January 27, 1986

Docket No. 50-220

DISTRIBUTION
Docket
NRC PDR
Local PDR
PD#1 RDG
JZwolinski
NMP1 File
OELD
EJordan
BGrimes
JKelly
RHermann
CJamerson

LICENSEE: Niagara Mohawk Power Corporation

FACILITY: Nine Mile Point Nuclear Station, Unit No. 1

SUBJECT: SUMMARY OF JANUARY 23, 1986 MEETING WITH NIAGARA MOHAWK POWER COOPERATION

A meeting was held on January 23, 1986, in Bethesda, Maryland between representatives of Niagara Mohawk Power Corporation (NMPC), the licensee, it's contractor, MPR Associates, and the NRC staff to discuss potential design deficiencies in the process piping penetrations at Nine Mile Point Nuclear Station, Unit No. 1 (NMP-1).

The issues related to the existing pipe penetrations not being in complete compliance with the design criteria in the FSAR were discussed at length. A list of the attendees is attached as Enclosure 1. A copy of the slides used by the licensee as a part of its presentation is also attached as Enclosure 2. Enclosure 3 was presented to the staff by the licensee's contractor, MPR Associates, regarding emergency condenser steam line breaks at the containment penetrations at NMP-1. Enclosure 4 is a copy of site drawings, specifically, penetration schematics along with process and instrumentation drawings. Enclosure 4 is available in the Public Document Room. Enclosure 5 is a copy of the Inservice Inspection isometrics indicating ultrasonic and liquid penetrant inspections performed during the last two outages at NMP-1.

The licensee provided a discussion of the circumstances leading to a plant shutdown on January 18, 1986. The licensee has performed fracture mechanics analyses for the systems in question and will provide a formal request for a change to the facility operating license in order to operate until the planned end of the current operating cycle. The licensee has committed to provide additional information regarding calculated leak rates for the systems in question, and to issue a Standing Order establishing interim limits on changes in the drywell unidentified leakage to less than 1 gpm in 24 hours.

The licensee in providing a formal request for a change to the facility operating license requested that the staff review their submittal in a timely manner. The staff agreed to a prompt review of the request.

Original signed by:
Janet L. Kelly, Project Manager
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
As stated

DBL:PD#1 JK	DBL:PD#1	DBL:PD#1
JKelly:jg	RHermann	JZwolinski
1/27/86	1/27/86	1/27/86

8602040483 860127
PDR ADOCK 05000220
S PDR

Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station,
Unit No. 1

cc:
Troy B. Conner, Jr., Esquire
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Frank R. Church, Supervisor
Town of Scriba
R. D. #2
Oswego, New York 13126

Niagara Mohawk Power Corporation
ATTN: Mr. Thomas Perkins
Plant Superintendent
Nine Mile Point Nuclear Station
Post Office Box 32
Lycoming, New York 13093

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 126
Lycoming, New York 13093

John W. Keib, Esquire
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Jay Dunkleberger
Division of Policy Analysis
and Planning
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Niagara Mohawk Power Corporation
Attn: Mr. C. V. Mangan
Senior Vice President
c/o Miss Catherine R. Seibert
300 Erie Boulevard West
Syracuse, New York 13202

ENCLOSURE 1

LIST OF ATTENDEES

Raj Auluck	NRC/NRR/BWD#1
Wayne Hodges	NRC/RSB
Horace Shaw	NRC/NRR/BWR/EB
Bob La Grange	NRC/NRR/BWR/EB
John R. Fair	NRC/IE
Raymond J. Pasternak	NMPC/Sr Nuclear Engineer
Bob Coward	MPR Assoc.
William R. Schmidt	MPR Assoc.
Tom Roman	NMPC - Station Supt.
Hans D. Giesecke	MPR Assoc.
Rob Oleck	NMPC/Lead Strucutral Engineer
Lee A. Klosowski	NMPC/Lead Mechanical Engineer
Larry Phillips	NRC/RSB
Joseph R. Levine	NRC/RSB
Robert Hermann	NRC/NRR/BWD#1
John A. Zwolinski	NRC/NRR/BWD#1
Janet L. Kelly	NRC/NRR/BWD#1
John Rogers	Observer - GPU Nuclear
Michael Laggart	Observer - GPU Nuclear
Jack N. Donohew, Jr.	NRC/NRR/BWD#1
Michael G. Mosier	NMPC/Assoc. Sr. Nuclear Engineer
Stan Wilczek, Jr.	NMPC/Manager Nuclear Tech

