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DESIGN ENGINEERING REPORT

WNP-2 FEEDWATER PIPING THERMAL DEFLECTION EVENTS

Washington Public Power Supply System Nuclear Plant No. 2

Richland, Washington

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WNP-2 Feedwater Piping Thermal Deflection Events

Executive Overview

A sequence of events occurred during power ascension testing of WNP-2 that resulted in unanticipated Reactor Feedwater Piping deflections which damaged pipe supports.

The first event in June, 1984, buckled a rigid strut and was attributed to a binding condition in the support combined with large predicted pipe thermal displacements. The buckling displacement was not excessive. Exact plant conditions at the time of the event were not known.

In August on restart from a 5 day hot-standby, three major supports were failed due to pipe deflection. From temperature recorders it was evident Reactor Water Cleanup water at temperatures greater than 400°F had recirculated through a portion of the RFW piping prior to the event. Upon injection of cold condensate (approximately 100°F) into the piping for reactor makeup, the support failures occurred. From parametric analysis with structural models it was concluded that top-to-bottom temperature gradients in long horizontal runs of RFW pipe had caused the three inch deflection and 80,000 lb force which was required to fail the supports. Following this event, instrumentation was installed to measure thermal gradients and pipe displacements at locations relevant to this postulated condition. Restart from hot-standby was precluded pending further engineering evaluations and data accumulation. Studies initiated included redesign of the pipe support concept to accommodate large pipe deflections.

In September following a scram, the RFW piping again experienced large deflections upon introduction of cold condensate into the hot RFW piping. The temperature of the RFW piping increased slightly post scram due to RWCU backflow. Instrumentation confirmed top-to-bottom temperature gradients of approximately 200°F. Pipe deflections were less than in August but one spring support bottomed and failed. Pipe support redesigns were installed to accommodate pipe deflections since it was evident deflections would reoccur throughout normal plant operation.

Fatigue Usage

Engineering studies were initiated as a result of these events to address the operational restraints required to minimize thermal gradients in RFW piping. Fatigue usage factor limits on RFW piping between containment isolation valves had been established by MEB 3-1 (NUREG 75/087) at 0.10. This limitation currently precludes restart from hot shutdown unless operator actions are taken immediately following scram to transfer RWCU from the RFW pipe. Administration of the current limit requires either thermal cycling of the RPV or diversion of the Operator from post scram safety parameters to limit RFW thermal transients in accordance with MEB 3-1. The program for resolution on WNP-2 will include a request to the NRC to increase the arbitrary 0.10 usage factor limit in the interest of overall plant safety and performance considerations.

RWCU Backflow

RFW pipe bowing occurs on WNP-2 because of top-to-bottom pipe temperature gradients. Thermal gradients will occur because reactor feedwater must service the RPV level demand under conditions in which the flow rates may be very low and the condensate temperature significantly less than the temperature of the water existing in the RFW piping and the RPV. RWCU backflow caused the stratification during a 5-day hot-standby. RWCU backflow can amplify the gradient following any shutdown. Regulatory change resulted in moving the RWCU return to RFW outboard of the containment check valves, a design modification not found on earlier BWR's. Further regulatory change resulted in substitution of a motor operated valve in place of a third check valve which would have precluded RWCU back flow in RFW piping on the BWR-4. Regulations which increased margins for improbable accidents clearly contributed to RFW pipe deflection events which have occurred at WNP-2 to date. There are no safety-related impacts resultant from the reactor feedwater piping deflections observed at WNP-2, however, these events have impacted plant availability.

I. INTRODUCTION

To date, three separate operating events have resulted in pipe support damage in a non-safety-related portion of the Reactor Feedwater System (RFW). The first and last events occurred after reactor shutdown from approximately 25% and 75% power respectively. The second and most damaging event occurred during power ascension after the plant had been in hot standby (approximately 1% power) for several days. These support failures could not be explained by traditional thermal or water hammer analyses; although it is clear that their failures were due to very large displacements.

In addition to the damaged supports, failure of several mechanical snubbers located in an adjacent portion of the affected pipe were discovered during the same time frame. The snubber failures have been attributed to void induced water hammer. This report is limited only to the failed supports and the phenomenon attributed to causing their failure. Each event is described and analyzed. Actions taken to isolate the cause are described. Subsequent support modifications initiated to accommodate the phenomenon are described and future plans are identified.

In the following discussion, it is important to note that the affected portion of the RFW piping is non-safety-related and is designed to ANSI B31.1 criteria and that this system had previously undergone several startups and shutdowns without damage.

II. CONCLUSIONS

Pipe support damage was determined to have been the result of very high loads and displacements at the support locations. Traditional methods used to evaluate thermal expansion and/or water hammer effects do not predict sufficient displacement to cause the observed failures. Rather, it was found that large top to bottom temperature gradients in a long horizontal run of piping were the cause of the large displacements. Temperature differences of up to 200°F were measured between the top and bottom of the pipe with the higher temperature being on top. Thermal expansion of the top of the pipe exceeds that of the bottom resulting in the pipe assuming a bowed down shape with several inches of unrestrained vertical displacement at the endpoints of the horizontal run. Displacements in excess of support capability occurred at several locations. Figure 1 shows the general piping arrangement and Figure 2 illustrates the concept of bowing as described above.

The large thermal gradients are the result of slow injection of relatively cold (100°F) condensate/feedwater into an initially hot (400°F) horizontal run of feedwater piping. The cooler water stratifies along the bottom of the pipe at low velocities while the hotter water remains at the top of the piping. Upon reaching a vertical run of piping, the cold water fills the entire area of the vertical pipe displacing the hotter water upward. Thus, the vertical runs of piping can be treated by traditional analytical methods.

Conditions as described above can occur in a variety of ways, an example of which is startup from a hot shutdown condition. In addition, the same effect can be generated by injecting hot Reactor Water Cleanup (RWCU) into a cold RFW pipe. The hot water stratifies at the top of the pipe, the colder water is displaced to the bottom and the horizontal piping run thermally expands to a bowed down shape.

In order to prevent future pipe support damage, the pipe support system was redesigned to accommodate large displacements. This was accomplished by replacing certain rigid restraint with large displacement capability spring supports and snubbers. The result being a more flexible support system during operational transients.

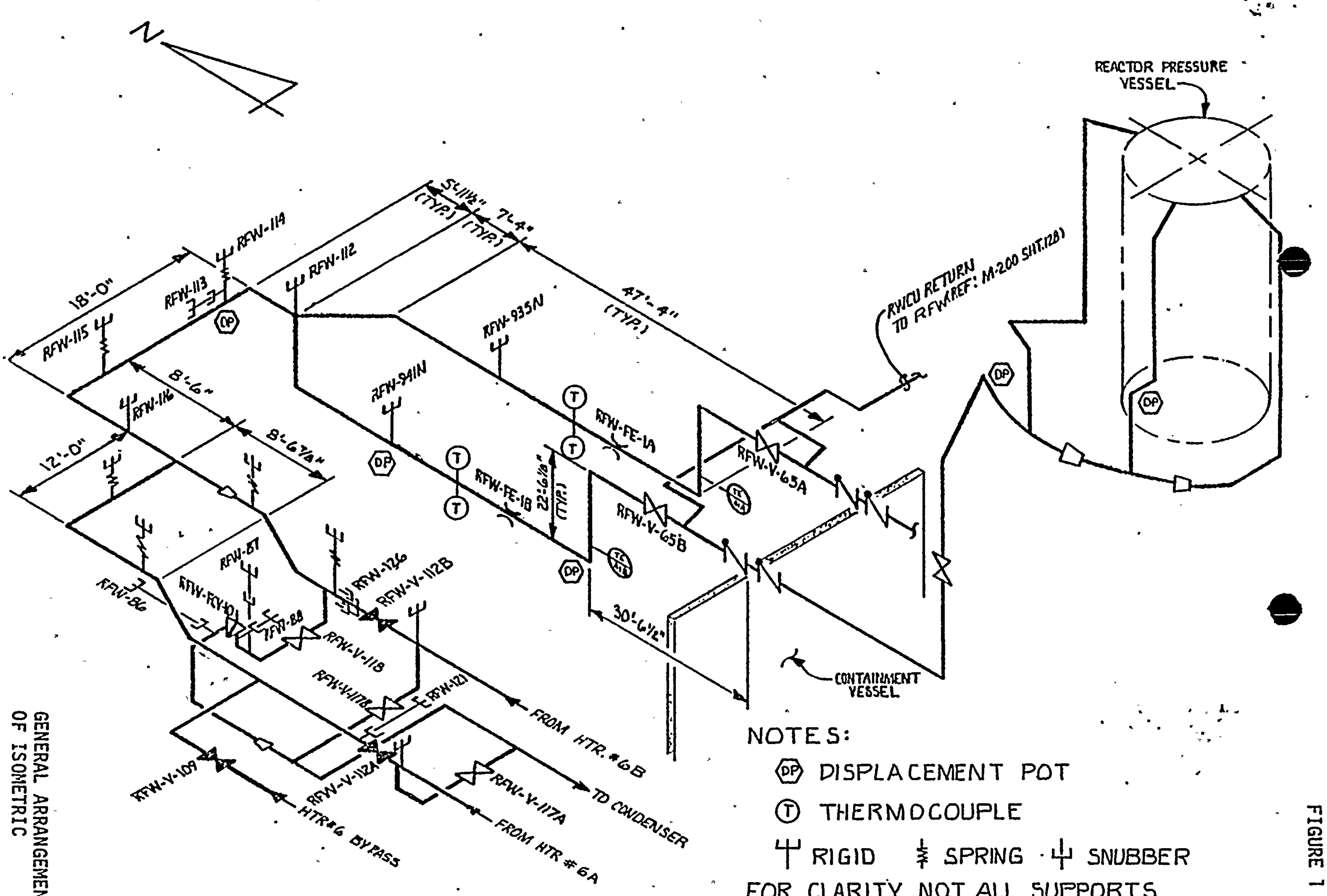
It should be noted that no unique characteristic has been identified that would suggest that the thermal stratification phenomenon discussed herein is applicable only to WNP-2. Rather, it would appear that all that is required to generate the phenomenon is low velocity injection of cold water into a long horizontal hot pipe or conversely low velocity injection of hot water into a long horizontal cold pipe. These parameters are not necessarily unique to WNP-2 or to Boiling Water Reactors.

