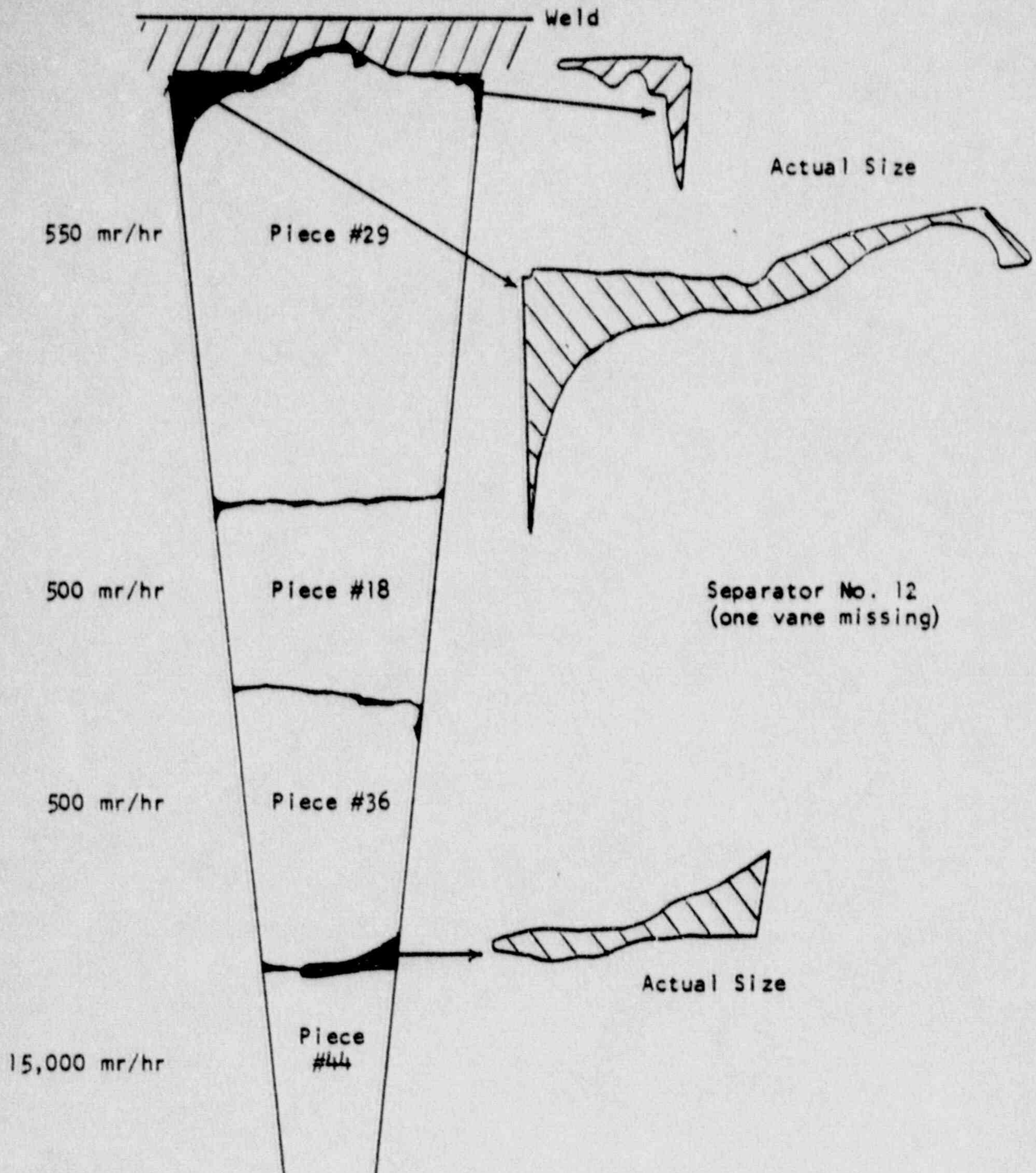


FIGURE 9

Scale: 1 = 3/8

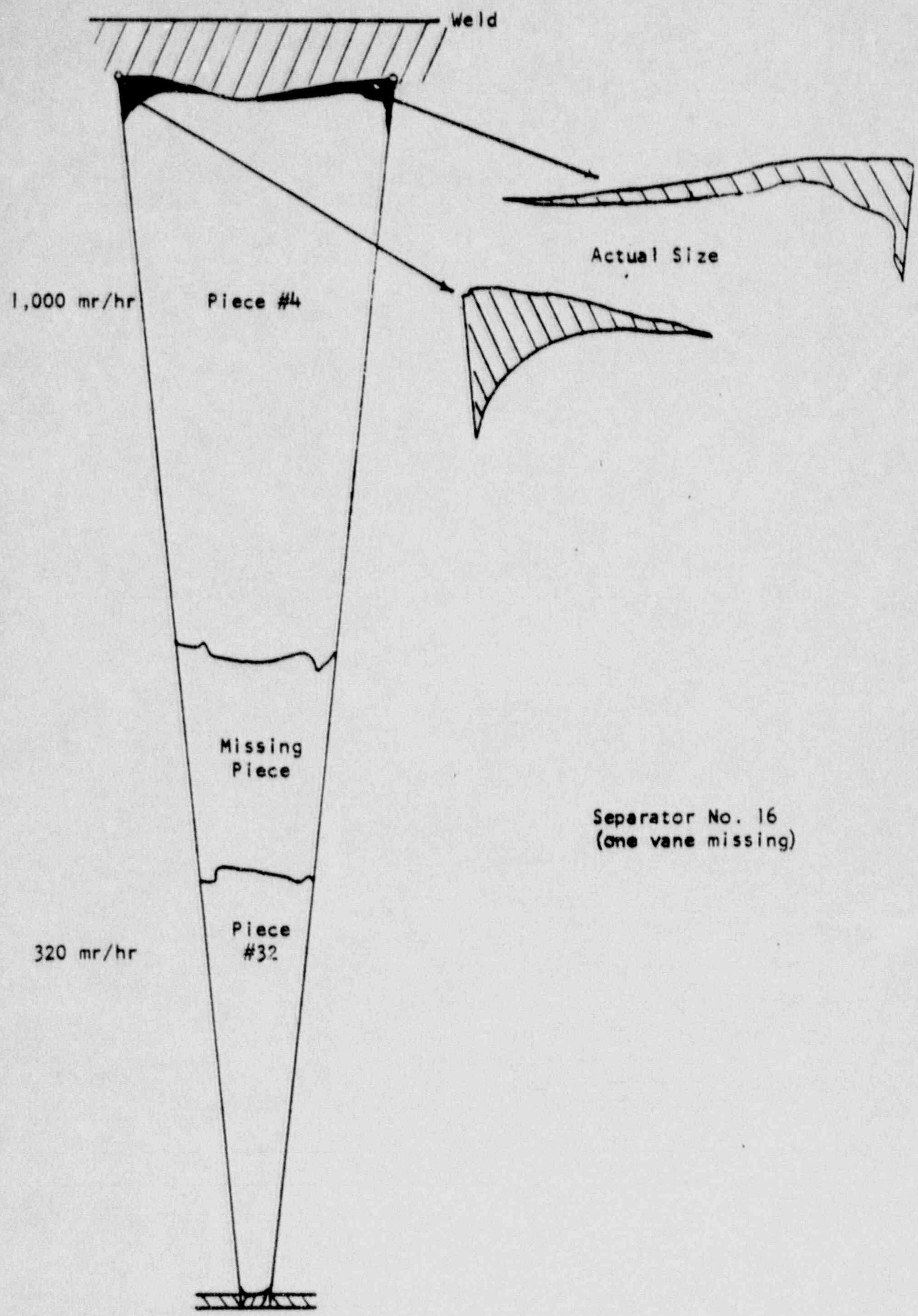


FIGURE 10

Scale: 1 = 3/8

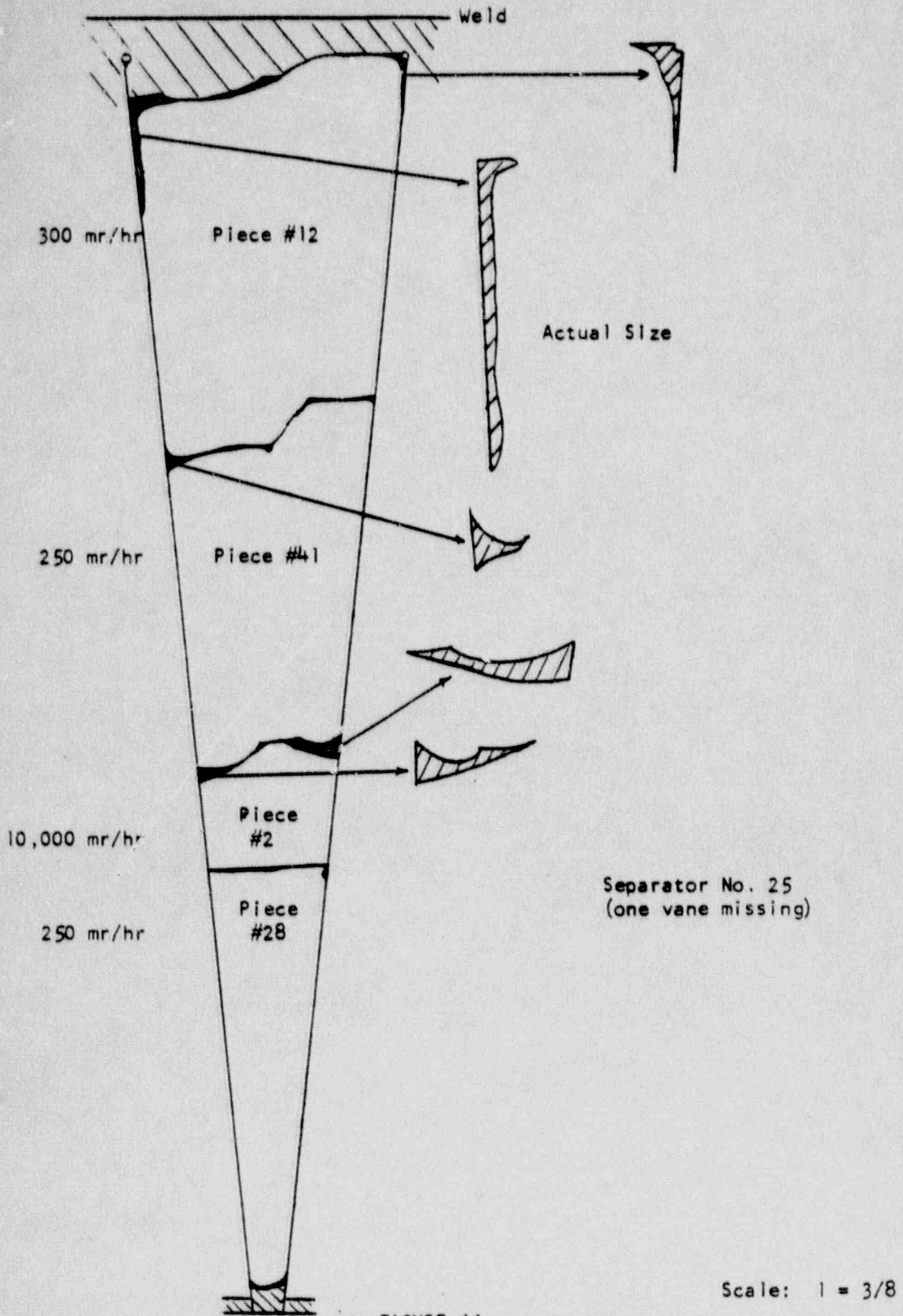

Weld
300 mr/hr
Piece #12
Actual Size
250 mr/hr
Piece #41
10,000 mr/hr
Piece
#2
250 mr/hr
Piece
#28
Separator No. 25
(one vane missing)
Scale: 1 = 3/8