PRESENTATION TO NRC
BY NIAGARA MOHAWK
ON
NINE MILE POINT UNIT 1
GUARD PIPE CONFIGURATION

JANUARY 23, 1986

AGENDA

I BACKGROUND

II PENETRATIONS AFFECTED - NMPC

III ORIGINAL DECISION CRITERIA - MMPC

IV CURRENT ANALYSIS AND RESULTS - NMPC, MPR

V MITIGATING FACTORS - NMPC, MPR

VI CONCLUSION

VII REQUEST FOR LICENSE AMENDMENT

ATTENDEES

NIAGARA MOHAWK

S. W. WILCZEK, JR.	MANAGER NUCLEAR TECHNOLOGY
T. W. ROMAN	NMP1 STATION SUPERINTENDENT
L. A. KLOSOWSKI	LEAD MECHANICAL ENGINEER
R. J. PASTERNAK	LEAD LICENSING ENGINEER
M. G. MOSIER	LICENSING ENGINEER
R. F. OLECK, JR.	LEAD STRUCTURAL ENGINEER

MPR ASSOCIATES

W. SCHMIDT

H. GIESECKE

R. COWARD

I BACKGROUND

A. EMERGENCY CONDENSER PIPE REPLACEMENT

B. ORIGINAL SUPPORT DESIGN CRITERIA

C. EXISTING CONFIGURATION

D. INITIATED PART 21 EVALUATION ON THURSDAY 1/16/86

E. INITIATED PLANT SHUTDOWN ON SATURDAY 1/18/86

F. NEXT REFUELING & MAINTENANCE OUTAGE SCHEDULED FOR 3/8/86

B. ORIGINAL SUPPORT DESIGN CRITERIA

NINE MILE POINT UNIT 1
SECOND SUPPLEMENT
TO THE
FINAL SAFETY ANALYSIS REPORT

QUESTION

DESCRIBE THE EFFECT OF JET AND REACTION LOADS ON THE INTEGRITY OF CONTAINMENT PENETRATIONS.

ANSWER

THE CONTAINMENT PENETRATIONS WERE ANALYZED FOR THE EFFECT OF A COMPLETE LINE BREAK INSIDE THE GUARD PIPES AS WELL AS AT LOCATIONS IN THE DRYWELL VESSEL UP TO THE CONNECTION AT THE REACTOR VESSEL.