The analyses and field data show that the pressure boundary integrity was not impaired by the event of August 22. The analyses show all piping forces, moments, and the consequent stresses remain within allowable limits under bounding case studies of the reactive (i.e., failed) support loads and deflections. Field data shows that in all cases the system was easily rebalanced and realigned to its original position following repair of the damaged supports.

III. DESIGN CONSIDERATIONS

A. Piping System Design

Refer to Figure 1 for the general piping system design. The major portion of the piping involved is between the the final stage of feedwater heaters, identified as heaters 6A and 6B, and the outboard containment isolation valves RFW-V-32A and 32B. During low power reactor operation, makeup water flows into this section of pipe through flow control valve RFW-FCV-10. During higher power operation RFW-FCV-10 is closed and flow is routed through valves RFW-V-112A and 112B. Flow continues into a section of 30 inch pipe, which is intended to provide mixing of the water leaving heaters 6A and 6B to assure uniform water temperature into the reactor. After the 30 inch section the flow branches at a wye into two-24 inch sections of pipe. These two long straight runs are flow straightening sections which provide uniform flow through reactor level control flow elements RFW-FE-1A and 1B. Flow is then directed upward and then back horizontally through the steam tunnel, and through the containment isolation valves on its way to the reactor. Immediately upstream of the outboard isolation valves, RWCU flow is injected into the RFW piping through a thermal sleeve branch connection in the upper portion of each RFW line.



NOTES:

DP DISPLACEMENT POT

T THERMOCOUPLE

RIGID SPRING SNUBBER

FOR CLARITY NOT ALL SUPPORTS ARE SHOWN

GENERAL ARRANGEMENT OF ISOMETRIC

FIGURE 1

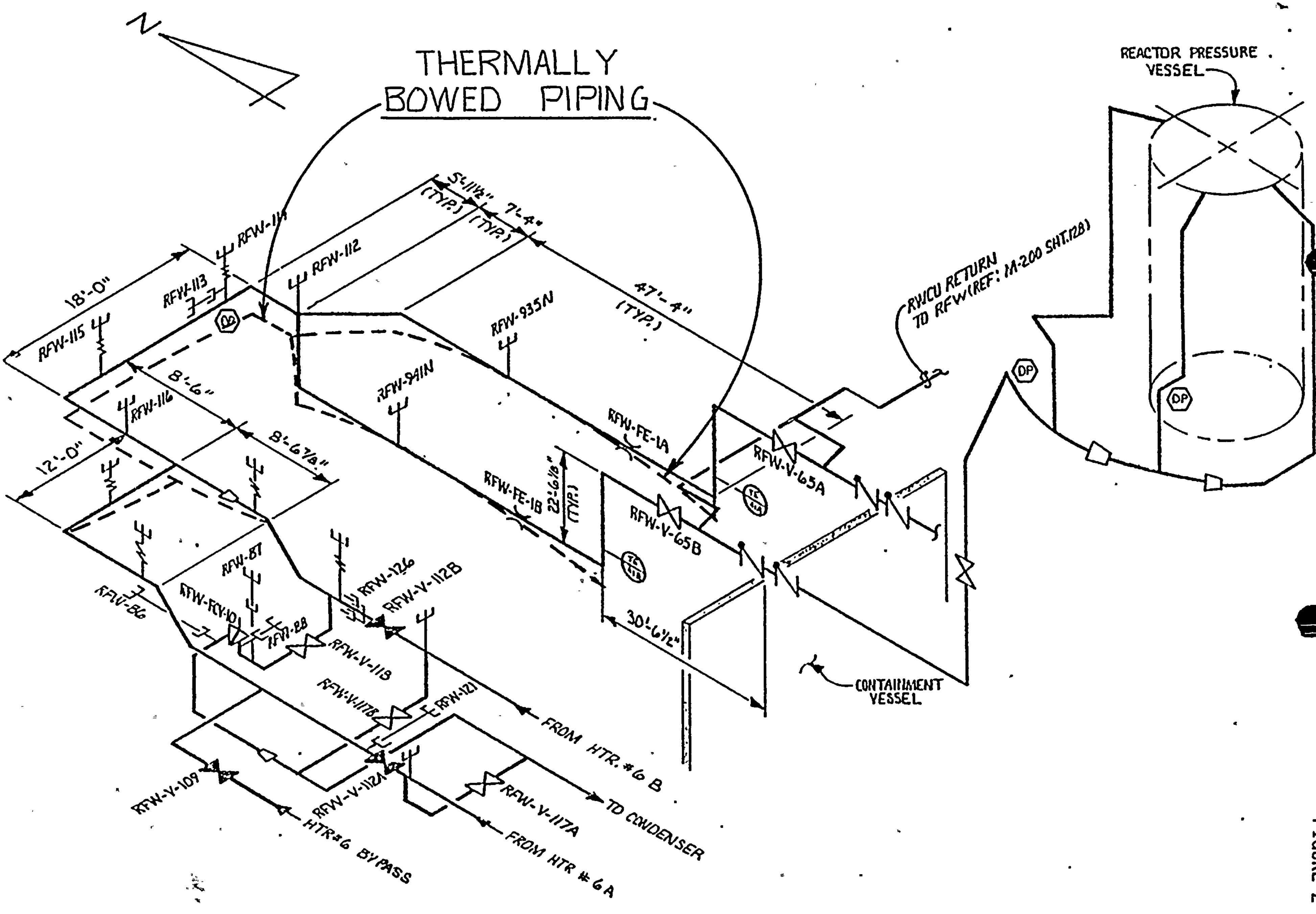

N
THERMALLY
BOWED PIPING
REACTOR PRESSURE
VESSEL
RWCU RETURN
TO RFW (REF: M-200 SHT.128)
DP
DP
DB
18'-0"
12'-0"
8'-6"
8'-6 7/8"
5'-11 1/2"
(TYP.)
7'-4"
(TYP.)
47'-4"
(TYP.)
22'-6 1/8"
(TYP.)
30'-6 1/2"
RFW-115
RFW-114
RFW-113
RFW-112
RFW-116
RFW-935N
RFW-941N
RFW-FE-1A
RFW-FE-1B
RFW-V-65A
RFW-V-65B
RFW-87
RFW-86
RFW-88
RFW-FCV-101
RFW-126
RFW-V-112B
RFW-V-118
RFW-V-117B
RFW-121
RFW-V-109
RFW-V-112A
RFW-V-117A
FROM HTR. #6 B
TO CONDENSER
FROM HTR #6A
HTR #6 BYPASS
CONTAINMENT
VESSEL

FIGURE 2

B. Support System Design

Burns & Roe, Inc., (BRI) the Architect Engineer for the Plant, completed the RFW piping system design both outside and inside containment. The RFW piping system design outside the primary containment isolation valves is designated as non safety-related Quality Class II piping and complies with the requirements of the Power Piping Code, ANSI B31.1. Utilizing the ADLPIPE computer code, this piping system, or anchor group, was analyzed by BRI for pressure, deadweight, thermal, and seismic load cases.

The BRI design basis thermal analysis considered only one load case. This case considered the piping to be at a uniform temperature of 420°F, which would be experienced during full power operation. The basis for this analysis approach was that the maximum temperature case yielded maximum expansion and therefore maximum thermal piping stresses and support loads. In the absence of thermal bowing gradients, the BRI design basis is conservative and generally accepted as industry practice.

IV. EVENT ANALYSES

Three separate events occurred between June 20 and September 10, 1984 which resulted in RFW piping support damage in the same general location. The primary emphasis of this report is on those supports and snubbers damaged by piping bowed due to thermal gradients. Several additional snubbers were found damaged in a related portion of the RFW system. However, those snubber failures have been attributed to classical water hammer events which occurred during a different time frame and operating mode.