FIGURE 11

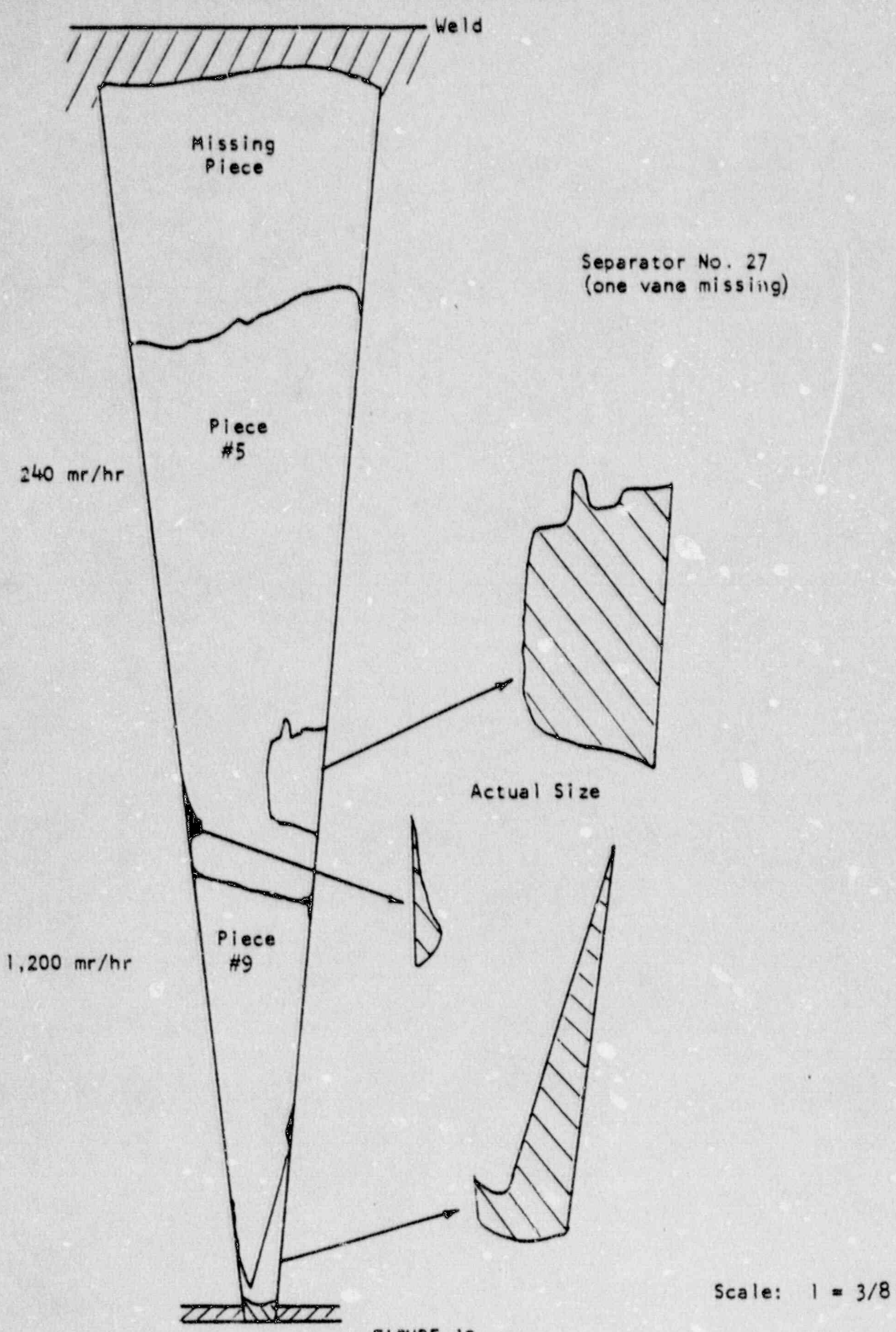

Weld
Missing
Piece
Separator No. 27
(one vane missing)
Piece
#5
240 mr/hr
Actual Size
Piece
#9
1,200 mr/hr
Scale: 1 = 3/8

FIGURE 12

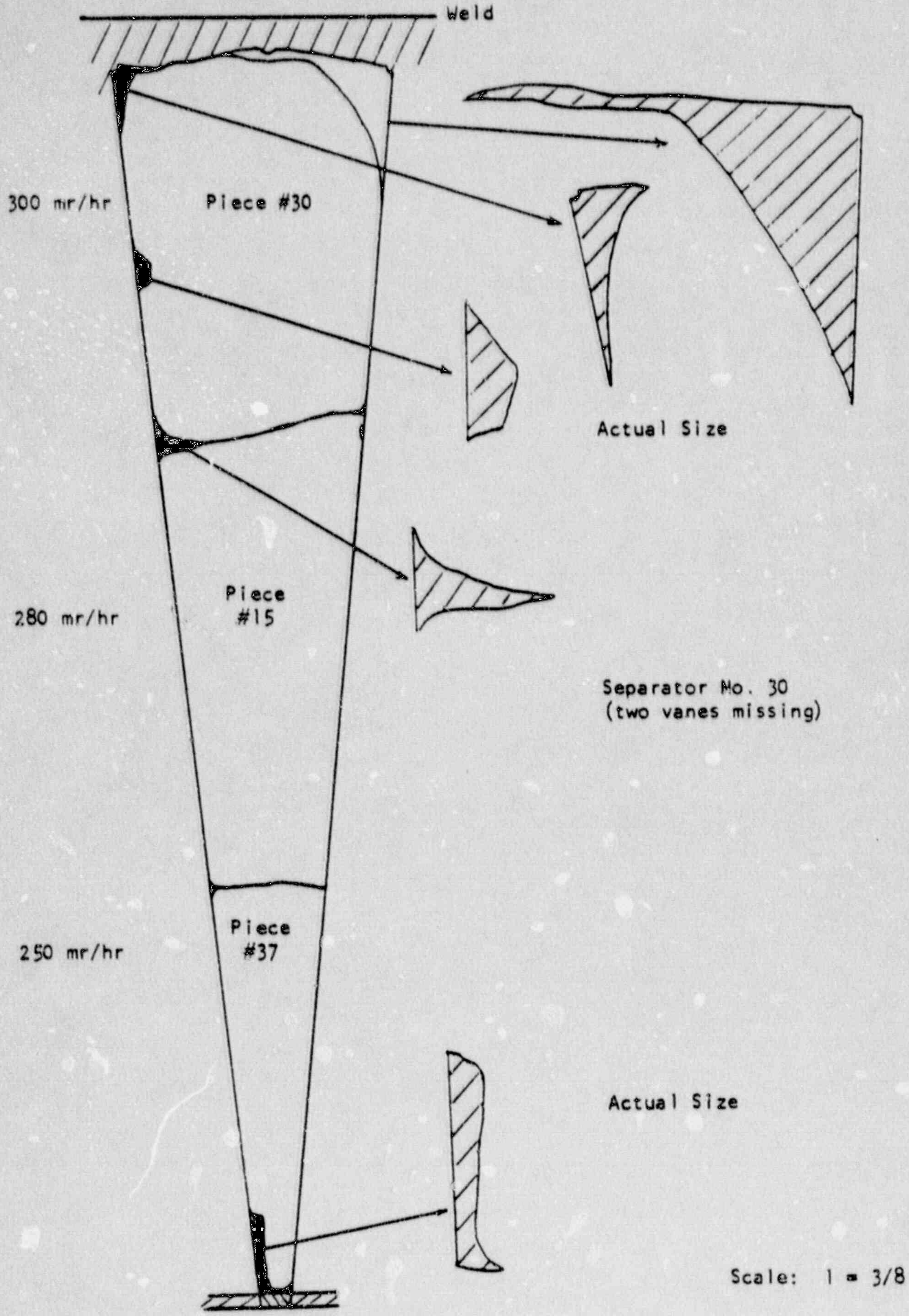

Weld
300 mr/hr
Piece #30
Actual Size
280 mr/hr
Piece
#15
Separator No. 30
(two vanes missing)
250 mr/hr
Piece
#37
Actual Size
Scale: 1 = 3/8

FIGURE 13

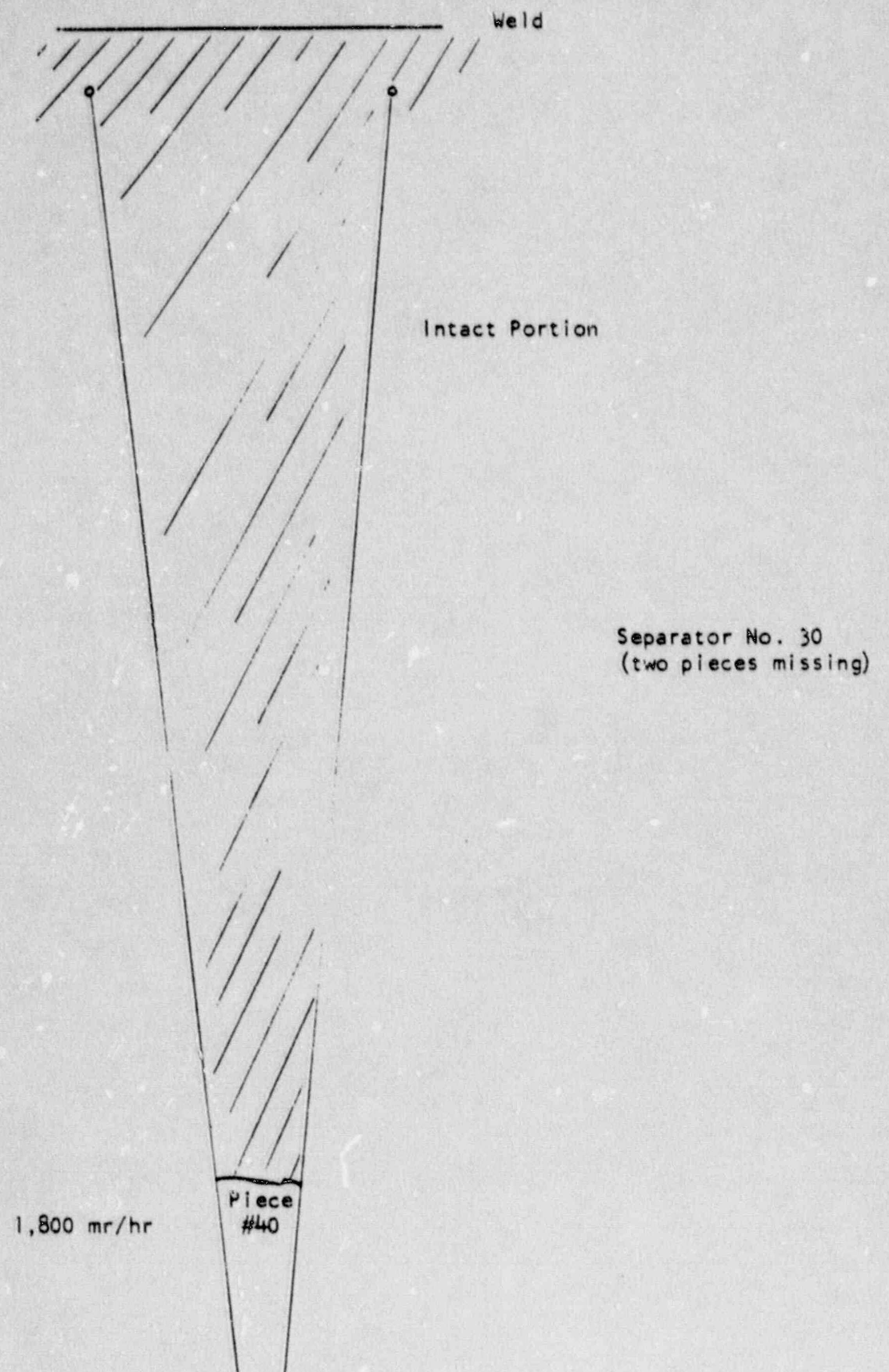


FIGURE 14

Scale: 1 = 3/8

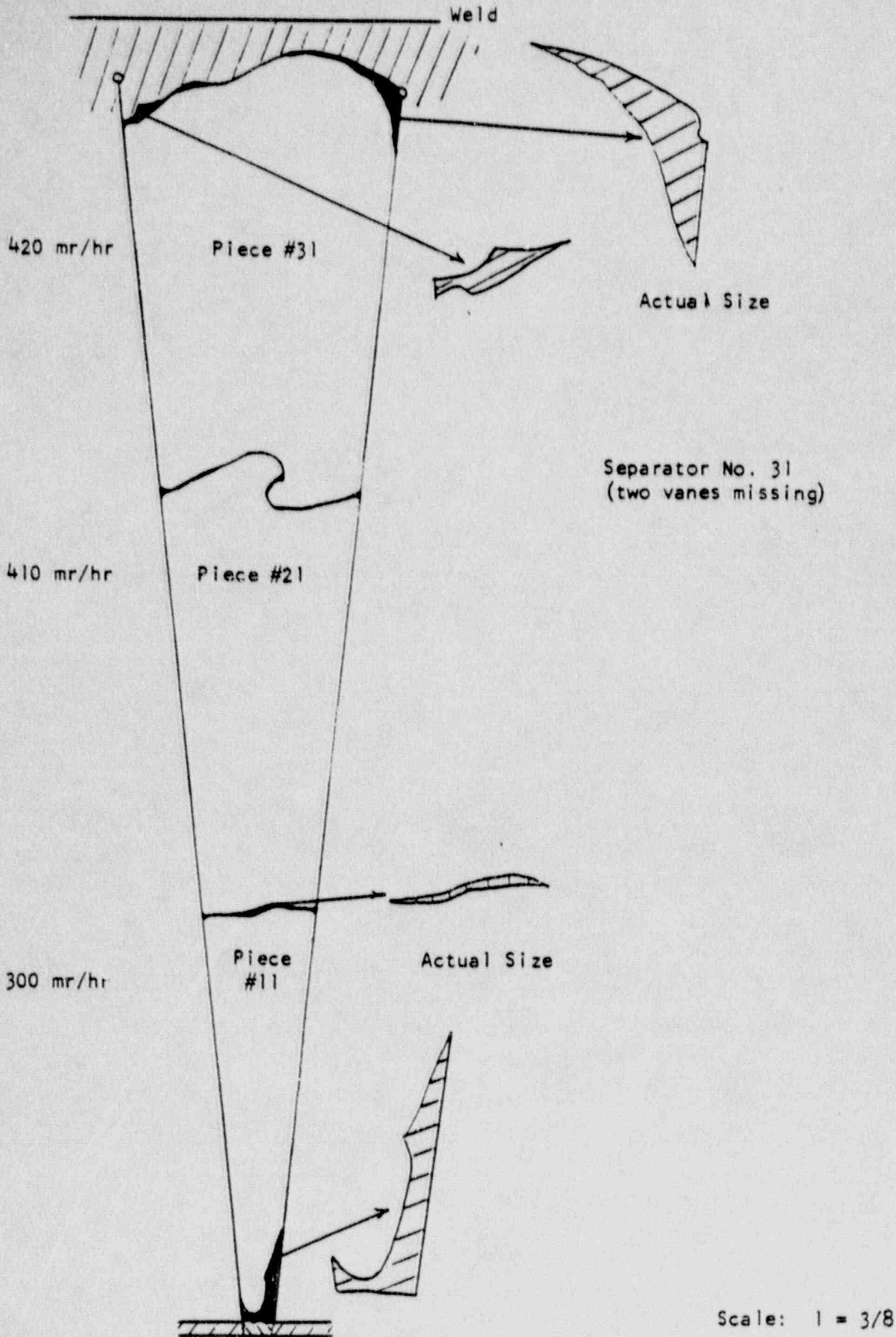

Weld
420 mr/hr
Piece #31
Actual Size
Separator No. 31
(two vanes missing)
410 mr/hr
Piece #21
Piece
#11
Actual Size
300 mr/hr
Scale: 1 = 3/8