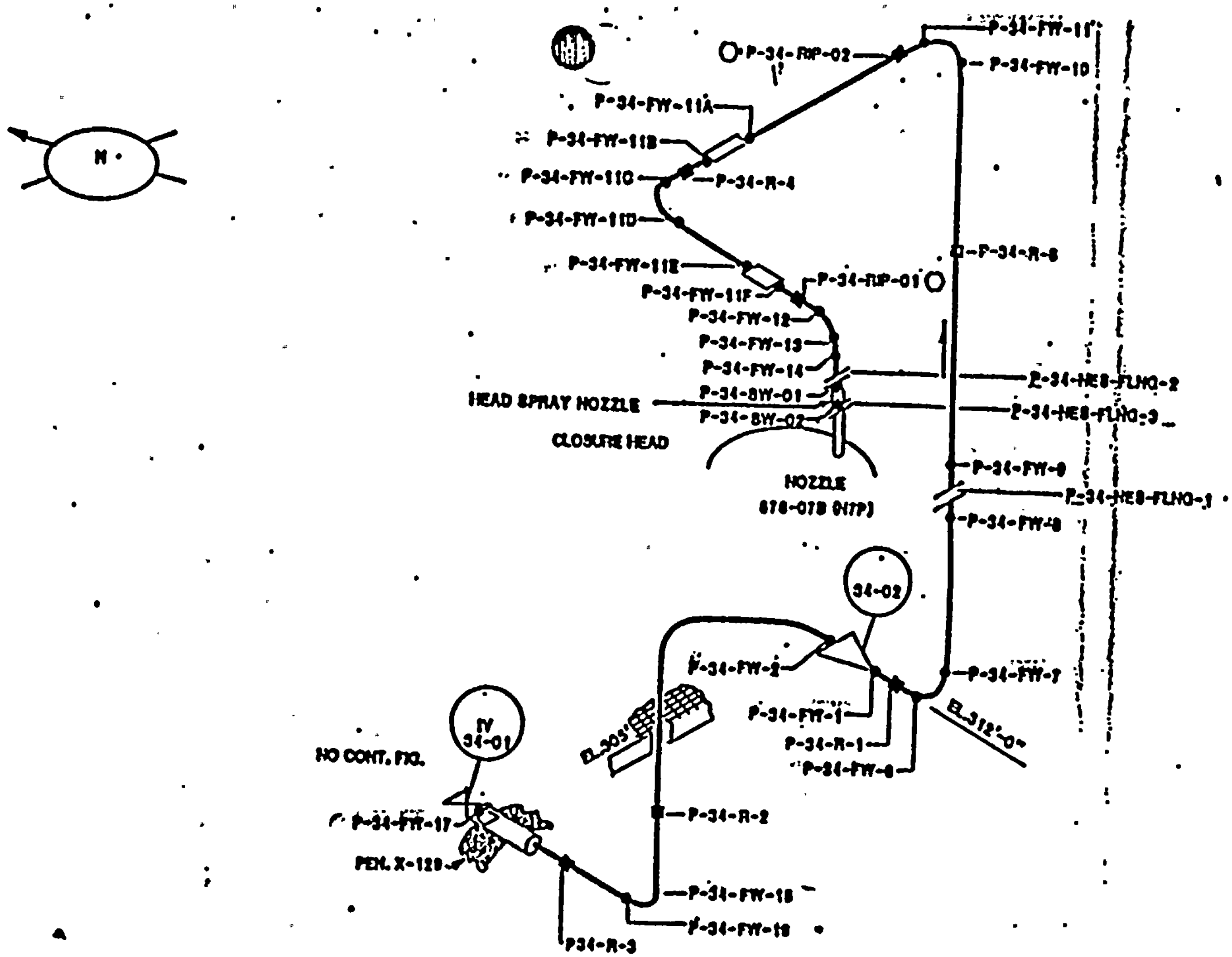
ALL CALCULATED AXIAL JET LOADS INSIDE THE GUARD PIPE WITH THE MAXIMUM POSSIBLE SEPARATION OF THE BROKEN PIPE CENTER LINES OR WITH THE GUARD PIPE FULLY PRESSURIZED BY PROCESS PIPE BREAKS ARE WELL BELOW THE DESIGN STRENGTH OF THE ANCHORS AT OR NEAR THE EXTERNAL ISOLATION VALVE.

PARTIAL PROCESS PIPE BREAKS SUCH AS A SPLIT INSIDE THE GUARD PIPE OR COMPLETE LINE BREAKS INSIDE THE DYRWELL WILL PRODUCE BENDING STRESSES IN THE PROCESS PIPE CONNECTION TO THE ISOLATION VALVE, BUT STRESSES BELOW THE CODE ALLOWABLES. THE SECTION MODULUS OF THE GUARD PIPE AND ISOLATION VALVES IS MUCH LARGER THAN THAT OF THE PROCESS PIPE; THEREFORE, THEIR UNIT STRESSES ARE MUCH LESS THAN THOSE IN THE PROCESS PIPE. UNDER ALL CONCEIVABLE CIRCUMSTANCES THE GUARD PIPES WILL REMAIN INTACT AND PREVENT JET IMPINGEMENT ON THE BELLOWS AND DRYWELL NOZZLES.

II POTENTIAL PENETRATIONS AFFECTED

A. TYPICAL GUARD PIPE CONFIGURATION

B. 15 POTENTIAL PENETRATIONS AFFECTED



REF. DWGS.