A. Event 1, June 20, 1984

1) Event Summary

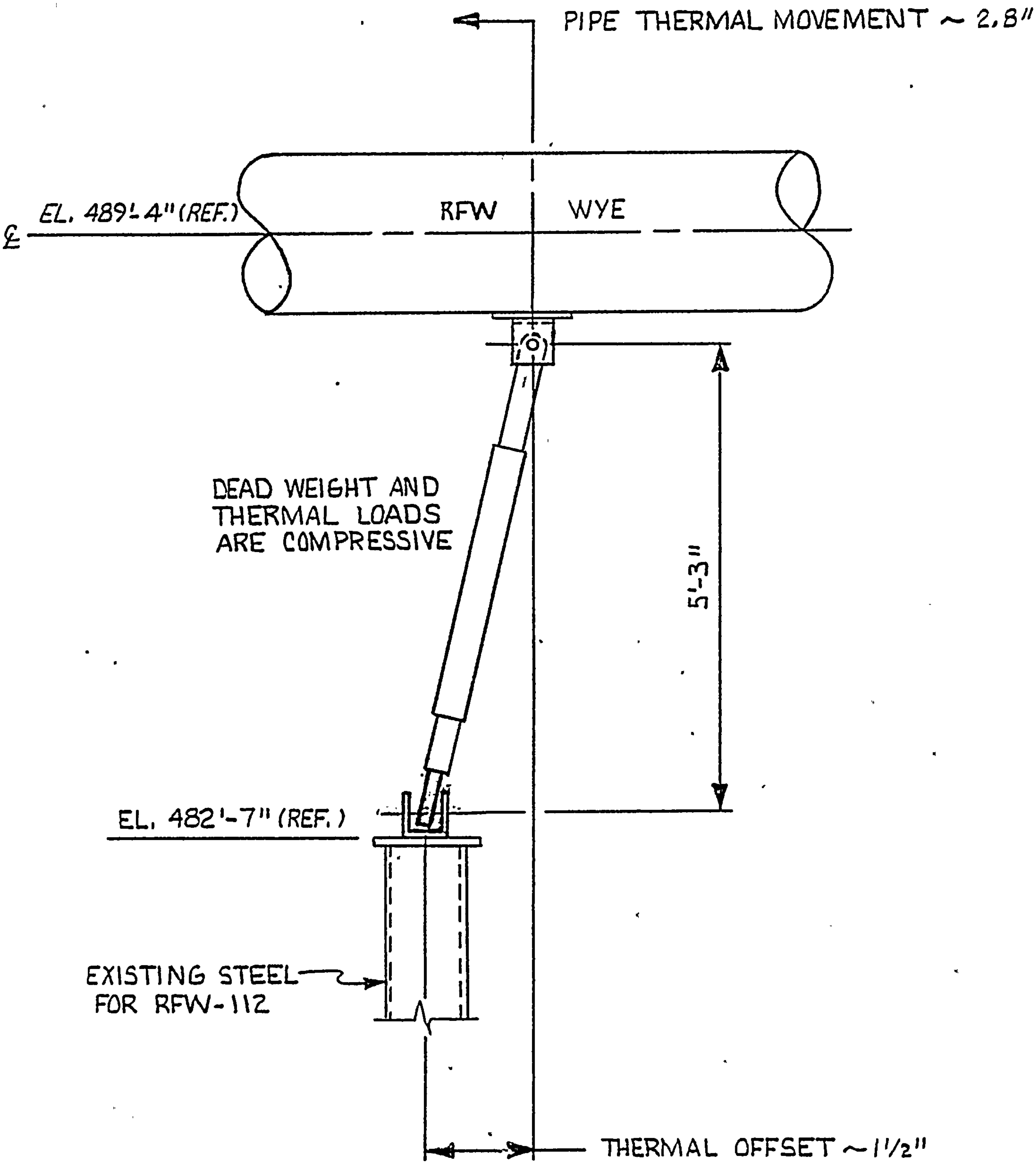
The following sequence of significant events led to the discovery of the failure of hanger RFW-112. On June 15, 1984, an inspection of all the subject hangers and snubbers was completed in accordance with 25% power testing requirements. At that time all the supports and snubbers were found to be mechanically sound.

Date	Time	Event
6/15/84	-	RFW-112 was inspected with the plant at approximately 25% power and found to be mechanically sound.
6/16/84- 6/19/84	17:36	The plant was running between 10-25% power with no shutdowns.
	17:36	A generator load rejection test was performed.
	18:03	The generator was again synchronized to the grid; turbine bypass valve #3 would not close.
	21:05	Reactor shutdown commenced.
	22:18	The turbine/generator was removed from the line.
	22:40	Flow was established through valve RFW-FCV-10 (in auto) to control reactor level, reactor feedpump A provided makeup water.
	23:40	The reactor mode switch was placed in startup/hot standby.
6/20/84	00:02	The MSIVs were shut and the condensate booster pumps secured.
6/20/84	-	Hanger RFW-112 was found buckled.

2) Event 1 Evaluation

On June 20, with the reactor in hot standby, hanger RFW-112 was found buckled. The immediate cause of the failure was uncertain and a thorough inspection of adjacent hangers and piping was completed while looking for clues. No other damaged hangers or snubbers were found and no evidence of pipe movement could be found at points where the RFW piping insulation was in close proximity or touching adjacent structures or piping. Although initially suspected, it was therefore concluded that water hammer was not the mechanism which caused the failure. Further examination of the damaged strut revealed gouging marks on the attachment, or paddle ends. This evidence was combined with a review of the piping thermal movements, reactive loads, and the general design concept of hanger RFW-112. The conclusion was that failure resulted from excessive preload on the strut combined with paddle binding within its attachment end brackets as the piping system was thermally expanding in the plane of the long horizontal pipe runs. (See Figure 3) To help confirm this conclusion, motion detectors were installed in the vicinity of hanger RFW-112 and signals routed to a recorder.

FIGURE 3



THERMAL OFFSET OF RFW-112

3) Structural Evaluation

As a result of the buckled strut, analyses were performed to verify the design basis analysis and pressure boundary integrity. These analyses found no anomalies and that the piping system was unaffected by the event. The buckled strut was removed and replaced with a similar strut. Re-alignment of the system to its original position was easily achieved, which supported analysis conclusions that the piping system (pressure boundary) was unaffected by the thermal transient. To preclude reoccurrence of hanger RFW-112 failure, the deadweight compressive load was relieved from the strut by minor adjustment of adjacent spring hangers. By removing the deadweight preload, the potential for strut binding was judged to be mitigated and enhanced freedom of movement for the large thermal movement (approximately 2.6 inches) would thus be attained. This action was considered adequate while the Supply System's Stress Group reanalyzed this anchor group in an effort to totally eliminate hanger RFW-112. This work was prompted by the analysis conclusion that hanger RFW-112, as originally designed, was a marginal concept for supporting the piping deadweight load. Specifically, utilization of a bottom mounted, compressively loaded, slender sway strut which incorporated large offsets to accommodate thermal movements was deemed not advisable, see Figure 3. Finally, it should be noted that it was not BRI's design practice to utilize sway struts as primary compressive elements with large imposed thermal movements, and as such hanger RFW-112 was essentially unique in the plant pipe support design.

B. Event 2, August 22, 1984

1) Event Summary

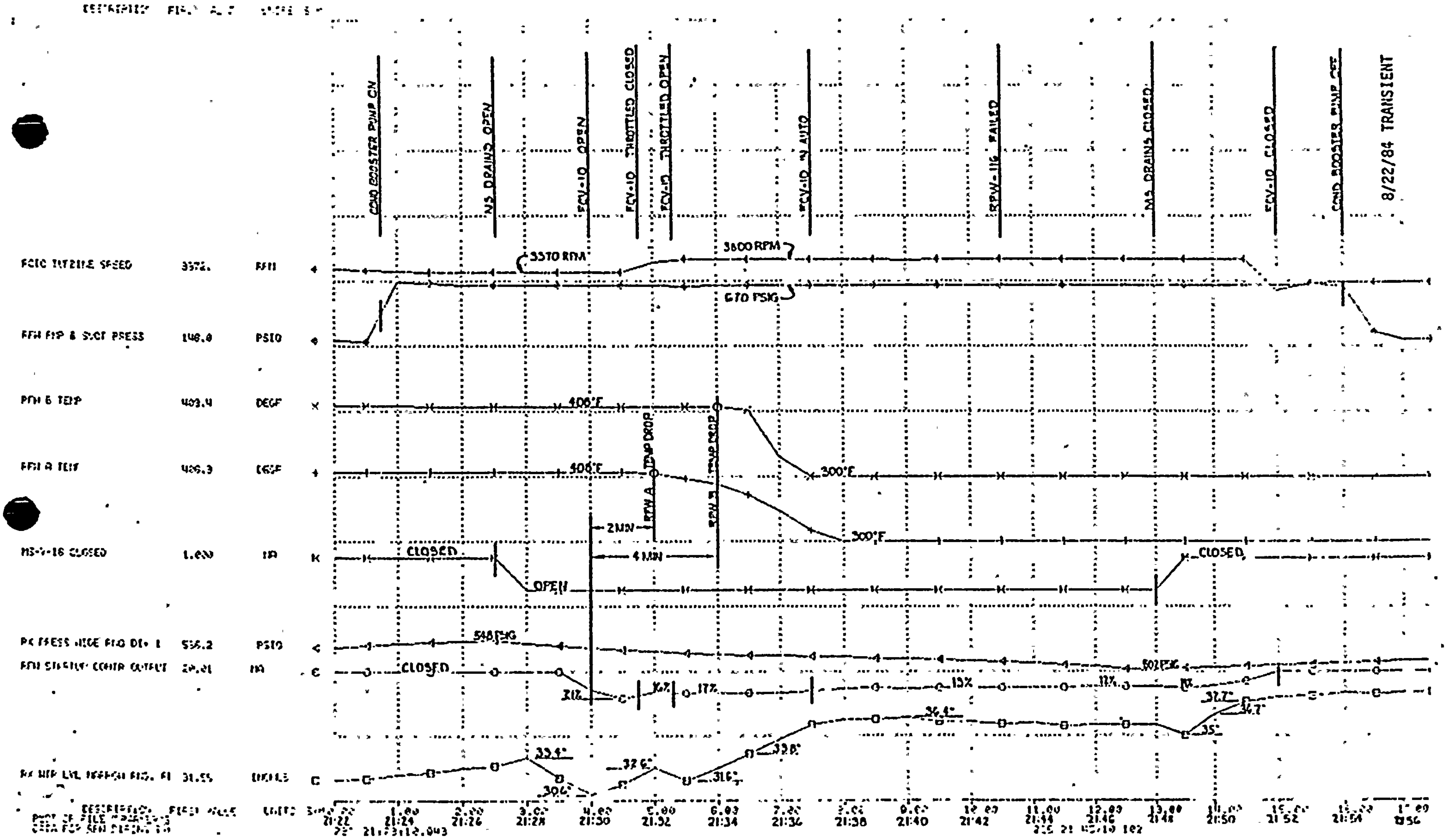
The following summarizes significant events leading up to and following the failure of several feedwater piping supports on Wednesday, August 22, 1984. The plant had previously been operating at about 60% power and was shutdown during testing. Attempts to restart were delayed pending resolution of high conductivity in the condensate system. For approximately five days the reactor was held in hot standby (approximately 1% power) with RWCU being returned to the vessel via the RFW piping. The condensate and RFW systems were completely shutdown. Refer to Figure 4 for graphic representation of pertinent data.