FIGURE 15

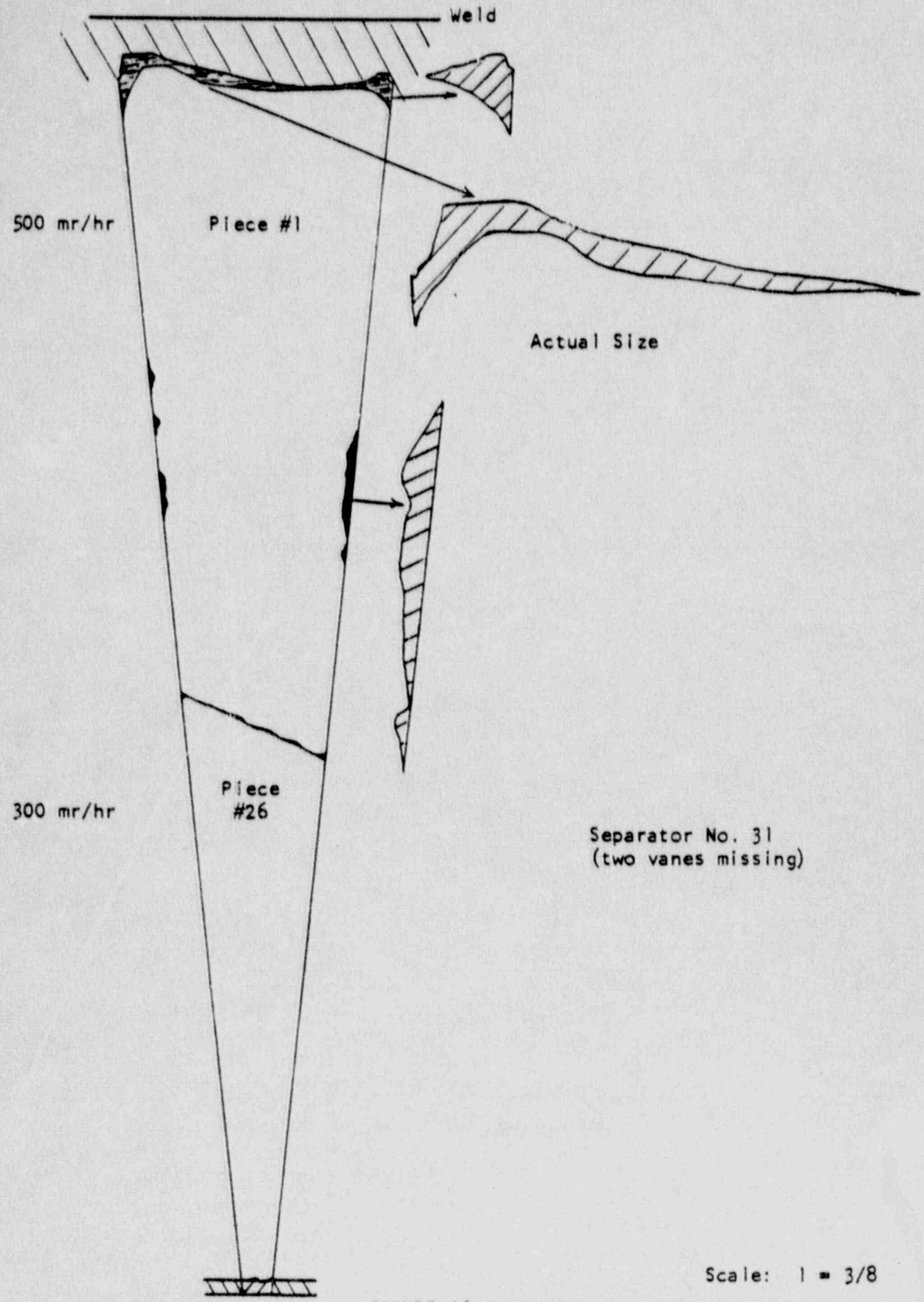

Weld
500 mr/hr
Piece #1
Actual Size
300 mr/hr
Piece
#26
Separator No. 31
(two vanes missing)
Scale: 1 = 3/8

FIGURE 16

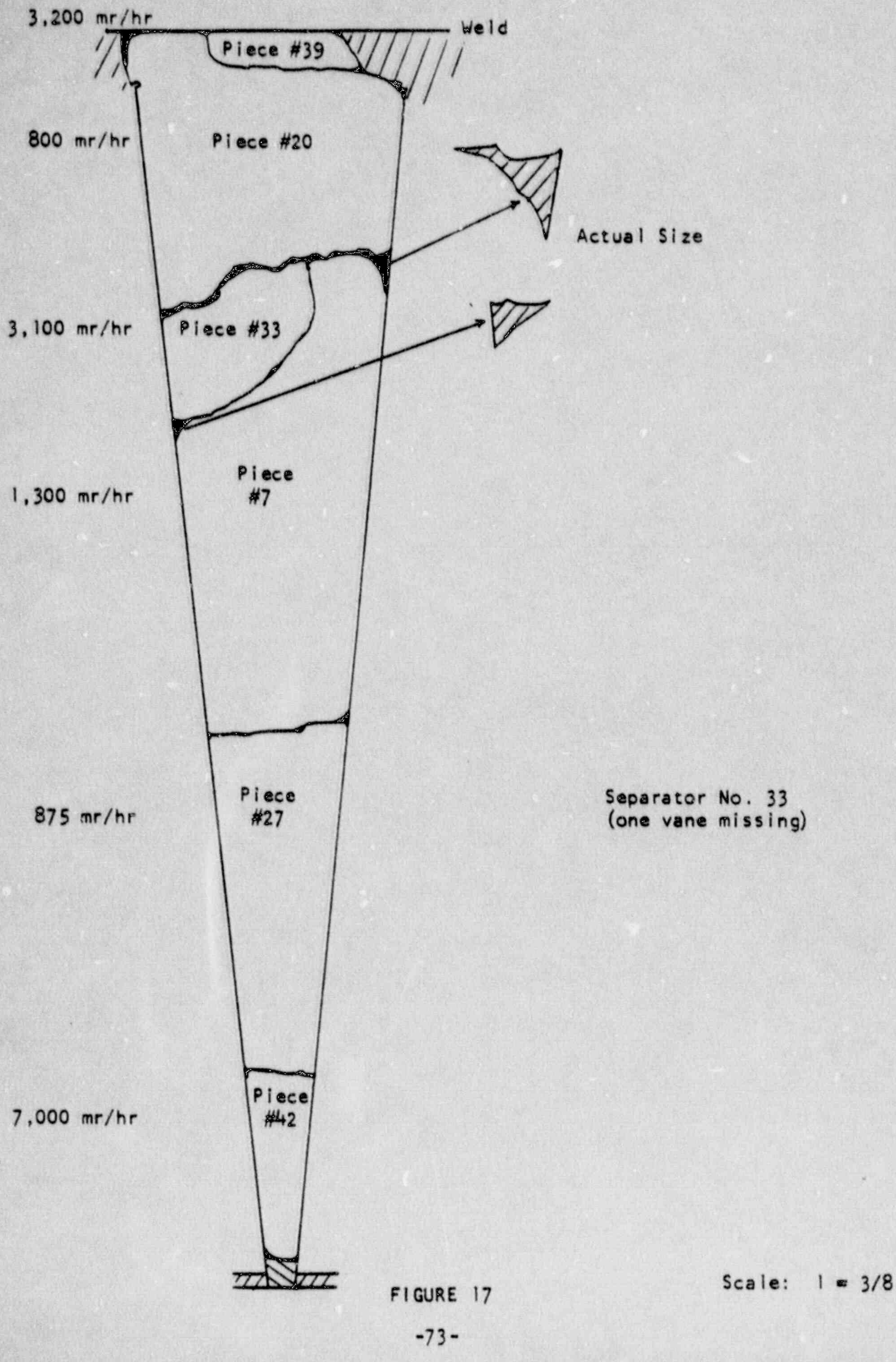

3,200 mr/hr
Weld
Piece #39
800 mr/hr
Piece #20
Actual Size
3,100 mr/hr
Piece #33
1,300 mr/hr
Piece
#7
875 mr/hr
Piece
#27
Separator No. 33
(one vane missing)
7,000 mr/hr
Piece
#42
Scale: 1 = 3/8

FIGURE 17

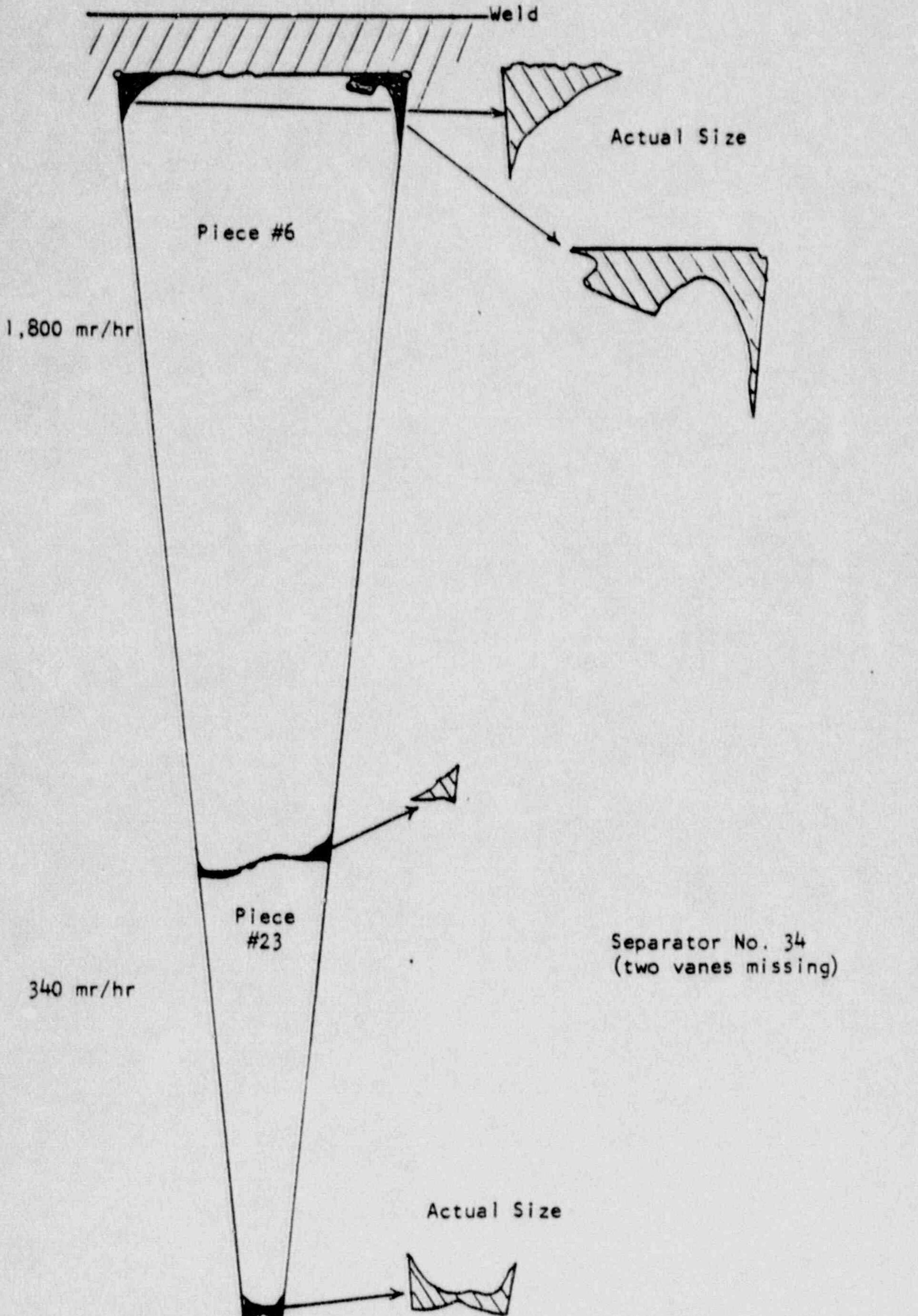

Weld
Actual Size
Piece #6
1,800 mr/hr
Piece
#23
Separator No. 34
(two vanes missing)
340 mr/hr
Actual Size
Scale: 1 = 3/8