N.M.P.C.-C-26851-C
KELLOGG-37-1 REV.2
37-2 REV.1

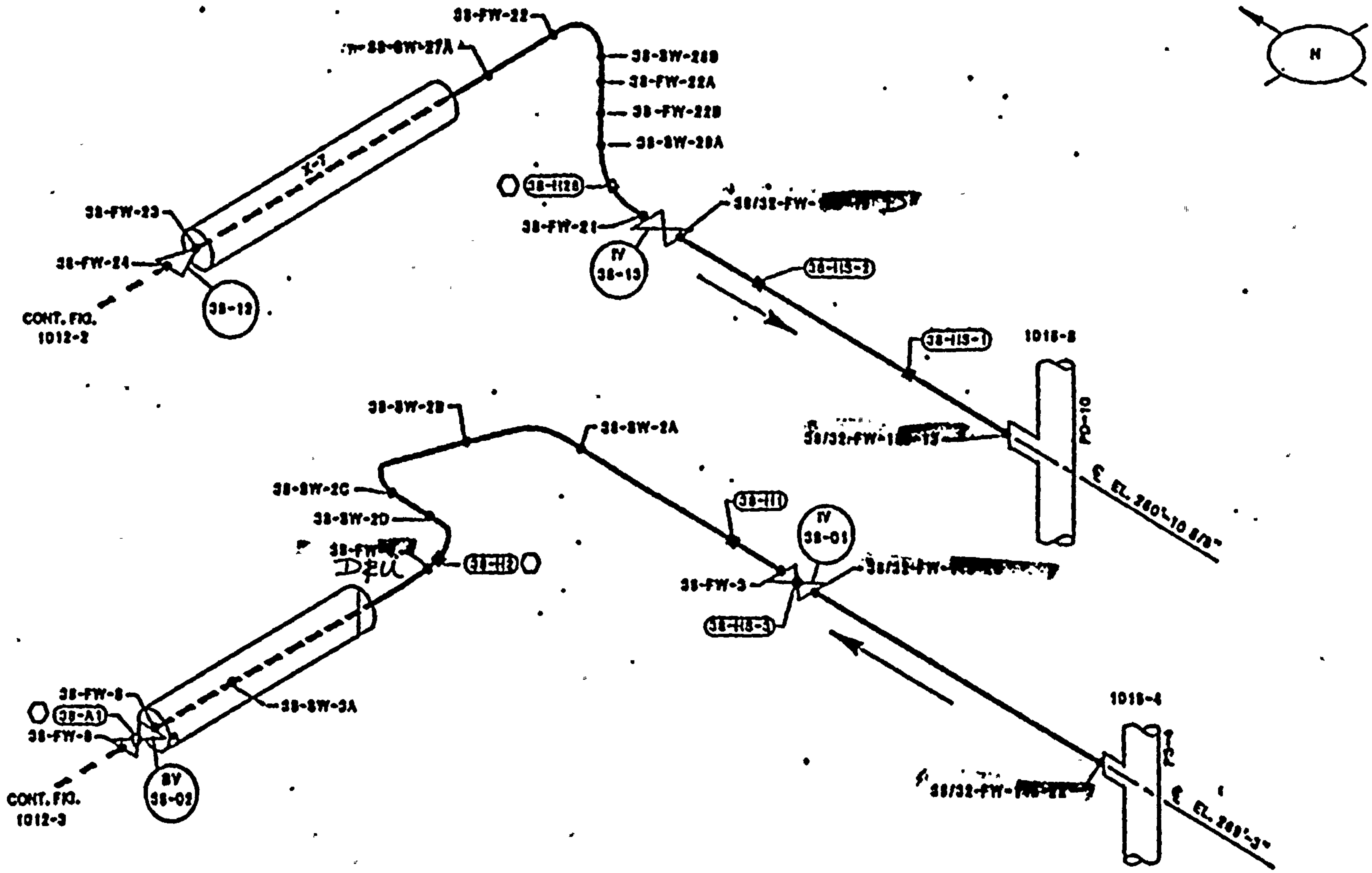
NOTES

● -PIPING WELD

◘ -HANGER

○ -INTEGRALLY WELDED ATTACHMEN

nes	
SYSTEM NO. 34	TITLE HEAD SPRAY WELD & HANGER MAP
REV. 1	FIG. NO. 1011-2
DRWN.	DATE
APP.	DATE 3/13/81
PROJECT 8530-NMP1	