Date	Time	Event
8/22/84	20:38	The condensate system was pressurized by starting condensate pump 1C (COND-P-1C).
	20:42	Condensate cleanup cycle was established from the main condenser, through COND-P-1C, through the feedwater heaters and back to the condenser through valve RFW-FCV-15.

Date	Time	Event
	21.22	Condensate Pressure 146 psig Condensate Temperature 90°F Reactor Pressure 536.2 psig Reactor Temperature 470°F Temperature at RFW-TE-41A 406.3°F (Refer to Figure 1) Temperature at RFW-TE-41B 408.4°F (Refer to Figure 1) Main Steam Lines and Drains Isolated Reactor Pressure was being controlled by running the Reactor Core Isolation Coolant (RCIC) Turbine at 3372 rpm Vessel level makeup was being supplied by control rod drive cooling flow
	21:23	Condensate Booster Pump 2A (COND-P-2A) was started raising condensate pressure to 670 psig.
	21:27	The main steam drains were opened placing additional demand on reactor vessel water makeup.
	21:30	Condensate system flow was established to the reactor through valve RFW-FCV-10 at a flow of 200-600 gpm.
	21:32	Temperature at RFW-TE-41A began decreasing.
	21:34	Temperature at RFW-TE-41B began decreasing.
	21:37	Temperature at RFW-TE-41A and 41B went below 300°F (off scale). RFW piping motion detectors inside containment detect movement.
	21:45*	Hanger RFW-116 pulled from the ceiling, hanger RFW-114 attachment welds failed, and hanger RFW-112 was observed buckled.
	21:48*	Flanges on the flow straightening sections began leaking. (3 of 4 sets)
	21:50	Valve RFW-V-65A and 65B closed to isolate reactor piping from RFW piping.
	21:50	Valve RFW-FCV-10 isolated.
	21:55	COND-P-2A secured.
	22:05	COND-P-1C secured.
	22:33	Reactor subcritical
	23:45	All reactor control rods inserted.

*These events were observed by plant personnel who were in the area on unrelated work assignments.

FIGURE 4



COND BOOSTER PUMP ON
MS DRAINS OPEN
FCV-10 OPEN
FCV-10 THROTTLED CLOSED
FCV-10 THROTTLED OPEN
FCV-10 IN AUTO
RPW-116 FAILED
MS DRAINS CLOSED
FCV-10 CLOSED
COND BOOSTER PUMP OFF
8/22/84 TRANSIENT
3570 RPM
3600 RPM
670 PSIG
400°F
300°F
RFW A TEMP DROP
RFW B TEMP DROP
2 MIN
4 MIN
CLOSED
OPEN
548 PSIG
502 PSIG
31%
13%
11%
33.4"
32.6"
30.6"
31.6"
33.8"
36.4"
35"
32.7"
34.7"
21:22
21:24
21:26
21:28
21:30
21:32
21:34
21:36
21:38
21:40
21:42
21:44
21:46
21:48
21:50
21:52
21:54
21:56

2) Event 2 Evaluation

The plant personnel who observed hanger RFW-116 pulling from the ceiling reported that a dull thud was heard as concrete was falling. However, no loud bangs or rapid piping movements associated with classical water hammer were evident.

The recorder which was connected to the previously installed motion detectors was not operating during this event so no additional displacement information leading to this event was available.

Those snubbers which have failure mechanisms identified as 3 and 4 in Table 1 have been determined to have failed independent of the thermal transient failure mechanism. Therefore, these snubbers will not be addressed in detail.

An inspection of this RFW anchor group was immediately begun to check all snubbers for operability and to visually inspect all supports for damage. The major emphasis of this inspection involved piping from the inlet of the No. 6 feedwater heaters to the containment penetrations and the main condenser. In addition, all feedwater supports inside containment were visually inspected, although data from RFW system displacement transducers placed inside containment for the power ascension test program showed piping movements to be within acceptance limits. Table 1 is a summary of hanger and snubber damage and possible failure mechanisms.

TABLE 1

Hanger Damage Found Following the August 22 Event

Component	Type	Failure Mode	Possible Failure Mechanism
RFW-112	Strut	Failed in compression	1
RFW-113	Snubber	Paddle end bent, locked-up in intermediate position	1
RFW-114	Spring	Welds broken in tension	1
RFW-116	Strut	Baseplate pulled from ceiling	1
RFW-87*	Snubber	Locked up in intermediate position	2,3,4
RFW-88	Snubber	Locked up in intermediate position	2,3,4
RFW-91	Snubber	Locked up in intermediate position	3,4
RFW-126	Snubber (2)	Internals stripped	2,3,4
RFW-133	Snubber	Locked up in intermediate position	3,4
RFW-28	Snubber	Locked up in intermediate position	3,4
RFW-31	Snubber	Locked up in intermediate position	3,4
RFW-84	Snubber	Locked up in intermediate position	3,4

Failure Mechanisms

1) Failures resulting from RFW thermal loads.

2) Failure resulting from dynamic buckling of hanger RFW-112.

3) Failure resulting from known cavitation of RFW-FCV-15.

4) Failure resulting from water hammer upstream of RFW-FCV-10 or RFW-FCV-15.

*All snubbers listed in this table are PSA-3, 6 Kip rating, except RFW-113 which is a PSA-10.

Thermal Hydraulics

The fact that the affected portion of the RFW system had just begun recovery from cold shutdown and that no water hammer was believed credible, initially created uncertainty for the failure mechanism. With this in mind, investigations into all feasible failure mechanisms were initiated; they included:

- o Water surge
- o Trapped air
- o Depressurization
- o Thermal Expansion
- o Thermal gradients

Water Surge - Transient flow or water hammer is capable of creating forces in the feedwater piping which could damage the pipe supports. The calculated maximum flow rate in the piping upon opening valve RFW-FCV-10 was 600 gpm. Assuming this flow rate, the fluid velocity in the 10 inch piping at the exit of valve RFW-FCV-10 would have been 3 ft/sec. The velocity in the 24 inch line immediately downstream of valve RFW-V-118 would have been 0.6 ft/sec, and the velocity in the 30 inch section of piping would have been 0.35 ft/sec. The flow splits upon flowing through the wye into two parallel 24 inch piping paths upstream of the containment isolation valves. The velocity in the parallel 24 inch piping would have been about 0.3 ft/sec. These velocities are rather low and even assuming an instantaneous valve opening, with an accompaning step increase from zero to final steady state velocity, the maximum pressure rise would not be extremely large. For example, in the 10 inch piping a maximum pressure rise was calculated to be less than 190 psi. In the 24 inch piping immediately downstream of RFW-V-118 the maximum pressure rise would be about 37 psi, and in the 30 inch piping the maximum pressure rise would be about 22 psi. These transient pressures or pressure waves conservatively assumed a step increase in flow from 0 to 600 gpm in the various piping segments, and also assumed the pipe was filled with water. As all pressures and velocities were calculated to be extremely low, water surge was determined not to be a credible failure mechanism.

Trapped Air - With regard to this event, air upstream of RFW-FCV-10 could potentially create problems upon starting the pumps and opening RFW-FCV-10 by creating water hammer forces in the vicinity of the valve. The condensate system had been shut down for several days. During shutdown, portions of the condensate system tend to drain down trapping air in the piping or creating the potential for a water hammer event upstream of RFW-FCV-10. As no personnel were in the area when RFW-FCV-10 was placed into operation, it is not known if the water hammer did occur. However, when hanger RFW-116 failed, plant personnel were in the area and no piping motion was observed nor a water hammer report heard. In addition, close examination of the piping insulation showed no evidence of water hammer displacements at points of near contact with various plant structures.

The conclusion here was that water hammer was a potential failure mechanism for failed snubbers around and upstream of RFW-FCV-10. However, it was not a credible failure mechanism of the failed spring and rigid hangers.

Depressurization - As the portion of the piping around the damaged hangers was just recovering from cold shutdown, depressurization should not even require consideration. However, it was observed that the temperature elements RFW-TE-41A and 41B just downstream of the flow straightening sections were indicating around 407°F. Further investigations have proven that temperatures of this magnitude were present throughout the flow straightening sections due to recirculation of RWCU flow upstream of the wye. At this temperature, depressurization below 270 psig could result in flashing of the hot water to steam creating a potential mechanism for water hammer.

A review of operational signals and log books proved that at all times the line pressure was maintained above 600 psig by either RWCU system pressure or condensate booster pump discharge pressure. For this reason depressurization was discounted as a credible failure mechanism.