FIGURE 18

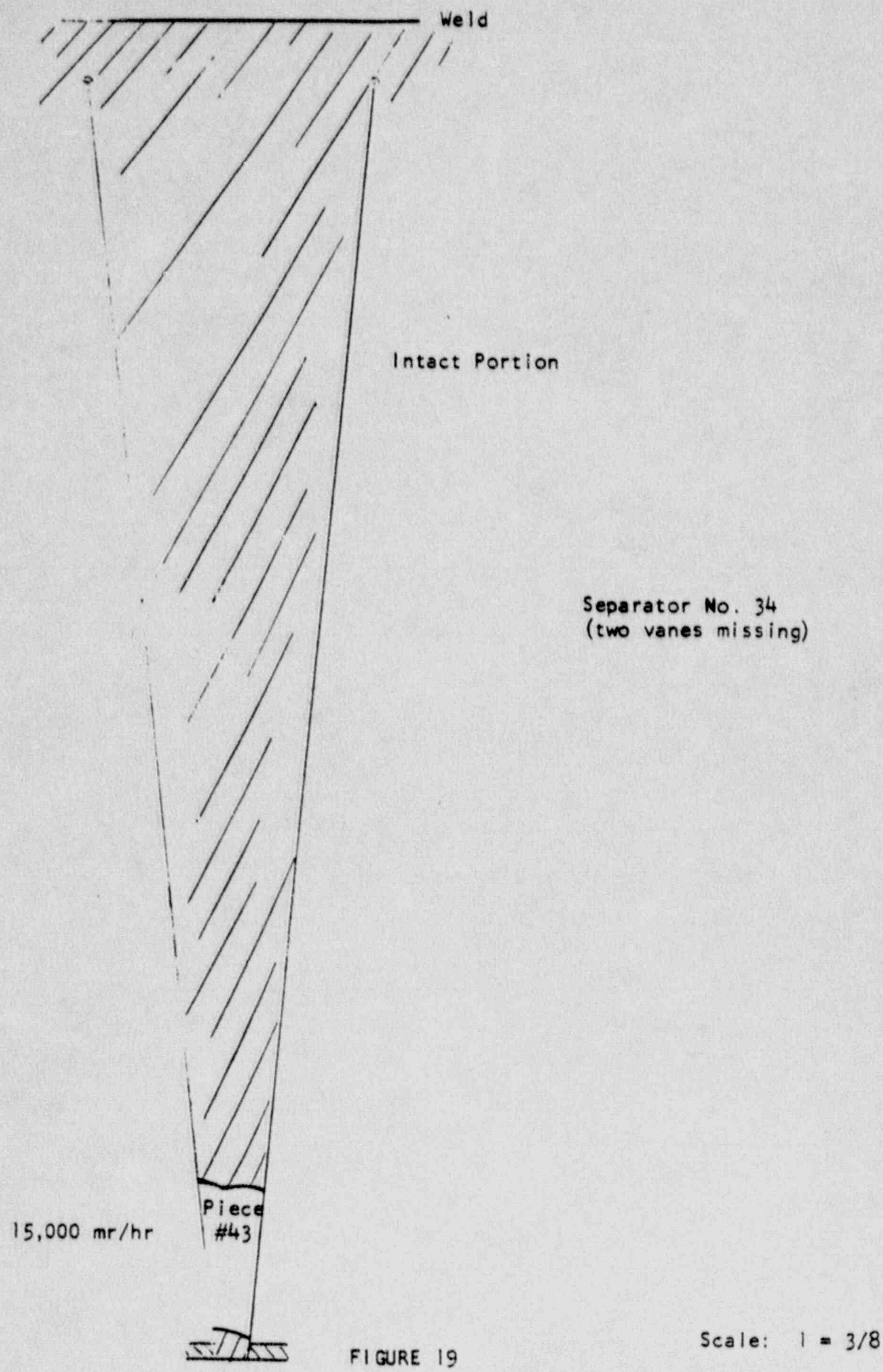

Weld
Intact Portion
Separator No. 34
(two vanes missing)
Piece
#43
15,000 mr/hr
Scale: 1 = 3/8

FIGURE 19

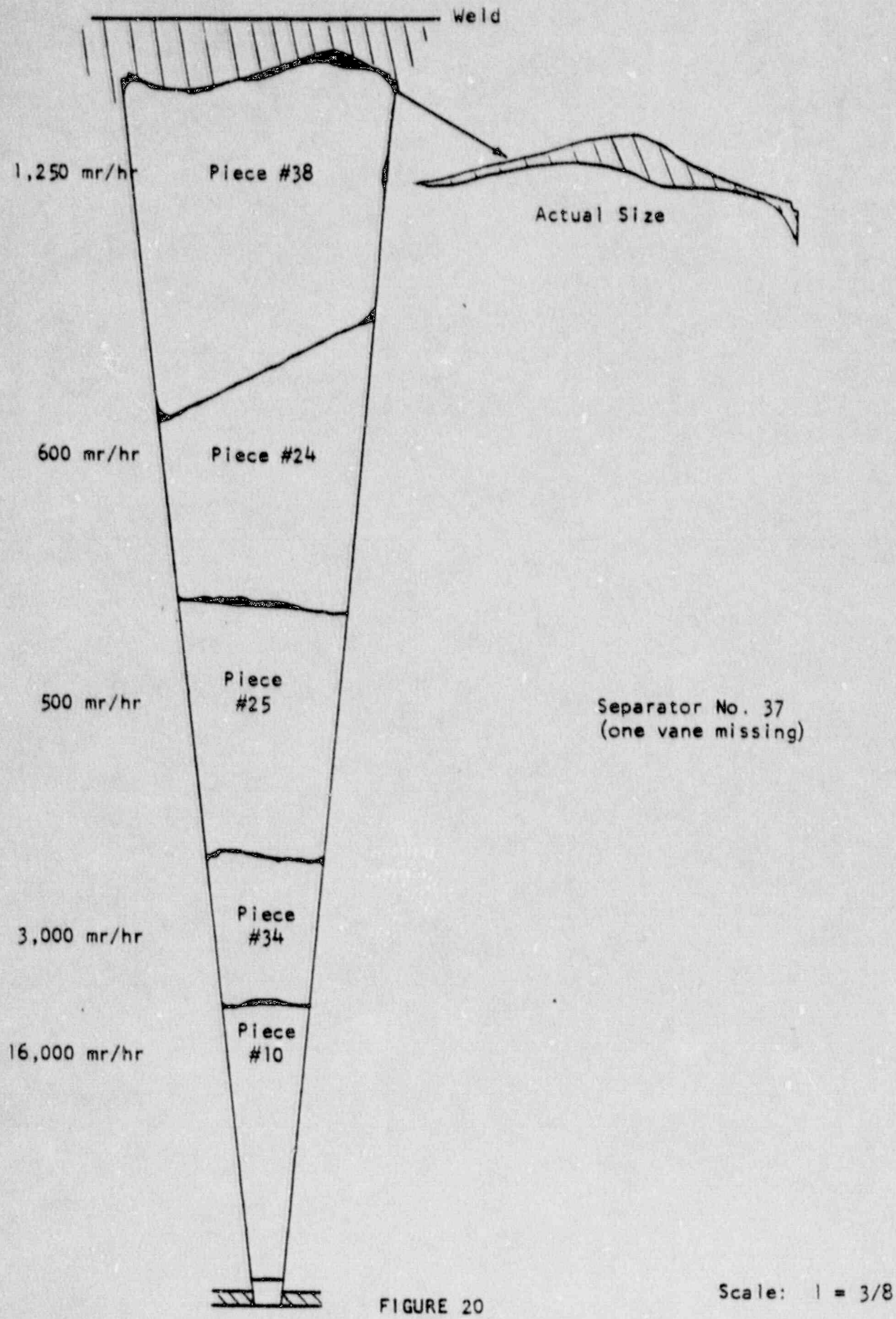

Weld
1,250 mr/hr
Piece #38
Actual Size
600 mr/hr
Piece #24
500 mr/hr
Piece #25
Separator No. 37
(one vane missing)
3,000 mr/hr
Piece #34
16,000 mr/hr
Piece #10
Scale: 1 = 3/8

FIGURE 20

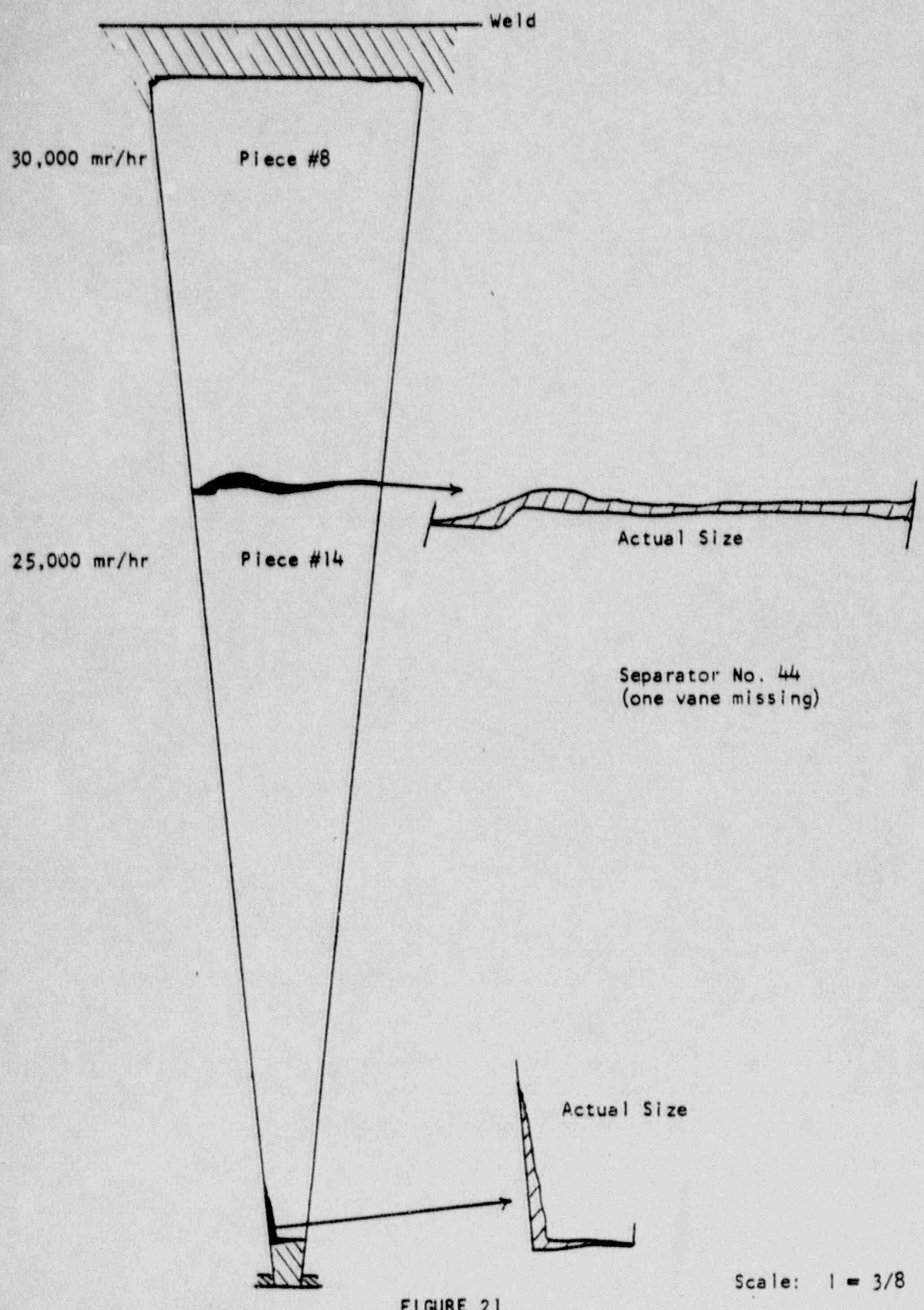

Weld
30,000 mr/hr
Piece #8
Actual Size
25,000 mr/hr
Piece #14
Separator No. 44
(one vane missing)
Actual Size
Scale: 1 = 3/8

FIGURE 21

L. Neutron Windows

The inspection of the steam separators revealed that the neutron windows were rubbing on the steam separators. This rubbing has caused wear marks on the separators adjacent to the windows. This wear is probably caused by low amplitude relative movement between the separators and the neutron windows. Also one window has apparently shifted position during operation.