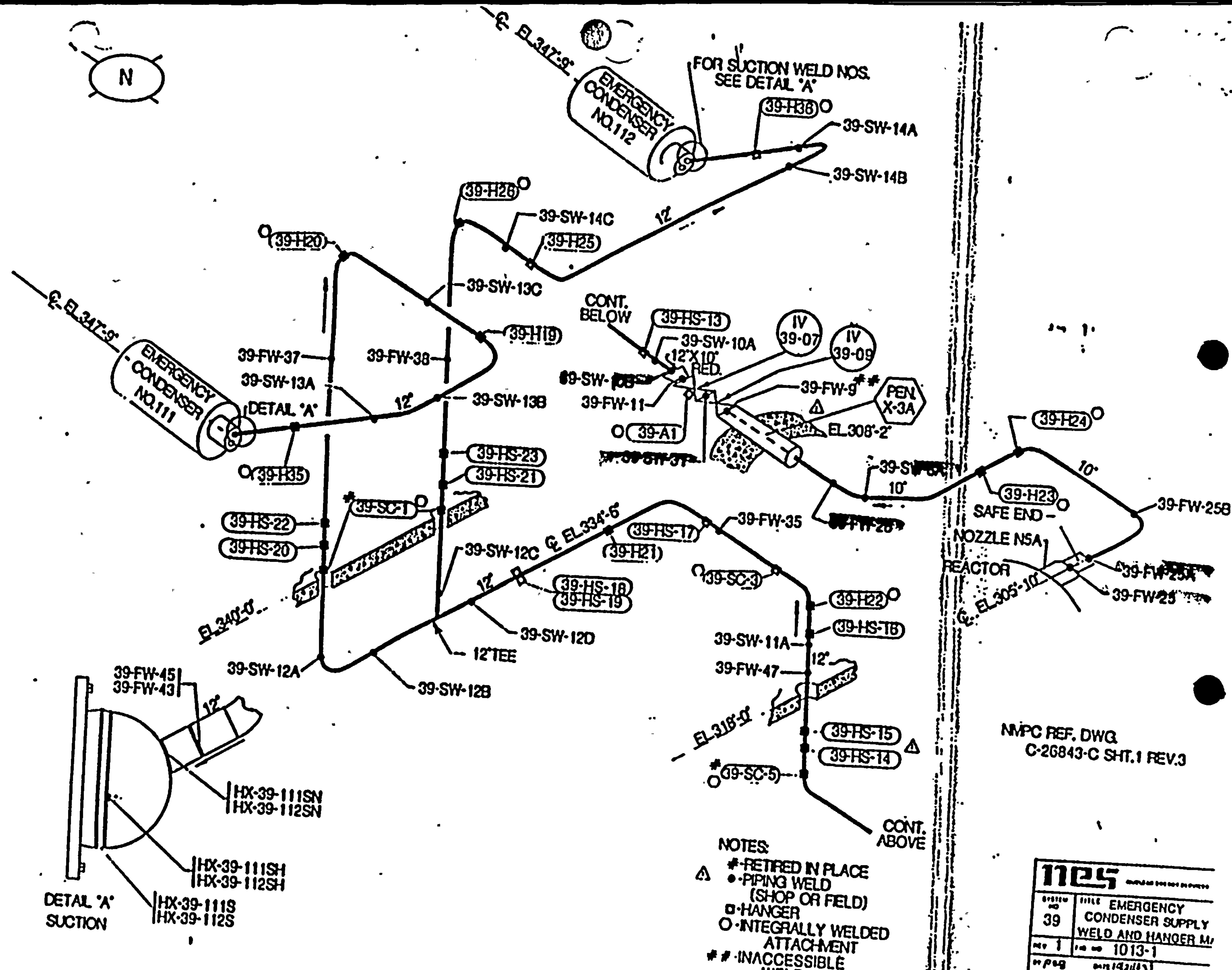


NOTE:

- ● - BUTT WELD (SHOP OR FIELD)
- ◘ - HANGER
- ⬡ - INTEGRALLY WELDED ATTACHMENT

REF. DWG
C-28847-C SHT 1 REV. 2
C-28847-C SHT 2 REV. 1

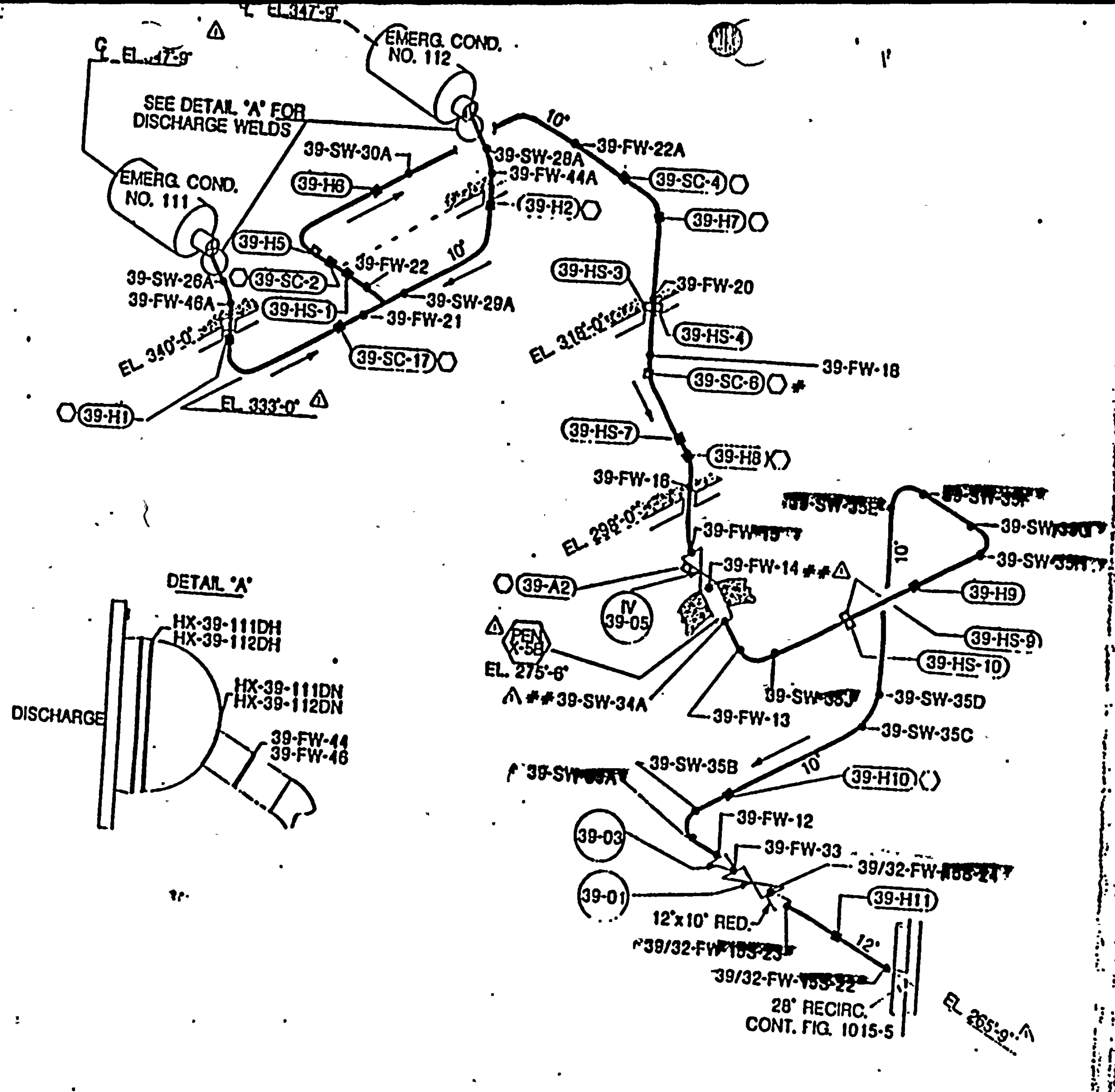
nes	
SYSTEM NO.	TITLE SHUTDOWN COOLING WELD AND SUPPORT MAP
REV. 0	FIG NO. 1012-1
BY JEG	DATE 11-83
APP. MKC	DATE 12-11-83
PROJECT	8530 UNIT 1
NOT TO SCALE	



NOTES:

△ *-RETIRED IN PLACE
● -PIPING WELD (SHOP OR FIELD)
□ -HANGER
O -INTEGRALLY WELDED ATTACHMENT
\# # -INACCESSIBLE WELD

nes	
SYSTEM NO. 39	TITLE EMERGENCY CONDENSER SUPPLY WELD AND HANGER M/
REV 1	1013-1
PROJECT	5530-NMP.1



NOTES:

- ● BUTT WELD (SHOP OR FIEL
- ◘ HANGER
- \# RETIRED IN PLACE
- ⬡ INTEGRALLY WELDED ATTACHMENT
- \## INACCESSIBLE

REF. DWG.