Thermal Expansion - Several conventional analyses were performed in an attempt to produce loadings near the magnitude required to damage the observed failed supports: RFW-112, RFW-114, and RFW-116. Analyses performed included bounding case studies which cooled the 407°F pipe down to 90°F over a matter of minutes. These assumed temperatures, although bounding, are also realistic. Recorded temperatures showed the RFW pipe to be above 400°F in the riser section upstream of the containment isolation valves. Meanwhile, the condensate system had been shutdown for several days and temperatures had approached 90°F. When flow to the reactor was established through RFW-FCV-10, 600 gpm of 90°F water was introduced into the hot piping system and higher than normal thermal forces could have occurred. However, analyses show worst case loads to be on the order of 15-20 kips which is only a fraction of the load required to do the hanger damage. Classical thermal expansion was therefore discounted as a credible failure mechanism.

Thermal Gradients - Top-to-bottom temperature gradients in a horizontal pipe containing two phase flow or in a moisture separator reheater containing superheated steam in the top and water in the bottom of the vessel or around a hot turbine rotor have been identified for some time. The concept of a temperature gradient forming in a pipe containing flowing water was not well understood, nor was it known if it could occur. However, if a temperature gradient between the top and bottom of a long pipe is postulated, the piping would begin to bow. For the conditions experienced, the 90°F condensate is approximately 10 pounds/cubic foot heavier than the 407°F water initially contained in the flow straightening section. If the heavier cold water enters the hot piping and flows along the lower portion of the line, a worst case top-to-bottom temperature gradient of 317°F is produced. Subsequent calculations showed that temperature gradients on the order of 200°F could produce large bowing displacements and loadings that exceed support limits (e.g. hangers RFW-112, 113, 114, and 116) which could result in their failure.

To verify this theory, the motion detectors were rearranged to better observe piping movements. In addition, thermocouples (four total) were installed on the top and bottom of the piping just upstream of RFW-FE-1A and 1B.

Until such a time that the loadings were better defined and the transient better understood, restrictions were placed on plant operation to preclude restart from hot standby.

3) Structural Evaluation

With the damage defined, the structural evaluation was focused on the following immediate tasks:

1) Assess the piping pressure boundary integrity as a result of the event.

2) Determine if thermal gradients could develop reaction loads at RFW-112 sufficient to cause failure.

3) Following the original design basis requirements, determine if RFW-112 could be eliminated or replaced by a snubber such that thermal restraint could be eliminated at this location.

4) Assess the adequacy of the support system to withstand future similar events.

Pressure Boundary Integrity

ADLPIPE Studies

Table 2 summarizes 14 load cases which were used to determine maximum load/stresses in the system as a result of the event (see Case A). These load cases also assess whether section-by-section variation in system temperature, in combination with snubber lockups, could develop excessive support loads in the observed failed rigid strut hangers (RFW-112 and 116, Cases C and 1 thru 12). The Table is self-explanatory; a quick review of the results shows that in each case, loads on supports RFW-112 and 116 remain well within their respective design capabilities. All results shown in Table 2 were obtained by modifying BRI's ADLPIPE models for this system.

Case A was executed to determine the post event maximum loads and stresses. A bounding four inch displacement (3 inches was the measured displacement) was imposed at the location of RFW-112, with both RFW-112 and RFW-116 restraint cards removed from the model in order to simulate these failed supports. The reactive force required to develop the imposed displacement at RFW-112 is 72.2 kips and the corresponding maximum system stress is on the order of 15 ksi; this is well below the allowable stress of 22.5 ksi. In addition, boundary loads on the containment are also acceptably low. It was therefore concluded that pipe overstressing did not occur during the event or as a result of the post accident configuration.

Thermal Gradient Study

ANSYS Model Studies

Since ADLPIPE cannot evaluate thermal bowing gradients an ANSYS finite element model was generated. The model utilizes ANSYS "Stiff-4" beam elements with equivalent piping section properties. The Stiff-4 element will accept cross sectional or top-to-bottom thermal variations. The piping system was nodalized into approximately five foot long sections and truncated at the rigid vertical supports upstream of RFW-116, see Figure 5. The ANSYS model was checked against the ADLPIPE model by repeating load Case A from Table 2. The reaction loads computed by ANSYS compared very favorably with the the ADLPIPE results. Therefore, the ANSYS model approximation was proved sufficiently accurate for the thermal bowing studies.

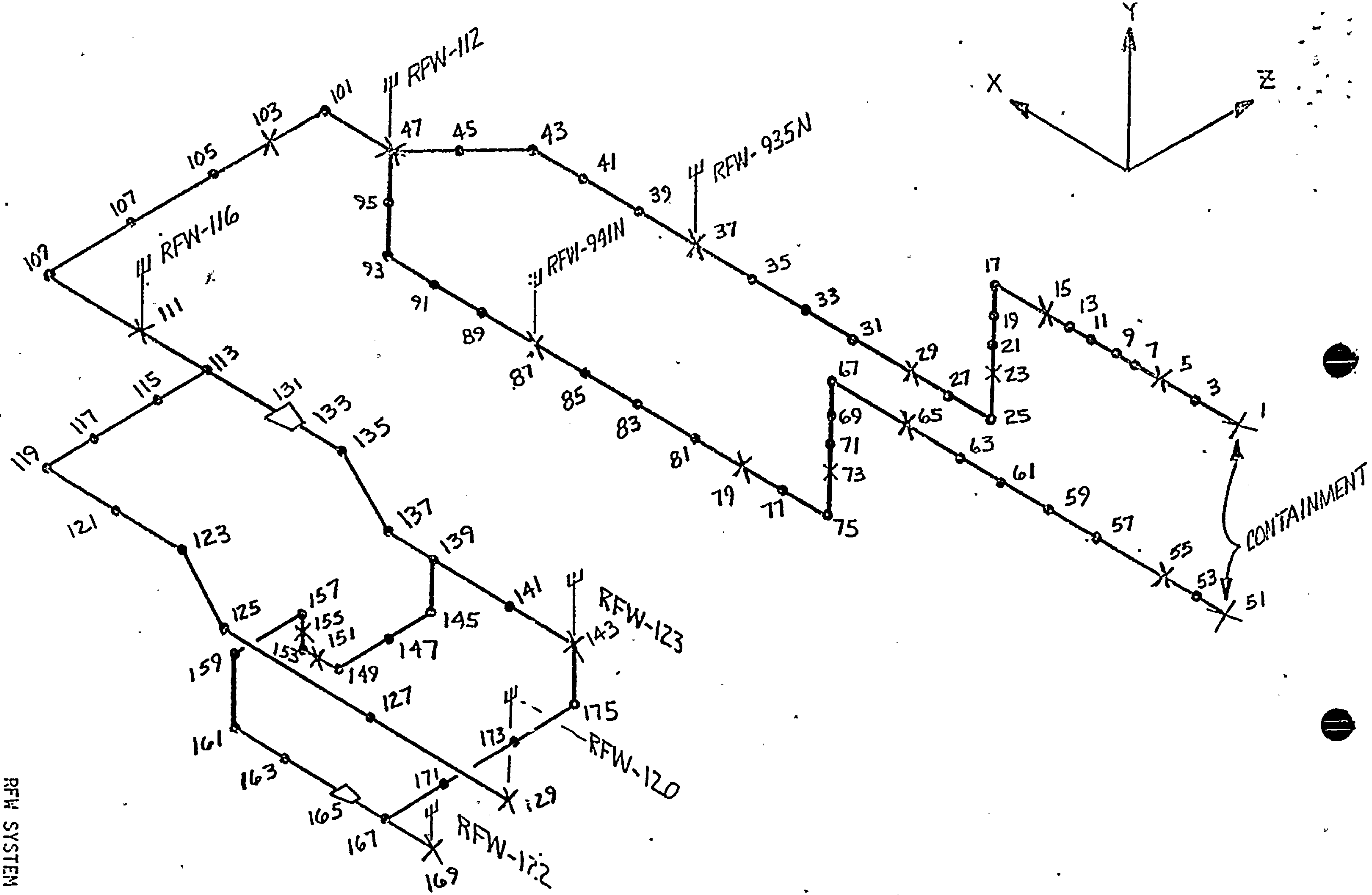


FIGURE 5

RFW SYSTEM ANSYS MODEL

Table 2

ADLPIPE MODEL DEFLECTION AND THERMAL VARIATION STUDIES

Case	Description	RFW-112*	RFW-116*
A	Dy = -4" @ RFW-112 w/RFW-112 & 116 removed	72248	-
B	Thermal Analysis w/all piping @ 420°F	5880	-5660
C	Thermal Analysis w/piping upstream of wye @ 90°F	9325	-7558
1	SE-1,** 420°F from Cont. to TE's SE-2, 408°F from TE's to wye SE-3, 408°F from wye to RFW-114 SE-4, 302°F from RFW-114 to RFW-115 SE-5, 196°F from RFW-115 to TEE SE-6, 90°F from TEE to RFW-V-109 SE-7, 365°F from RFW-V-109 to End	8677	-8281
2	Same as Case #1 with failed snubbers locked up	10400	-10955
3	SE-3, 4, 5 changed to 90°F	8338	-7872
4	Same as Case #3 with locked snubbers	9912	-10470
5	Leg A of SE-2 @ 383°F SE-3, 4, 5 @ 90°F	7950	-7563
6	Same as Case #5 with locked snubbers	9421	-10030
7	Leg A of SE-2 @ 383°F SE-3 @ 390°F SE-4 @ 310°F SE-5 @ 230°F	8244	-7898
8	Same as Case #7 with locked snubbers	9843	-10397
9	Leg A of SE-2 @ 327°F Leg B of SE-2 @ 299°F SE-3, 4, 5 @ 90°F	7199	-6588
10	Same as Case #9 with locked snubbers	8544	-8866
11	Leg A of SE-2 @ 327°F Leg B of SE-2 @ 299°F SE-3 @ 310°F SE-4 @ 240°F SE-5 @ 170°F	7421	-6955
12	Same as Case #11 with locked snubbers	8865	-9178

*Maximum design basis load capability of these hangers is approximately 30 kips.