During the inspection of the steam separators, marks were noticed on the sides of certain steam separators. These marks were found to be caused by the neutron windows rubbing on the separators. Each separator was inspected for the location, number, and severity of these marks. The neutron windows were weighed and inspected for wear on the braces that are used for points of contact with the separators.

The following is a detailed description of the neutron window inspection and the associated marks on the separators. See Figure I, page 80, for a plan view of the windows location and Figure II, page 81, for a side view of the window.

Neutron Window No. 1

This window has worn areas in both positions No. 1 and No. 2 (see Figure III, page 82) on separators numbers two and forty-four. The most severe wear marks were caused by the weld beads at the top and bottom caps. The weld beads have worn into the separators by themselves as far as possible. The entire length of the neutron window is now rubbing on the separator body. This has stopped the concentrated wear by the weld beads and has resulted in a greatly reduced rate of penetration. The depth of weld bead wear is approximately 1/16 inch. Approximate calculations show that the time required for the same amount of penetration into the .109 inch wall of the separator by the entire window is 34 times the present "wear" time. Indications are that the probability of wearing a hole in the separator body is slight.

As indicated by Figure III separators No. 2 and No. 44 each have 3 distinct marks. Two of the marks are caused by the window being in one position and the third mark by the window being in the other extreme position. Three separate marks appearing on the opposite separator verified that the window had moved. This window is the only one that has shifted and has the deepest wear marks. This may be explained by its being in front of the two-separator group area at the water column. Local flow conditions could have caused greater turbulence than was experienced by the other windows. The top half of the windows are above the bottom of the inlet to the separators (see Figure II) and are subjected to forces from the flow of water into the separators. The window was weighed for a check on its water tightness. The window weighed close to the expected 60 lbs.

Neutron Windows 2, 3, and 4

These windows all have wear marks on the adjoining separators but do not show any evidence of shifting. The windows all have two points of contact on one separator and one point of contact on the other. The deepest penetration on the separators is again caused by the weld beads but the marks are not as deep as those caused by No. 1 window.

These windows were also weighed to check for water tightness and found to weigh around the expected 60 lbs. The windows also show wear on the braces and sides of the windows.

Safety Considerations

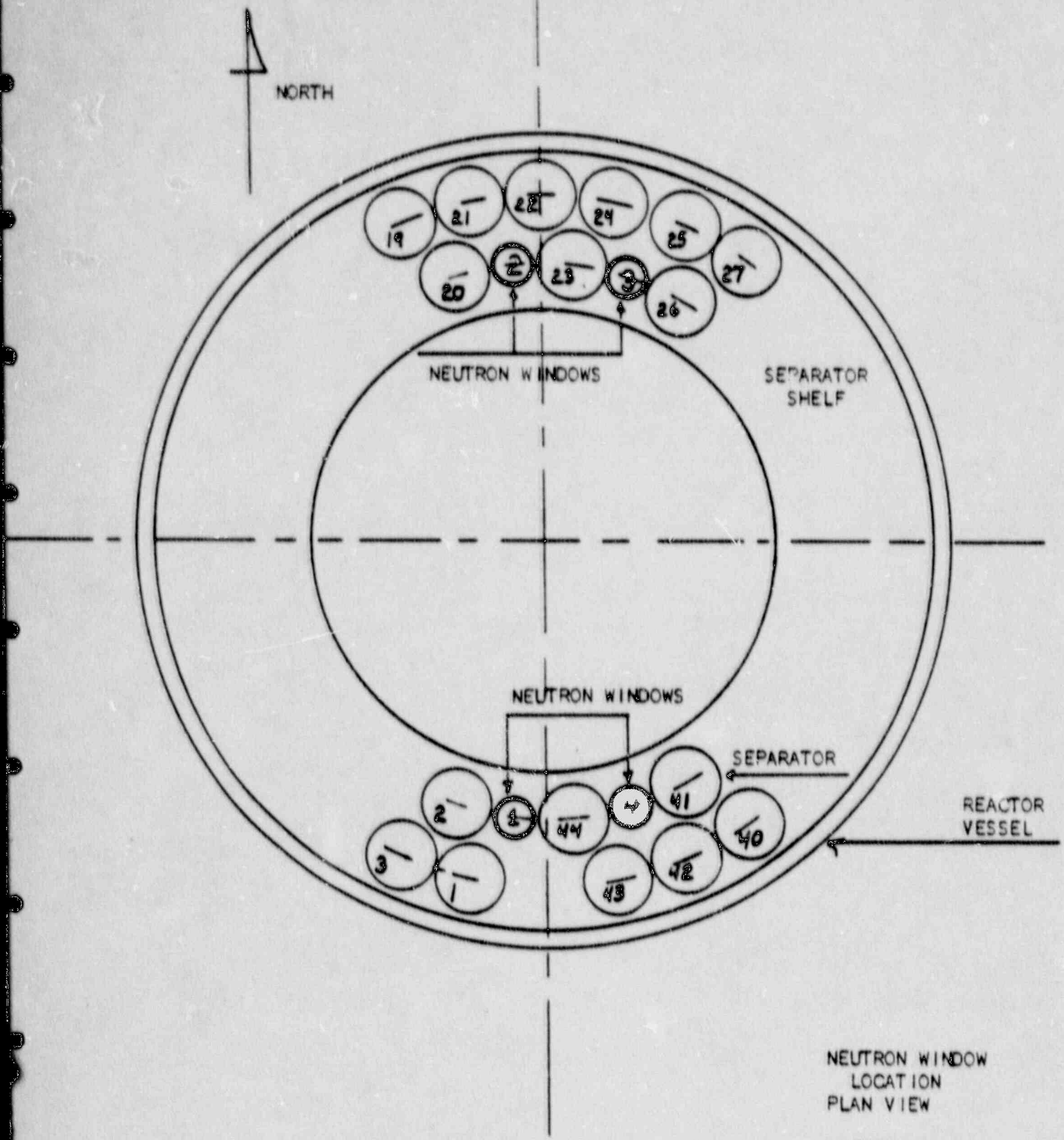
Flooding of the neutron windows would result in loss of the effectiveness of the voids used to increase the count-rate for startup. Since this could not conceivably happen to more than one window at a time the change in counts on the affected channel would be noticed and investigated.

In a letter from J W Lewellen to D A Lampe the question of reactivity change due to window movement and flooding is discussed as follows:

> "In general, core configuration changes result in reactivity changes; hence, the possibility of window movement from its reference position raises the question of reactivity insertion by this movement. It is held that the magnitude of reactivity thus inserted is undetectable. Three factors account for this: (1) the window is confined to the reflector region, where neutron flux and importance are low, (2) the boiler core shroud enhances this effect by restricting the window so that it cannot approach the core more closely than about 5 inches, and (3) there is always a significant amount of water "backing up" the window, so that the reflector properties are insensitive to the particular window location. The same considerations apply to window flooding."

Window movement wearing holes in the steam separator is not considered to be a safety problem because the holes would be small and would not effect the structural strength of the separators. Also water flowing into the separator through these holes would have a very small effect on the vortex formed by the inlet nozzles.

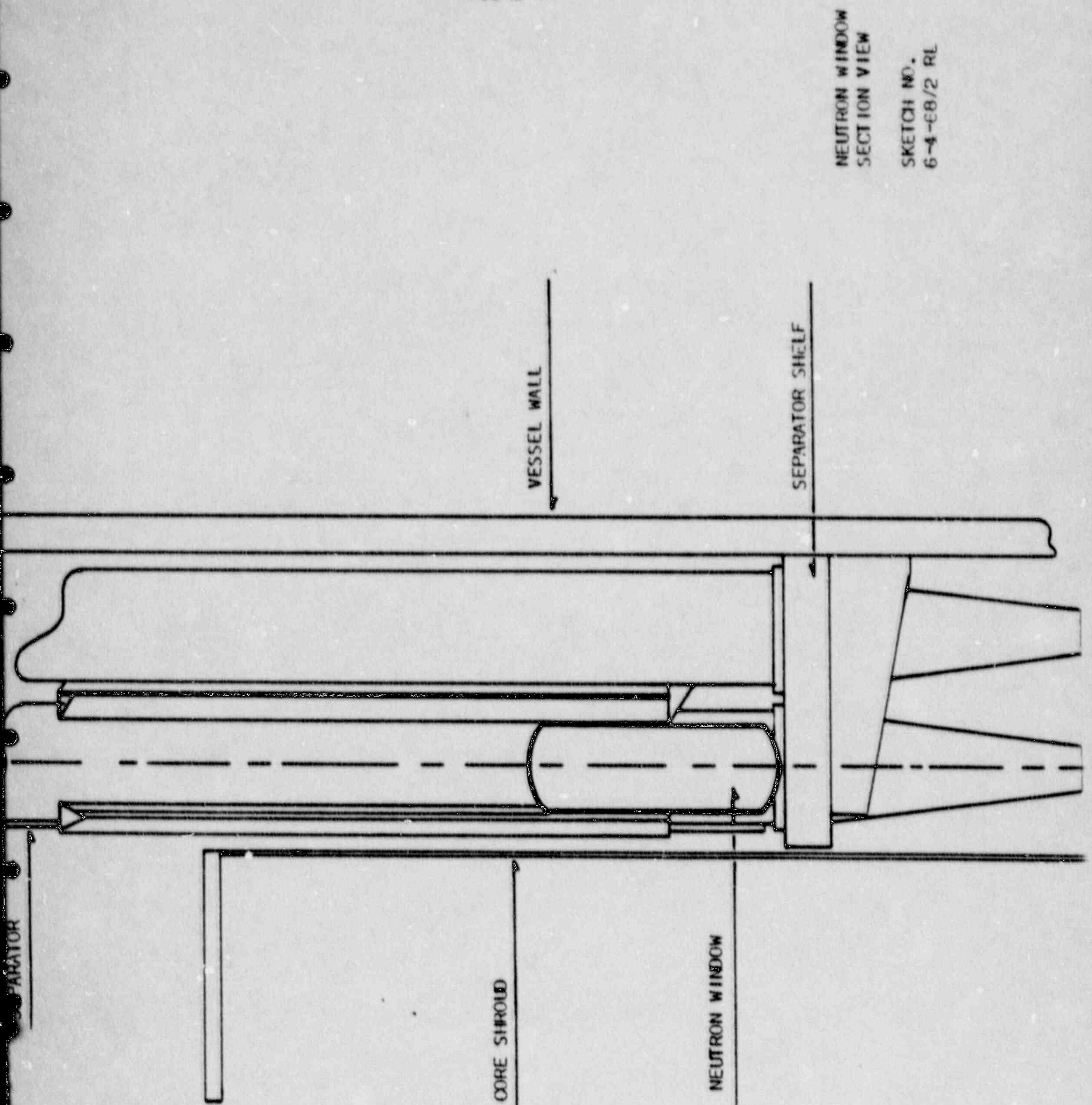
FIGURE 1



NEUTRON WINDOW
LOCATION
PLAN VIEW

SKETCH NO.
6-4-68-RL

FIGURE 11


NEUTRON WINDOW
SECTION VIEW
SKETCH NO.
6-4-68/2 RL
VESSEL WALL
SEPARATOR SHELF
CORE SHROUD
NEUTRON WINDOW

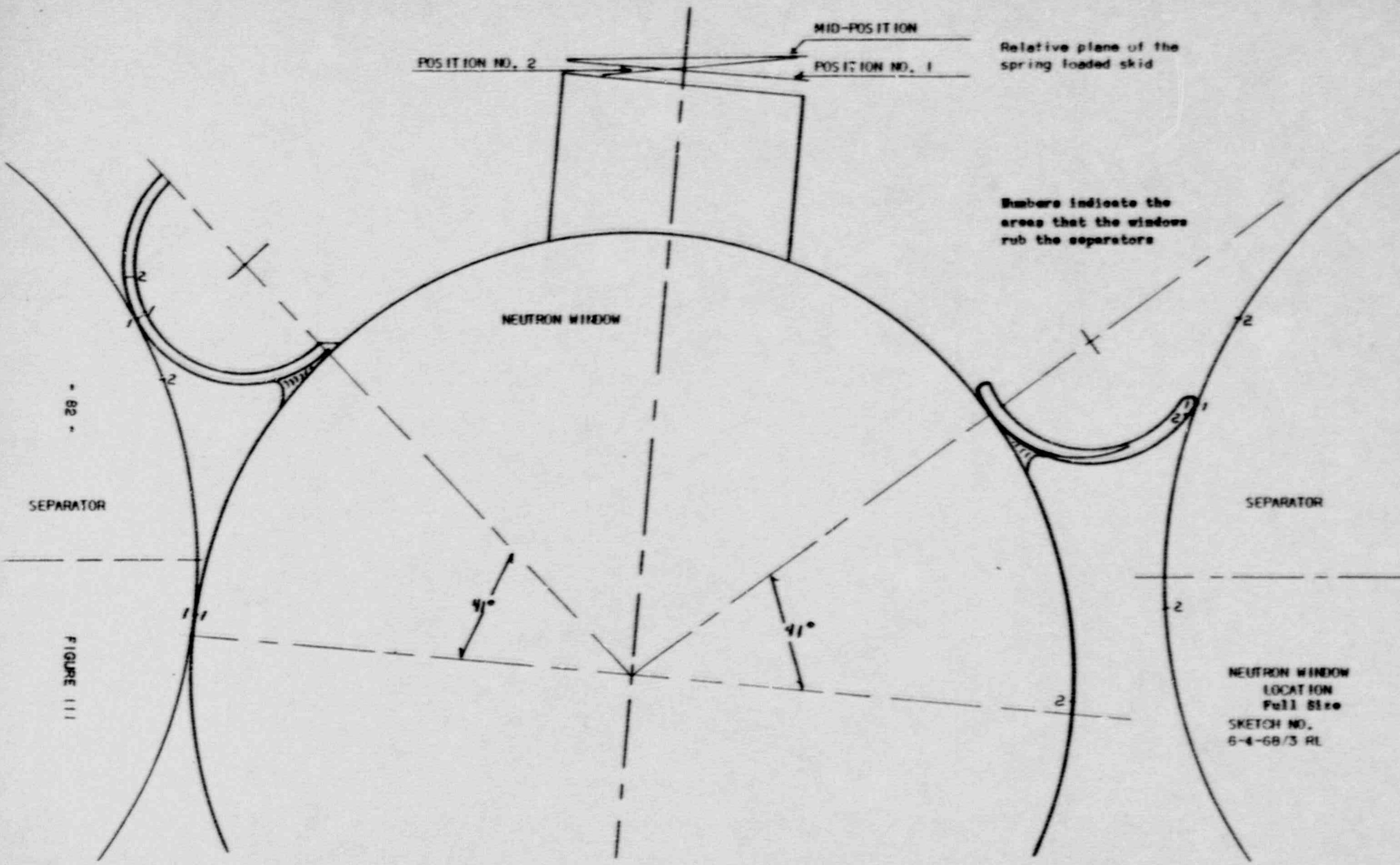

MID-POSITION
POSITION NO. 2
POSITION NO. 1
Relative plane of the spring loaded skid
Numbers indicate the areas that the windows rub the separators
NEUTRON WINDOW
SEPARATOR
SEPARATOR
41°
41°
1
2
NEUTRON WINDOW LOCATION
Full Size
SKETCH NO.
6-4-68/3 RL

FIGURE III

M. Reactor Vessel

Inspection of the vessel wall was primarily to observe deposit patterns above the separator shelf and to look for separator vane pieces below the shelf. The water column penetrations and separator lugs were looked at for signs of gross failure.

Deposits on the wall appear to be a yellowish brown when viewed with the binoculars. Deposits removed with a white cloth swab were more a grayish brown.

A notable lack of deposit area was observed slightly above the normal operating water level. The upper edge of this area is a rather straight line around the vessel at the upper edge of the separator lugs. The lower edge is a wave pattern from the bottom of one separator lug to the next.

Below this lack of deposit area, buildup on the wall appears to be its heaviest and then tapers off deeper into the vessel. About 1/3 of the way down from the separator lugs to the shelf, a light ring was noted.

Above the lugs, the deposit is about as heavy as mentioned above. It appears quite even here up as far as can be seen. There are spots in the deposit which are about 1/2 by 2 inches. These appear somewhat larger than spots below the lugs. Another ring which looks like a lack of deposit area circumscribes the wall at the level of the upper water column. This is the approximate level of the steam outlet from the steam separators.

No signs of erosion were noted around the upper water column nozzle. Deposits on this insert were darker than the rest of the wall. Only the radiation shield around the lower water column nozzle could be observed. This was done with the periscope as opposed to the binoculars used for all other observations above the separator shelf. Close up views of some of the welds on the radiation shield suggested that everything was still intact and not distorted or out of alignment.

Welds on all separator lugs seemed very good. None of the lugs appeared to be distorted or out of alignment.

Both the recirculation water inlet and outlet cavities were vacuumed and inspected with the boroscope. No separator vane pieces were found. Two small socket head cap screws were retrieved from the southwest corner of the recirculation water outlet cavity. These are believed to be screws used in fastening an external electrical connector to the upper part of control rod drives.

N. Recirculation Pumps

No. 12 recirculation pump was pulled on October 18, 1967. At that time a piece of steam separator nozzle was found in the stationary discharge vanes of the pump casing. Scrape marks were observed on the impeller which were believed to have resulted from the piece passing through the pump. It is now known that many pieces must have passed through the pump.

Maintenance and inspection of No. 12 recirculation pump resulted in the following observations and actions: 1) The face and seat ring were worn and were replaced. 2) The mechanical seal was in good condition. The four floating seal rings were worn to double the design clearance and were replaced, 3) The shaft was lightly scored in the area of the lower bushing seal ring and was cleaned up before reassembly, 4) The lower bushing seal ring was replaced.

The wear was not beyond expectations considering that the seal had not received maintenance for 27 months and the pump had been subjected to severe vibration at times during the last couple months of operation. The machinery records show that this pump has 7800 hours of operation since the last overhaul which is equivalent to almost a year of continuous operation. A major portion of seal wear takes place during pump starting and this probably will be minimized in the future.

Recirculation pump No. 12 was replaced on April 30 before pulling pump No. 11 on May 2. Since there are still a few pieces of separator nozzles that have not been accounted for it was hoped that some of these pieces would be found when this pump was pulled. No pieces were found in the pump casing or in the pipes up to the butterfly valves, however slight scratches on the impeller indicate that pieces of separator nozzle may have passed through this pump also.

The seals were observed to be in very good condition. Two new parts were installed: The lower floating seal ring, and the lower bushing ring. Three parts were reconditioned: the seal seat ring, the face seal ring, and the floating seal ring container. The lower motor bearing was re-packed although there was no slop in the bearing.

At the end of this report period preparations were being made to replace No. 11 recirculation pump and pull No. 13 pump. Again it is hoped that the missing pieces will be found in the No. 13 casing or in the piping on the pump side of the butterfly valve.

Since the operating conditions and hours on No. 13 pump were about the same as the other two pumps (7800 hours), it is expected that the amount of wear will be the same. The maintenance on No. 13 pump should be completed within a short time and it is expected that the pump will be back in place early in June. All three recirculation pumps will than be in good operating condition.

O. Chemical Analysis of In-Vessel Deposits

Deposits from the bottom of the shield pool floor and from the bottom, inside of the reactor shroud were taken on December 28, 1967. Subsequent chemical analysis then was undertaken on the deposits for qualitative and quantitative content.

The deposits were not uniform in size. They ranged in size from small granular type particles to particles of roughly a millimeter in diameter. The particles were also obviously not homogeneous in physical make-up. All of the particles yielded easily to crushing to a powdered form. After crushing and mixing, the mixture was of a typical iron-ore coloration (brown-red). The sample was radioactive and had an activity of about 2.5 R/hr at contact.

The first step was to make qualitative tests for various suspected isotopes. This was done dissolving a portion of the powdered mixture in concentrated HCl. A residue of undissolved solids was left after several hours which was separated by filtration.

The solution of dissolved solids was qualitatively analyzed by wet chemistry methods. Also the gross (total) sample and the undissolved solids portion were analyzed spectrally. The results are as follows:

Nickel	=	negative
Manganese	=	trace
Cobalt	=	trace
Copper	=	trace
Zinc	=	trace
Zirconium	=	positive
Iron	=	positive

The spectral analysis of the undissolved solids yielded a similar spectrum as the gross sample but it had an extra peak. The peak had an energy of about 0.72 - .74 mev which is the approximate energy range of zirconium 95. The fact that zirconium is present seems reasonable since the element cladding at Pathfinder is zircalloy, a zirconium alloy. The positive indication of zirconium was borderline and probably in actuality, would only be a trace amount.

These undissolved solids were steel gray in color and were soluble only to an appreciable amount in HF. It follows that these solids were wholly or in part then probably composed of silicides. These are very insoluble compounds usually only soluble in HF and/or alkali.

$ZrSi_2$ is a compound soluble only in HF and alkali. It is steel gray in color. Therefore, it appears that the undissolved solids were silicides (in part or in whole) and were at least partially composed of $ZrSi_2$.

Since a very positive test was qualitatively obtained for iron, a quantitative determination of iron was performed by permanganate reduction and titration. It was a guess at best to determine in what form the iron existed but most probably it was in the form of an oxide of the formula Fe_2O_3. This assumption was borne in mind when the following data was calculated.

	Dissolved Sample	Total Sample
% Fe	55.90	48.37
% Fe_2O_3	79.75	69.17
% undissolved solids		13.46
% of unknown which contained traces of Mn, Co, Zn, & Cu		17.37

The 17.37% of unknown solids was not analyzed since it was only of primary interest to know that the majority of the deposits consisted of iron apparently in the form of Fe_2O_3.

Conclusions

It was very surprising not to detect a greater copper concentration, since Pathfinder does have admiralty tubing in the condenser and feedwater heaters. This would, however, seem to indicate very little corrosion of the admiralty tubing. It is, however, quite possible that a greater concentration of copper would be found in another part on the primary system.

The undissolved solids apparently contained $ZrSi_2$ with traces of $MnSi_2$ which has similar characteristics as $ZrSi_2$. Visual inspection and solubility tests were the only criteria used in establishing proof of silicides. Therefore, the proof is indicative but not totally conclusive.

The source of the iron could have resulted from corrosion of the structural materials of the condenser and condensate and feedwater piping. These parts are of carbon steel structure. Corrosion of the stainless steel parts of the reactor is probable but highly unlikely. The total corrosion effect has had several years over which to buildup.

P. Radiation Levels

Radiation measurements were made in air using calibrated gamma survey meters (CP-TP and Model 2586 Nuclear-Chicago) and underwater using the Landsverk Roentgen Meter Model L-64 and associated thimble ionization chambers. The gamma survey meters are rate meters while the thimble ionization chambers are integrating devices.

Radiation measurements underwater were made by placing two or three thimble chambers in a weighted 2 oz. bottle. The bottle was then lowered into the water using a nylon line and positioned at contact with the component to be measured. The time of exposure was then kept with a stop watch.

The Model L-64 Roentgen Meter is a quartz fiber type of electrostatic voltmeter. Together with the Roentgen chamber models L-120 through L-129 (thimble chambers) it comprises a method of accurate measurement of gamma rays. The range of the chambers is from 1 mr to 1000 R.

The inherent error of measurement using the voltmeter is small compared with the error associated in positioning the chambers. Movement of the bottle containing the chambers during the measurement effects the result because of attenuation in the water environment.

In general, there was good reproducibility. For example, the shroud measured 8100 R/hr. When measured again one month later at approximately the same position a measurement of 8400 R/hr was obtained. We experienced some difficulty measuring the control rods near the bottom of the cruciform blade. The bottoms of the rods were very radioactive and positioning the bottle was critical. There was not good reproducibility measuring control rods.

The measurement made underwater do not necessarily reflect the gamma field which would be encountered if the component were suspended in air. The reactor internals are large and constitute large sources when suspended in air. Underwater the detector sees only a small portion due to the attenuation by water.

Pieces of a steam separator and small foreign pieces of metal collected with the underwater vacuum cleaner were examined with the gamma spectrometer. Gamma emitting isotopes identified were cobalt-60, iron-59, zinc-65, and manganese-54. The zinc-65 is contamination on the surface of the components.

P. Radiation Levels

Radiation measurements were made in air using calibrated gamma survey meters (CP-TP and Model 2586 Nuclear-Chicago) and underwater using the Landsverk Roentgen Meter Model L-64 and associated thimble ionization chambers. The gamma survey meters are rate meters while the thimble ionization chambers are integrating devices.

Radiation measurements underwater were made by placing two or three thimble chambers in a weighted 2 oz. bottle. The bottle was then lowered into the water using a nylon line and positioned at contact with the component to be measured. The time of exposure was then kept with a stop watch.

The Model L-64 Roentgen Meter is a quartz fiber type of electrostatic voltmeter. Together with the Roentgen chamber models L-120 through L-129 (thimble chambers) it comprises a method of accurate measurement of gamma rays. The range of the chambers is from 1 mr to 1000 R.

The inherent error of measurement using the voltmeter is small compared with the error associated in positioning the chambers. Movement of the bottle containing the chambers during the measurement effects the result because of attenuation in the water environment.

In general, there was good reproducibility. For example, the shroud measured 8100 R/hr. When measured again one month later at approximately the same position a measurement of 8400 R/hr was obtained. We experienced some difficulty measuring the control rods near the bottom of the cruciform blade. The bottoms of the rods were very radioactive and positioning the bottle was critical. There was not good reproducibility measuring control rods.