**"SE-1" defines the piping SEction-1 from containment to the temperature elements (TE). Refer to Figure 1 as an aid in identifying other piping sections defined by this table.

Table 3 summarizes the thermal bowing load cases. These results include no deadweight loads. However, the deadweight reactions can be obtained from the ADLPIPE analyses (e.g. 15 kips at RFW-112). A survey of the results revealed that load Case 5 yields thermal loads of sufficient magnitude to cause buckling of RFW-112. This load case imposed thermal gradients ($\Delta T = 300°F$) over the region from the wye downstream to the elbows at the entrance to the steam tunnel (nodes 47 to 75 or 25, see Figure 5). Load Case 7 revealed that with RFW-112 removed (or failed) and the same worst case thermal loading imposed, sufficient load was then applied to cause failure of RFW-116 (i.e. 72 kips plus 17 kips dead load).

The ANSYS model results also provide an estimate of the pre-event maximum loads/stresses. Since the failure mechanism was buckling induced by quasi-static thermal loading, the pre-event maximum loads are limited by the critical buckling load. In other terms, the load was caused by the restraints, when the restraints fail (buckle), the load was removed. The stresses generated by ANSYS in load case 5 were in the range of 10 to 15 ksi (Note: yield stress approximately ~35 ksi). This result compared favorably with the ADLPIPE result of Table 2 Case A (imposed 4 inch deflection) where reactive loads of 72 kips yield the same order of magnitude stress. Similarly, ANSYS containment boundary loads were well below containment nozzle load limits. These analyses confirm a general design axiom that the piping is usually stronger than the supports. This becomes clear when it is recognized that a moment of 2 million foot-pounds is required to induce pipe yielding in the 24 inch RFW line. At the peak support failure loads, moments of less than half the yield moment were developed.

To ensure that pipe bearing stresses were not exceeded at RFW-112 the Supply System's "LUG" program was executed. A conservative 120 kip local bearing load was applied to the piping pressure boundary using the RFW-112 end bracket as the "foot print" area. Results show that bearing stresses were approximately 75% of the allowable.

A hand calculation was completed to assess the minimum compression strength of RFW-112. The calculation showed that a vertical force of 80 kips will plastically deform the pin on the sway strut sufficiently to allow the paddle to bottom out in the end bracket. This in turn would develop sufficient friction force (i.e. 24 kips) between the paddle and end bracket to cause the threaded rod of the sway strut to fail in bending during the horizontal thermal movement of the pipe. Specifically, the force required to develop a full plastic moment in the threaded rod was 7.6 kips which is much less than normal friction force of 24 kips.

Table 3

RFW - Reactive Support Forces

Load Case	FY (Kips)	
	RFW-112 (Node 47)	RFW-116 (Node 111)
1	-55.46	35.21
2	-56.04	35.86
3	29.28	58.77
4	28.65	59.08
5	89.96	16.60
6	89.27	17.49
7*		72.11
8	57.90	11.13
9	-74.16	
10**	---	---

*Case 5 rerun with RFW-112 failed

**Free Thermal run

Load Case
1. 400/100 temp on 30" pipe up to Node 123+ & 135.
2. 400/100 temp on 30" pipe up to Node 123 & 135 with X- restraint at Node 129, 143 & 169.
3. 400/100 temp on 24" & 30" pipes up to Node 123 & 135.
4. 400/100 temp on 24" & 30" pipes up to Node 123 & 135 with X-restraint at Node 129, 143 & 169.
5. 400/100 temp on 24" pipe up to Y.
6. 400/100 temp on 24" pipe up to Y with X- restraint at Node 129, 143 & 169.
7. 400/100 temp on 24" pipe up to Y without support at Node 47.
8. 400/200 temp on 24" pipe up to Y.
9. 100/100 temp through entire RFW piping with displacement UY = 4.0 at Node 47 & without supports at Node 47 & 111.
10. 400/100 temp on 24" pipe up to Y without supports.

+(Refer to Figure 5 for node numbers)

Requalification of RFW-112 as a Snubber

This task was completed using BRI's final ADLPIPE models for this anchor group. The reanalysis results showed that deadweight and thermal load cases were acceptable, however without RFW-112 in the seismic load cases, the support loads increase such that adjacent existing hangers would exceed design basis allowables. Therefore, RFW-112 could not be eliminated entirely but a snubber would satisfy the analysis requirements. A PSA-10 snubber (15 kip rating) bounded the seismic load and was installed in the system. In the revised configuration, the piping stresses satisfied the allowables of the governing B31.1 code for all load cases. Revised hanger loads and balancing data were generated per the analysis results and issued to the field. Impacts of load changes on existing unmodified hangers were also completed and each affected hanger was found acceptable as designed.

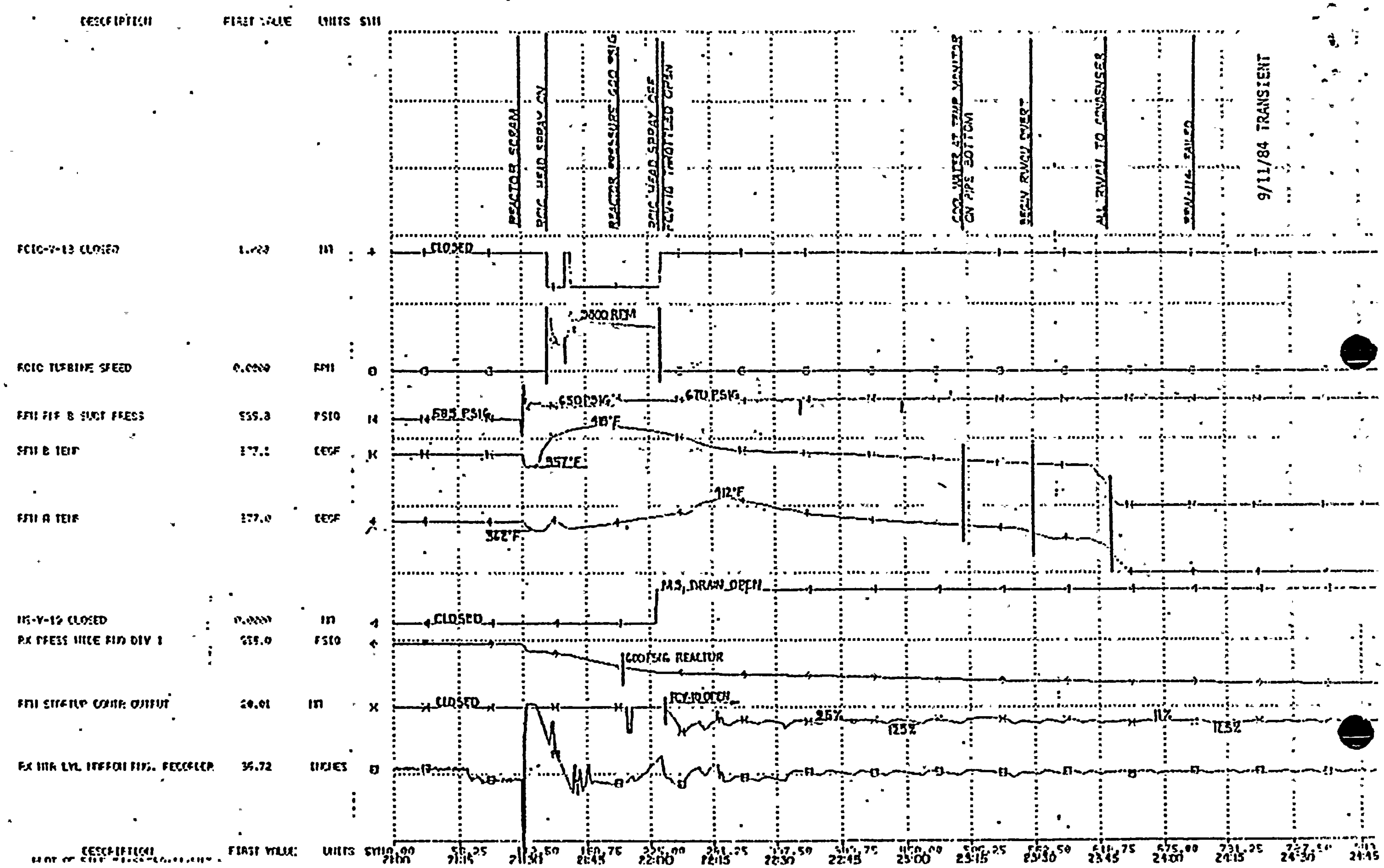
Two areas in the initial evaluation task have not been fully completed. First, analysis of the RFW system's (Class 1) fatigue design basis is incomplete. Preliminary reviews indicate bounding or near bounding thermal transients were assessed by BRI. A second area of investigation needing completion involves a dynamic buckling analysis of RFW-112. Recognizing buckling as an instability condition, it is speculated that the strut static thermal load would result in a dynamic response once strut buckling instability was achieved. This analysis would determine if sufficient inertial loads could be generated to fail the snubbers which were found damaged. In particular, snubbers RFW-87, -88, and -126 which are in the immediate area of RFW-112 and RFW-116, and whose failures may have been associated with vibration caused by RFW-FCV-10, water hammer, or possibly dynamic buckling loads. Snubber RFW-113 was simply bent when RFW-112 failed. The remaining snubber failures have been attributed to known cavitation at RFW-FCV-15 and/or water hammer. The dynamic buckling analyses will be completed later utilizing the ANSYS model which was developed to assess thermal bowing gradients.