The measurement made underwater do not necessarily reflect the gamma field which would be encountered if the component were suspended in air. The reactor internals are large and constitute large sources when suspended in air. Underwater the detector sees only a small portion due to the attenuation by water.

Pieces of a steam separator and small foreign pieces of metal collected with the underwater vacuum cleaner were examined with the gamma spectrometer. Gamma emitting isotopes identified were cobalt-60, iron-59, zinc-65, and manganese-54. The zinc-65 is contamination on the surface of the components.

Measurements

Component	Location	Approximate R/hr
Demister	Contact	0.3
Demister	Contact	0.5
Shroud	Inside at contact with support port east	8100
Shroud	Outside contact with middle south	3500
Shroud	Inside at contact with support port east	8400
Holddown	Inside at contact with a middle section	23
Holddown	Inside at contact with a bottom section	350
Holddown	Inside at contact with a top section	5.4
Holddown	Outside at contact with a middle section	1.5
Control Rod	#6, Group 3, contact with middle	10
Control Rod	#6, Group 3, contact with bottom	20
Control Rod	#6, Group 3, contact with top of blade	3
Control Rod	#6, Group 3, contact above transition	0.18
Control Rod	#6, Group 3, contact with shaft	0.01
Control Rod	Group 2 rod, contact with bottom	1500
Control Rod	Group 2 rod, contact with top of blade	560
Control Rod	Group 2 rod, contact with middle	1400
Control Rod	#16, Group 4, contact with middle	0.6
Control Rod	#16, Group 4, contact with top	0.6
Control Rod	#7, Group 4, contact with middle	15
Control Rod	#7, Group 4, contact with bottom	150
Control Rod	#7, Group 4, contact with top of blade	7.5
Control Rod	#15, Group 5, contact with middle	14
Control Rod	#15, Group 5, contact with top of blade	9
Control Rod	#4, Group 3, contact with stem	0.015
Control Rod	#4, Group 3, contact at transition	0.060

Component	Location	Approximate R/hr
Grid Plate Bolt	#1 contact	0.50
Grid Plate Bolt	#2 contact	0.60
Grid Plate	Contact with chimney surface at marker	105
Grid Plate	Contact with chimney surface opposite marker	100
Grid Plate	Contact with top surface	65
Quad Box	Cr-2, contact outside at top	690
Quad Box	CR-2, contact outside at middle	4300
Quad Box	CR-2, contact outside at bottom	720
Quad Box	CR-9, Group 2 rods, contact outside at top	1080
Quad Box	CR-9, Group 2 rods, contact outside at middle	5200
Quad Box	CR-9, Group 2 rods, contact outside at bottom	600
Boiler Box	BH-4, contact outside at top	155
Boiler Box	BH-4, contact outside at middle	2400
Boiler Box	BH-4, contact outside at bottom	230
Separator Group	Group 12, #34, contact with middle	18
Separator Group	Group 12, #36, contact with bottom	20
Separator Group	Group 12, #35, contact with middle	140
Separator Group	Group 12, #34, contact with chimney	0.4
Separator Group	Group 12, #34, at transition	1.2
Separator Group	Group 12, #34, 2 feet below transition	3.9
Separator Group	Group 12, #34, at inlet nozzle	30
Separator Nozzle	#35, contact	28
Neutron Window	Contact	600
Six Vessel Specimens	3 high flux and 3 low flux, at 6"	20

Component	Location	Approximate R/hr
Separator Vane	#1 (found in pump) contact	0.5
Separator Vane	#2 (found at fuel element flow channel) contact	10
Separator Vane	#3 contact	0.4
Separator Vane	#4 contact	1.0
Separator Vane	#5 contact	0.24
Separator Vane	#6 contact	1.8
Separator Vane	#7 contact	1.3
Separator Vane	#8 contact	30
Separator Vane	#9 contact	0.12
Separator Vane	#10 contact	16
Separator Vane	#11 contact	0.19
Separator Vane	#12 contact	0.30
Separator Vane	#13 contact	0.42
Separator Vane	#14 contact	25
Separator Vane	#15 contact	0.28
Separator Vane	#16 contact	0.22
Separator Vane	#17 contact	0.36
Separator Vane	#18 contact	0.50
Separator Vane	#19 contact	0.50
Separator Vane	#20 contact	0.80
Separator Vane	#21 contact	0.41
Separator Vane	#22 contact	0.20
Separator Vane	#23 contact	0.34
Separator Vane	#24 contact	0.60
Separator Vane	#25 contact	0.50
Separator Vane	#26 contact	0.30

Component	Location	Approximate R/hr
Separator Vane	#27 contact	0.88
Separator Vane	#28 contact	0.25
Separator Vane	#29 contact	0.55
Separator Vane	#30 contact	0.30
Separator Vane	#31 contact	0.25
Separator Vane	#32 contact	0.45
Separator Vane	#33 contact	3.1
Separator Vane	#34 contact	3.0
Separator Vane	#35 contact	0.44
Separator Vane	#36 contact	0.50
Separator Vane	#37 contact	0.25
Separator Vane	#38 contact	1.25
Separator Vane	#39 contact	3.2
Separator Vane	#40 contact	1.8
Separator Vane	#41 contact	0.25
Separator Vane	#42 contact	7
Separator Vane	#43 contact	15
Separator Vane	#44 contact	12
Separator Group	Group 12, upside down, bottom at surface of water vane removed, measured at 6"	8
Separator Group	Group 12, upside down, bottom 14" below pool surface, vane removed	1.2
Recirc Pump	#12 pump at contact	4.5

VI. Containment Leakage Rate Testing

The Class C Leakage Rate Test that was started in October 1967 was completed on January 19, 1968. Leakage of the Reactor Building Cooling Coil Water Return Isolation Valve (CV1087) and the Reactor Building Ventilation Back-up Isolation Valve-Outlet (CV1259) was excessive. Both valves were subsequently repaired; their leakage was below the Tech Spec limits when retested. Leakage through the Vacuum Drain Tank Isolation Valves was 0.5 SCFH at 78 psig; leakage through the Shield Pool Cooling Isolation Valve was also 0.5 SCFH at 78 psig. These were classed as "new leak" since they were not present at the last Class A test.

A second Class C Test was completed on March 6, 1968. A valve disk dowel pin was found leaking on each of the Reactor Building Ventilation Outlet valves. These pins were sealed, thereby reducing the leakage through both the valves to less than 0.5 SCFH. The Main Steam Line Drain Valve leakage was 10 SCFH at 78 psig and was classed as a "new leak". Although the leakage through this valve is acceptable it was repaired after the test was completed.

The future testing program includes a Class A Test during July 1968. An inspection of the Reactor Building Equipment Door is tentatively scheduled to be performed prior to the above Class A test.

VII. Core Monitoring Changes

Just prior to boiler fuel unloading the core was being monitored with three startup channels and the power range channels. At that time (October 30, 1967) Channel 1 & 2 indication was 5.48 cps and 1.59 cps respectively. As boiler fuel unloading started, startup channel indication decreased as expected. Near the end of fuel unloading, Channel 2 was connected to a BF_3 detector placed between the vessel wall and the shield wall and a CIC was placed in the reactor vessel with readout on Channel 6.

Boiler fuel unloading was completed on November 10, 1967. At this time Channel 1 & 2 were indicating 0.5 cps and 0.3 cps respectively. Core monitoring with these two startup channels and the in-vessel CIC continued for the remainder of the calendar year.

During the first week of January 1968, Channel 1 detector was placed between the vessel and shield walls next to the Channel 2 detector. At this time routine "bugging" of Channel 1 & 2 detectors with a 1 curie PuBe neutron source was initiated to demonstrate that they are capable of monitoring neutron flux.

Since boiler fuel unloading, one of the two incore sources have been temporarily withdrawn from the core on three occasions to verify that Channels 1 & 2 are monitoring neutron flux. On two of these occasions, Channels 1 & 2 decreased about a factor of two, and on the third occasion by a factor of 1.4.

On March 11, 1968, Channel 1 detector was removed from the vessel wall and placed on No. 12 Recirculation Pump vault floor at the outside wall. Counts were taken with the detector in this position for about 12 hours. The following day the detector was returned to the vessel wall and again counts taken for a like 12 hour period. From this data it was determined that Channel 1 noise level was about 0.12 cps.

On April 1, 1968, a BF_3 detector was placed in the reactor vessel at the southeast corner of the superheater and connected to Channel 3. During the first week of May a second BF_3 detector was placed in the vessel and connected to Channel 1. Both Channel 1 & 3 gave reliable indication with the above detectors.

VIII. Chemistry and Radiation Experience

A. Chemistry Analysis

During this reporting period the reactor was open to the shield pool and on continuous cleanup via the pool cleanup system. Boiler fuel and some reactor internals are stored in the fuel storage pool which is also on continuous cleanup via the pool cleanup system. Water from both these systems is monitored for pH, conductivity, chlorides, temperature, and gross beta activity. Gross beta activity is usually in the 10^{-5} uc/ml range for both pools and was never higher than 2×10^{-4} uc/ml. Chlorides are usually undetectable ($<$12 ppb) but on occasion levels no higher than 40 ppb were measured. Other parameters remained constant throughout the reporting period.

The four radioisotopes detected by gamma spectrometry were Zn-65, Co-60, Fe-59, and Mn-54. No other isotopes have been identified. Tritium levels were last measured on February 23, 1968, and were in the 10^{-4} uc/ml range.

A crud analysis was performed on material gathered from the reactor vessel with the underwater vacuum cleaner. A complete report of the analysis was presented in section IV, D. It was found that the largest contributor was iron oxide.

There was no off gas flow during this reporting period and therefore no isotopic gas analysis.

B. Cooling Water Chemistry

Pathfinder is presently experimenting with various cooling water treatments to find one which is suitable. Two previous treatments proved expensive and did not give the corrosion control desired. A third treatment is now being tested. This new treatment is a combination corrosion inhibitor and anti-foulant containing no phosphate or chromate. It is a patented product containing combination of organic and inorganic polyvalent ions.

The new operating guidelines for the treatment are:

Chemical Conc	65-75 ppm
Calcium Hardness	800 ppm
Alkalinity	60-80 ppm
pH	7.3-7.7

C. Radiation Measurements

Measurements were made on reactor internals encountered during the present inspection program. Refer to section IV, P of this report for these values.

There are presently four areas designated as high radiation areas. They include the reactor building, the pool cleanup area, the hot side of the turbine building mezzanine floor, and the solid waste storage yard.

The reactor building is a high radiation area due to the presence of pulled recirculation pumps which run as high as 4.5 R/hr at the pump impeller and reactor components stored in the shield pool. The pool cleanup area contains filters and demineralizers which are above 100 mr/hr. The startup heater on the mezzanine floor of the turbine building hot side is 400 mr/hr. There are many barrels containing solid waste located in the waste storage yard which are above 100 mr/hr. All these areas are controlled as required by 10CFR20.

There are other areas designated as radiation areas (radiation levels > 2.5 mr/hr or any smearable contamination above 100 dpm/smear or any airborne activity above limits) which usually exist temporarily. Radiation monitoring has control over such areas and routinely surveys the plant to detect the presence of any new areas.

Because there were no reactor operations there were no general surveys performed (radiation measurements at various power levels).

D. Health Physics

The summary of radiation exposures for 1967 indicated that eight people received greater than one quarter of the yearly limit (5 rem). The highest accumulated exposure was 2250 mr received by one individual.

Work during this reporting period which resulted in significant exposures was as follows:

#12 recirculation pump replacement	1000 mr for 11 people (average 91 mr each person)
#11 recirculation pump pull	655 mr for 6 people

Barrelling accumulated cuno filters	2830 mr for 13 people
	highest individual exposure was 450 mr
Steam Dryer inspection	350 mr for 3 people
Separator vane measurements	370 mr for 4 people

Frequent air sampling and continuous air monitoring did not detect significant amounts of air activity during this reporting period. Spectrums of CAM tapes showed the presence of low concentration of Co-60, Mn-54, and Zn-65.

E. Liquid Waste Monitor

A new liquid waste monitor was put into service. The new monitor consists of a 6 inch diameter pipe volume chamber with a NaI detector mounted on the outside surface. The pipe is nickel plated to prevent corrosion and to facilitate cleaning. The monitor is shielded by lead shot which reduced the background to 2000 cpm. The monitor is easily backflushed to the sump to remove any solids which settle in the chamber.

The monitor was calibrated using liquid waste batch releases and a sensitivity of approximately 1×10^{-5} uc/ml was achieved. The monitor is not used as the primary information on activity levels in the liquid stream. This is accomplished in the lab and the waste is released in batches. The monitor is used to monitor the waste stream and will close the discharge valve immediately if it detects a concentration higher than expected.

F. Solid Waste Handling

The used cotton cartridges (Cuno filters) stored temporarily in the turbine building hot side were barrelled and moved to the new waste storage yard. The cartridges were placed on pegs anchored in 55 gallon drums and moved outside where the barrels were filled with concrete.

Concrete lined barrels were used for disposal of cartridges from the pool cleanup filters. These cartridges are radioactively hotter than those from the feedwater system and shielding is required for handling.