C. Event 3, September 11, 1984

1) Event Summary

The following summarizes significant events leading up to and following the failure of feedwater piping support RFW-114. The plant had been running at approximately 75% power and the reactor was shutdown during testing. See Figure 6 for graphic representation of pertinent data.

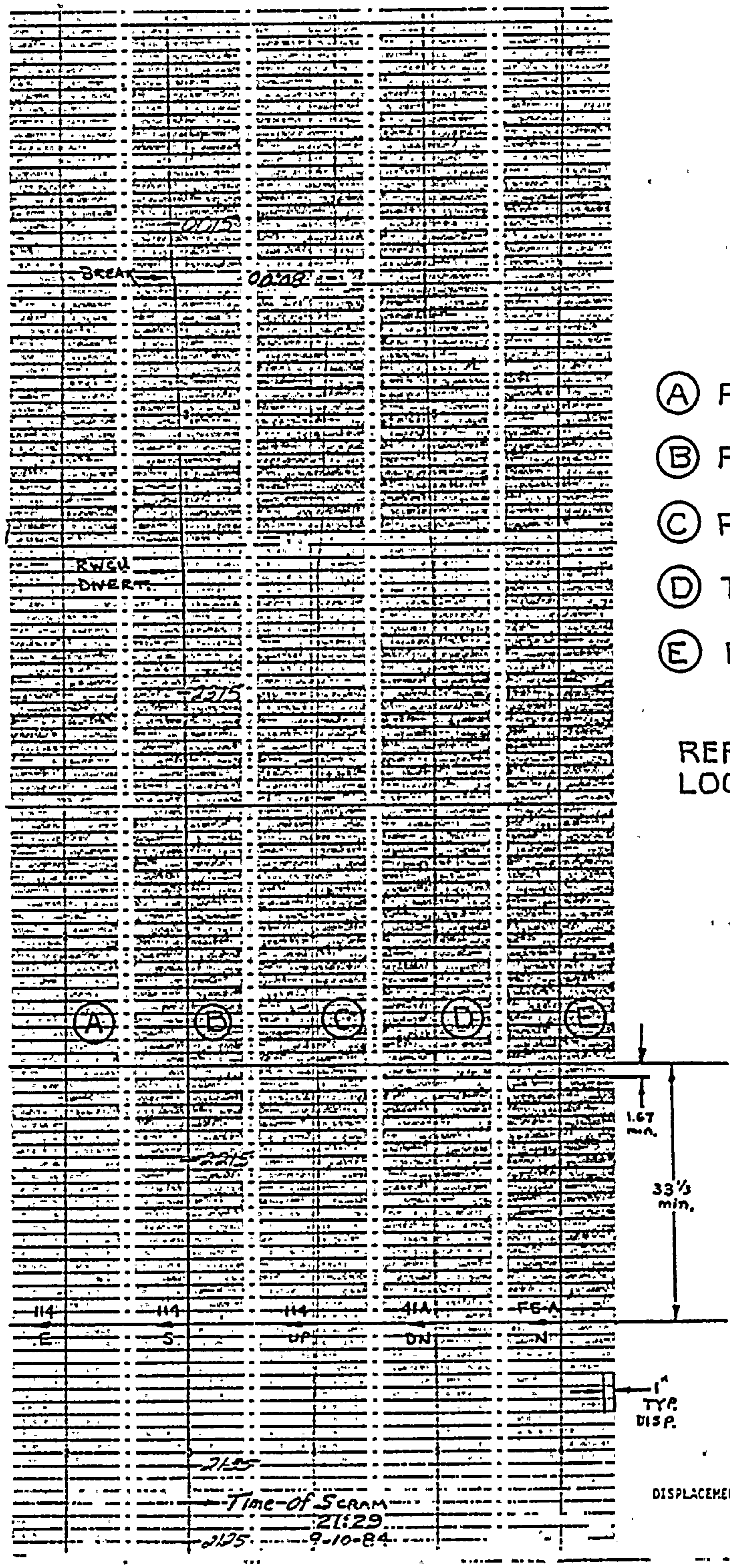
FIGURE 6


9/11/84 TRANSIENT
DESCRIPTION
FIRST VALUE
UNITS
REACTOR SCRAM
CLOSED
3000 RPM
RCIC TURBINE SPEED
585 PSIG
650 PSIG
670 PSIG
410°F
357°F
412°F
362°F
MS. DRAIN OPEN
600 PSIG REACTOR
FCV-10 OPEN
95%
125%
11%
115%
PSIG
DEGF
INCHES
21:15
21:30
21:45
22:00
22:15
22:30
22:45
23:00
23:15
23:30
23:45
24:00
24:15
24:30
24:45

Date	Time	Event
9/10/84	21:28	Reactor Feedwater 377.1°F Reactor Vessel Level 34 inches Reactor Pressure 958.0 psig
	21:29	Reactor Scram (Shutdown)
	21:31	Both reactor feedwater pumps tripped (one manually, one on high reactor water level).
	21:33	Feedwater temperature at RFW-TE-41A and 41B at approximately 360°F.
	21:35	RCIC Head Spray on
	21:50	Feedwater at RFW-TE-41B peaked at 418°F.
	21:53	Vessel pressure at 600 psig.
	22:02	Main Steam drains open.
	22:03	RCIC head spray off.
	22:04	RFW-FCV-10 open and controlling vessel level.
	22:20	Feedwater at RFW-TE-41A peaked at 412°F.
	23:13	Cool feedwater reached RFW piping bottom test thermocouples.
	23:30	RWCU diversion to condenser began.
	23:43	Sharp decrease in RFW temperature at RFW-TE-41A and 41B.
	23:48	RWCU diversion to condenser complete.
	24:08	RFW-114 failed.

2) Event 3 Evaluation

Although it was not predicted to occur on normal shutdown of the reactor, the mechanism for all the previous hanger failures became clearly understood as cold shutdown conditions were achieved. All the pipe monitoring instrumentation was working perfectly. Large top-to-bottom temperture differentials occurred in the flow straightening sections as the reactor was being cooled down. This resulted in large vertical displacements in the area of the previously failed hangers. The peak vertical displacement at hanger RFW-114 was 2.4 inches. The extension limit of the spring-can on RFW-116 was 1.6 inches and the attachment welds broke during the event. Refer to Figure 7 for piping displacements and Figure 8 for top-to-bottom temperature differentials.