The technique used for barrelling Solka Floc, filter aide material, waste is routinely being used. Rather than a homogeneous mixture of cement and Solka Floc we now line barrels with concrete leaving a cavity for more concentrated filter bags. The Solka Floc is filtered through muslin bags which releases much of the liquid which would ordinarily be barrelled. The cavity is partially filled with dry cement to tie up any liquid remaining. Barrels are moved outside where they are capped with concrete before storage.

An underwater vacuum cleaner was designed and built to clean "crud" from the reactor vessel and retrieve separator vane pieces. Muslin bags employed underwater were used to collect the high activity material. A total of 9 of these bags were used to date and handled as solid waste. Handling was tedious due to the high gamma field (some were from 20 to 40 R/hr at contact). Special drums were constructed which contained cement and lead shielding to reduce the radiation levels on the outside to < 200 mr/hr. The vacuum cleaner bags were moved to the storage pool via the fuel transfer carriage and then lifted into the drums using long handling tools. The drums were then capped with lead shot and concrete. The highest exposure received by any one individual during this type of waste handling was <10 mr/hr.

The new waste storage yard is now completed and being used. The yard is 75 by 75 feet and surrounded by a Cyclone fence. A boom is located in the middle of the yard for moving barrels. There are presently about 125 barrels ready for shipment. The barrels are being stored for shipment in economical lots.

G. Shipments

Three shipments containing vessel specimens, steam separator pieces, steam separator nozzles, and pieces of the holddown structure were shipped to Battelle Memorial Institute. Each time a shipping cask was furnished by Battelle. The total activity was calculated to be 17.3 curies.

H. Radiation Incidents

There were no radiation incidents during this reporting period.

I. Routine Releases

Liquid released to the Big Sioux River

Month	Activity (uc)	Average Concentration (uc/ml)	*Fraction of MPC
November	2.08×10^4	6.51×10^{-8}	0.0022
December	1.11×10^4	3.36×10^{-8}	0.0011
January	1.05×10^4	3.18×10^{-8}	0.0011
February	1.65×10^4	5.32×10^{-8}	0.0018
March	5.93×10^3	2.45×10^{-8}	0.0008
April	3.48×10^3	2.85×10^{-8}	0.0009

*MPC of 3×10^{-5} uc/ml based on most restrictive isotope identified by routine gamma spectrometry analysis.

The average concentration was determined by gross beta times a correction factor for Zinc-65. The efficiency for counting Zinc-65 with our counter is less than that of a pure beta emitter.

The average concentration is that which was released from the plant without taking account of dilution in the river.

Tritium levels were last measured February 23, 1968, and showed only very low concentrations in liquid waste before dilution. Because of the low levels and the fact that we are not operating, tritium samples are being processed less frequently.

Gaseous and Particulate Releases:

There was no off gas flow during this reporting period. Stack gas and particulate monitor did not detect releases above background.

J. Off Site Monitoring

A study of the environmental data from November 1, 1967, to April 1, 1968, showed no increases or deviations. Air sampled continuously at Vermillion, Sioux Falls, and Pathfinder has been <1.0 pc/m^3. Fallout samples from the three locations was about 1.0 nc/m^2. There is no significant difference in water sampled upstream and downstream from the effluent ditch. Values are usually around 2 to 3 pc/l. Weekly milk samples run $<$ 1000 pc/l gross beta minus K-40. No leafy plant or sludge samples were processed during this period. All the environmental film badges showed no detectable radiation.

IX. Major Plant Maintenance and Construction

The following major maintenance and construction items were completed during this reporting period:

Installed a jib crane and completed the lighting and alarm circuits in the radwaste storage yard.

Replaced the broken louvres on the cooling tower. The lower row of louvres were changed from 3/16" to 3/8" thickness to better withstand ice buildup.

Repaired No. 11 circulating water pump discharge valve. The original cast iron bearing had seized and contributed to the difficulties described in Appendix B, Unusual Occurrence Report No. 17 in the previous six months report.

Removed the gooseneck from the reactor water column, installed a blank flange, and modified the pipe hanger support. The gooseneck had previously provided penetrations for steam sampling and the non-continuous superheater thermocouple leads which were removed during this shutdown.

Installed new liquid waste monitor piping and a new shielding pig. The new volume chamber consists of a piece of 6" pipe which was bored, honed, and chrome plated to prevent the deposition of crud from the liquid waste stream.

Pulled and sectioned three condenser tubes for analysis by the NSP testing lab. These tube locations were plugged with stub tubes.

Removed the waste drumming concrete mixer from the fuel handling building. The operation has been changed to decanting into filter socks and using pre-cast drums for disposal.

Completed repairs on No. 11 reactor feed pump motor. The outboard journal was resurfaced and a new bearing and oil rings were installed.

Piped all sample lines on the pool cleanup systems to a sample sink outside the shield wall.

Overhauled No. 11 air compressor which had failed due to a broken vane in the second stage.

Performed a complete inspection and routine maintenance of the turbine building crane.

Performed complete inspection and maintenance of the instrument air drying system.

Removed the failed "B" phase transformer from the 12.5/480 KV screen house transformer bank. The secondary leads from the bank to the screen house were replaced with insulated wire.

Disassembled CV1008 (condensate pump recirculation). Dye checked and inspected the valve body. The valve body was found to be very thin and porous. A new valve has been ordered.

Performed maintenance on the following isolation valves:

a. CV-1086 - cooling water return. Cleaned and lapped the plug and seat.

b. CV-1251 - main steam above seat drain. Machined the plug and lapped the seating surfaces.

c. CV-1087, CV-1088, CV-1258, CV-1259 - ventilation isolation. Reseated and sealed the dowel pins which hold the stub shafts to the wafer. Cleaned the valve body rubber liners with dry graphite.

Overhauled the mechanical seal on No. 11 reactor recirculation pump.

X. Personnel Changes

A. General

The Pathfinder Plant technical supervisory staff organization remained essentially as reported in the previous Six Month Report (NSP 6704). The plant organization now includes 57 personnel.

The plant organization is shown on the diagram on page 100.

B. Plant Personnel Changes

The following personnel changes were made during this six month report period:

1. Promotions

(a) Mr M N Bjeldanes was promoted to Assistant Plant Superintendent effective January 1, 1968.

(b) Mr T E McFadden was promoted to Plant Results Engineer effective January 1, 1968.

(c) Mr E L Watzl was promoted to Health Physics Supervisor effective January 1, 1968.

2. Transfers

The following personnel were transferred to the Monticello staff:

(a) Mr C E Larson effective January 1, 1968.

(b) Mr M H Clarity effective January 1, 1968.

(c) Mr L R Eliason effective January 1, 1968.

(d) Mr G H Jacobson effective January 1, 1968.

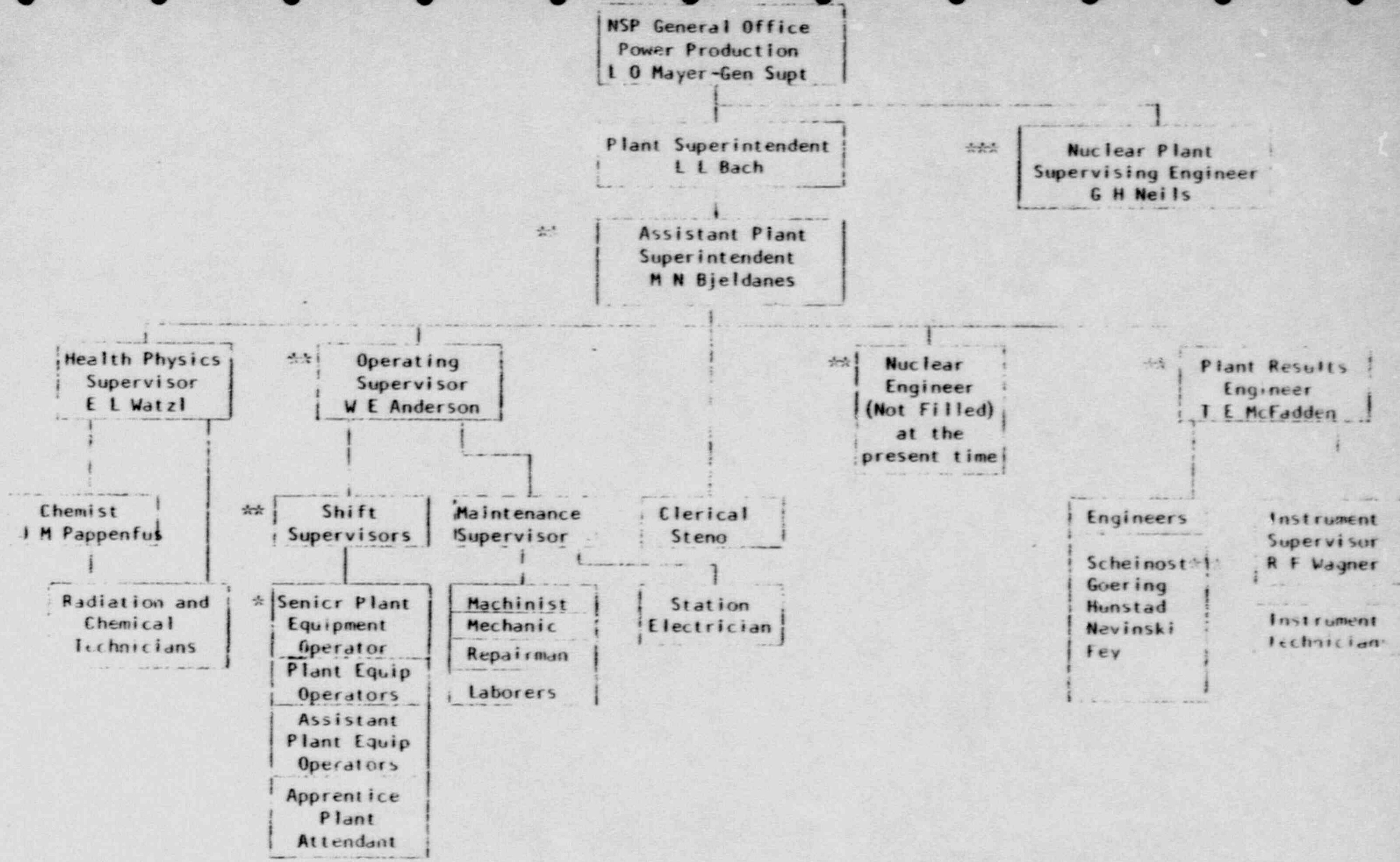
(e) Mr M H Voth effective March 1, 1968.

(f) Mr R D Jacobson effective March 1, 1968.

(g) Mr W A Shamla effective March 1, 1968.

(h) Mr W J Hill effective May 1, 1968.

(i) Mr M F Dinville effective May 1, 1968.



*AEC Licensed Operator positions
**AEC Licensed Senior Operator positions
***AEC Licensed Senior Operator

Northern States Power Company
Pathfinder Atomic Power Plant
Organizational Diagram
May 19, 1968

3. New Personnel

(a) Mr Martin F Dinville was assigned to the Pathfinder Results Staff on January 1, 1968. Mr Dinville is a graduate of Iowa State University (BSME). He came to NSP with six years of Results Engineering experience.

(b) Mr Fred L Fey started working at Pathfinder on January 15, 1968. Mr Fey holds a BS degree in physics and math from the University of Wisconsin and a Masters Degree in Health Physics and a Masters Degree in Public Health from the University of Michigan. He came to NSP with two years nuclear engineering experience obtained at Los Alamos Scientific Laboratory.

(c) Mr Douglas E Nevinski joined the Pathfinder Results Staff on February 1, 1968. Mr Nevinski is a recent graduate (nuclear engineering) of the University of Wisconsin.

D. Operator Licensing

The licensed reactor operator status is:

Senior Reactor Operators (22)

M N Bjeldanes	- Assistant Plant Superintendent
W E Anderson	- Operating Supervisor
T E McFadden	- Plant Results Engineer
R L Scheinost	- Engineer
W A Sparrow	- Shift Supervisor
J B Brokaw	- Shift Supervisor
R A Mielke	- Shift Supervisor
R T McKaughan	- Shift Supervisor
S L Pearson	- Shift Supervisor
L W Severson	- Senior Plant Equipment Operator
L V Triebwasser	- Senior Plant Equipment Operator
R D Emerson	- Senior Plant Equipment Operator
F J Schober	- Senior Plant Equipment Operator
H Seibel	- Senior Plant Equipment Operator
M J Baik	- Plant Equipment Operator
D L Magill	- Plant Equipment Operator
D E Severson	- Plant Equipment Operator
D W Cragoe	- Plant Equipment Operator

Reactor Operators (4)

R C Dolge	- Assistant Plant Equipment Operator
E W Kruse	- Assistant Plant Equipment Operator
R S Holthe	- Assistant Plant Equipment Operator
V R Stoefen	- Assistant Plant Equipment Operator

D. Summary

NSP management has reviewed each of the above changes and finds that the organization, as it exists on May 19, 1968, provides the technical qualifications necessary for safe plant operation.