Ⓐ RFW-114 EAST MOTION

Ⓑ RFW-114 SOUTH MOTION

Ⓒ RFW-114 VERTICAL MOTION

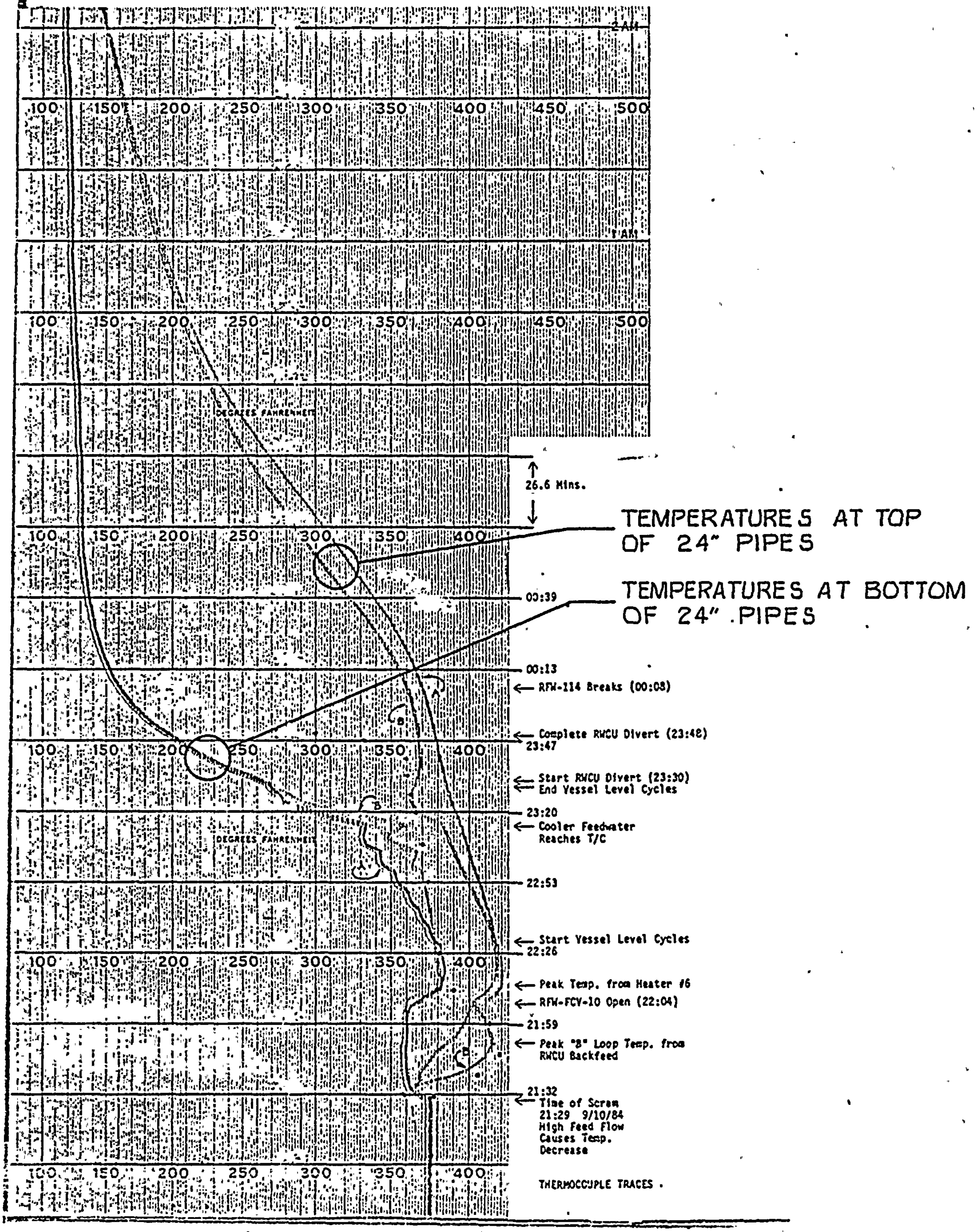
Ⓓ TE-41A VERTICAL MOTION

Ⓔ FE-A SOUTH MOTION

REFER TO FIGURE 1 FOR LOCATIONS AND ORIENTATION

DISPLACEMENT TRACES

FIGURE 8


2 AM
100 150 200 250 300 350 400 450 500
1 AM
100 150 200 250 300 350 400 450 500
DEGREES FAHRENHEIT
26.6 Mins.
TEMPERATURES AT TOP OF 24" PIPES
100 150 200 250 300 350 400
TEMPERATURES AT BOTTOM OF 24" PIPES
00:39
00:13
RFW-114 Breaks (00:08)
Complete RWCU Divert (23:48)
23:47
100 150 200 250 300 350 400
Start RWCU Divert (23:30)
End Vessel Level Cycles
23:20
Cooler Feedwater Reaches T/C
DEGREES FAHRENHEIT
22:53
Start Vessel Level Cycles
22:26
100 150 200 250 300 350 400
Peak Temp. from Heater #6
RFW-FCV-10 Open (22:04)
21:59
Peak "B" Loop Temp. from RWCU Backfeed
21:32
Time of Scram 21:29 9/10/84 High Feed Flow Causes Temp. Decrease
100 150 200 250 300 350 400
THERMOCOUPLE TRACES.

Stratification

As a result of normal reactor shutdown, stratification occurred from two sources. Shortly after the scram both reactor feed pumps tripped dropping the pressure upstream of the RFW containment isolation valves to approximately 600 psig. Because reactor pressure was higher than 600 psig, RWCU flow into the RFW line flowed backward and down to the flow straightening sections. The hot RWCU water stratified at the top of the line. Inspection of Figure 8 shows that shortly after scram, top pipe temperatures rise to about 420°F while bottom temperatures remain below 380°F.

When the reactor pressure dropped below 600 psig, condensate flow was allowed to flow to the reactor through RFW-FCV-10. This flow eventually depleted the residual heat in the feedwater heaters and cool water was sent into the hot flow straightening sections. The cool water stratified at the bottom of the pipe resulting in peak top-to-bottom differential temperatures of 220°F.

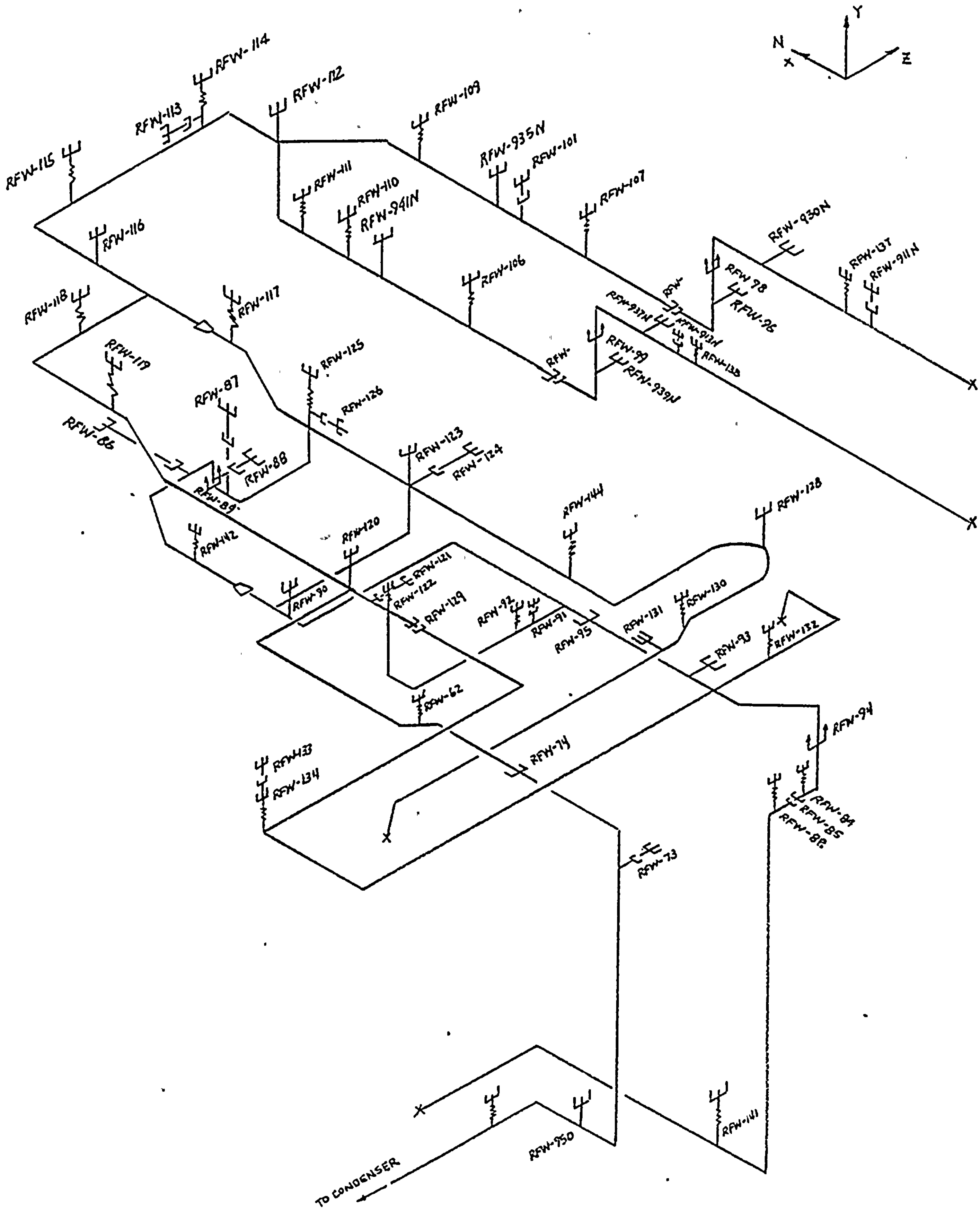
Inspection of Figure 7 shows that displacements at both ends of the flow straighteners dropped. Using hangers RFW 941N and 935N as a pivot point, the north end of the line dropped 2.38 inches while the south end dropped .9 inches. The resultant loadings caused failure of hanger RFW-114.

3) Structural Evaluation

With the field test data in hand the redesign mission became self evident. That is, stratification was proved and the system's supports would have to be modified to accommodate the thermal bowing response. Stemming from the analysis efforts of the August 22 event, the ANSYS model was further refined (see Figure 9) to predict the system's bowing response. A design basis bowing T of 250°F was imposed on the ANSYS model from the wye to the elbows at the riser section upstream of the isolation valves. This thermal gradient resulted in a four inch downward deflection in the area of RFW-114, 115 and 116. Table 4 summarizes the specific hanger modifications required to accommodate the design basis bowing response. To requalify the system per the governing piping code requirements, the entire anchor group model (ADLPIPE) was re-executed with the modifications cited by Table 4 in deadweight, thermal and seismic load cases.

The system modifications were implemented in the field and the system realigned and balanced. As noted in the August 22 Event, fatigue analyses of the ASME Class 1 portion of the system (isolation valves to containment) remains as a final task in closing out the design record file. This effort will be completed over the next few months as operating procedures, system line-up, and thermal analysis refinements are completed.

FIGURE 9



RFW SYSTEM ANSYS MODEL

TABLE 4

HANGER MODIFICATIONS

RFW-112	Changed to a snubber to permit thermal movement yet maintain seismic design basis.
RFW-113	Snubber end bracket rotated to enhance freedom of movement in vertical direction.
RFW-114	Eliminated, load transferred to adjacent springs.
RFW-115	Snubber added to existing long travel variable spring at this location. Snubber addition was required since RFW-116 was changed to spring (i.e. non-seismic support).
RFW-116	Converted from a rigid to a constant force spring to provide needed displacement capability and deadweight capacity.
RFW-131	Thermal expansion studies showed load increase beyond design basis. Support modified by utilization of larger strut.

ACKNOWLEDGEMENT

This report is based on contribution by many engineers who enabled the WNP-2 Plant to maintain power ascension activities while concurrently successfully diagnosing, evaluating, repairing and developing on-going mitigation and monitoring programs to resolve this unique piping system thermal performance problem. The creativity and perseverance of these engineers was necessary and their contributions to the content of this report and the technical solutions is acknowledged.

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