N.M.P. C-26843-C SH. 4 REV. 3

nes

EMERG. COND. RETU

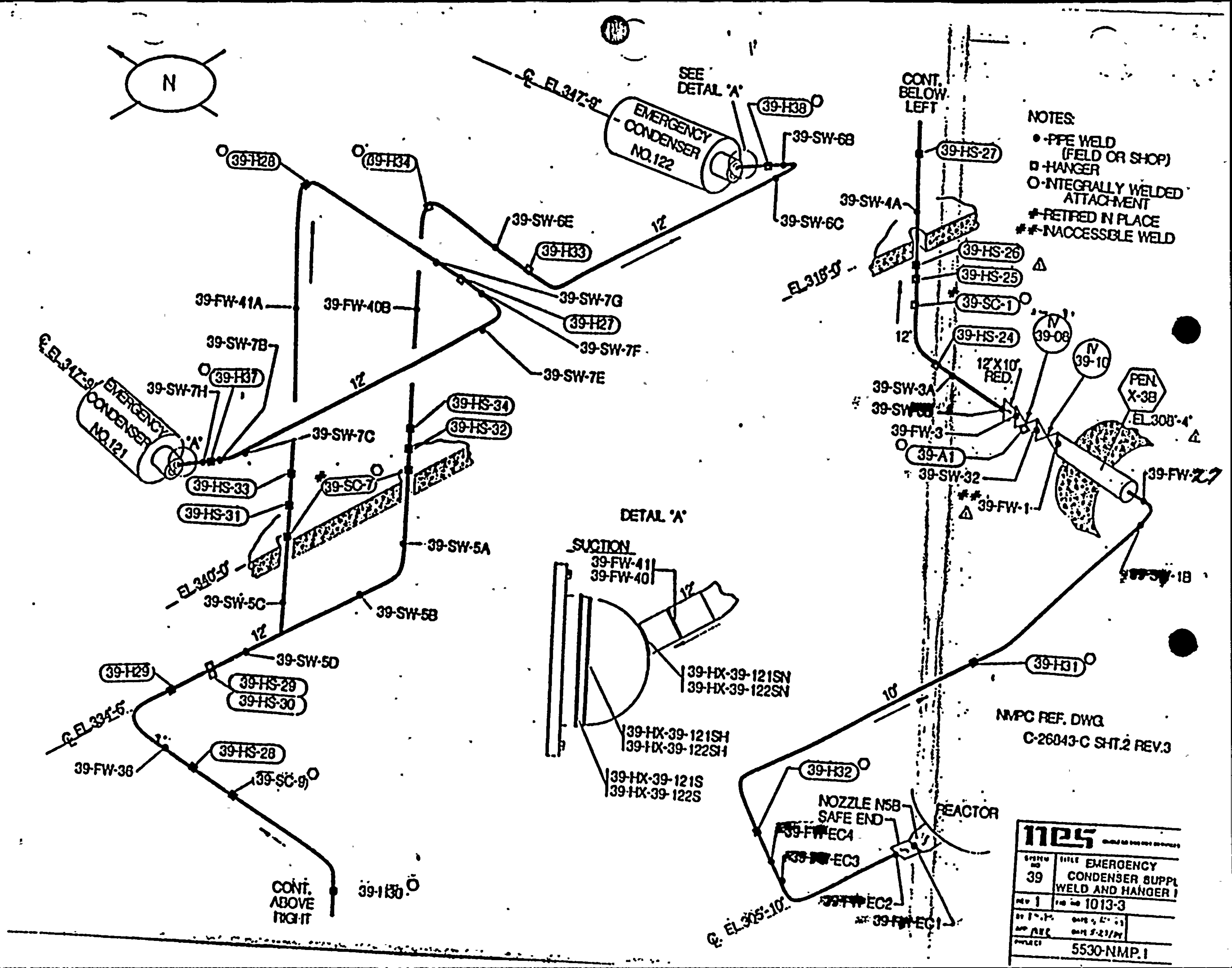
WELD & HANGER M/

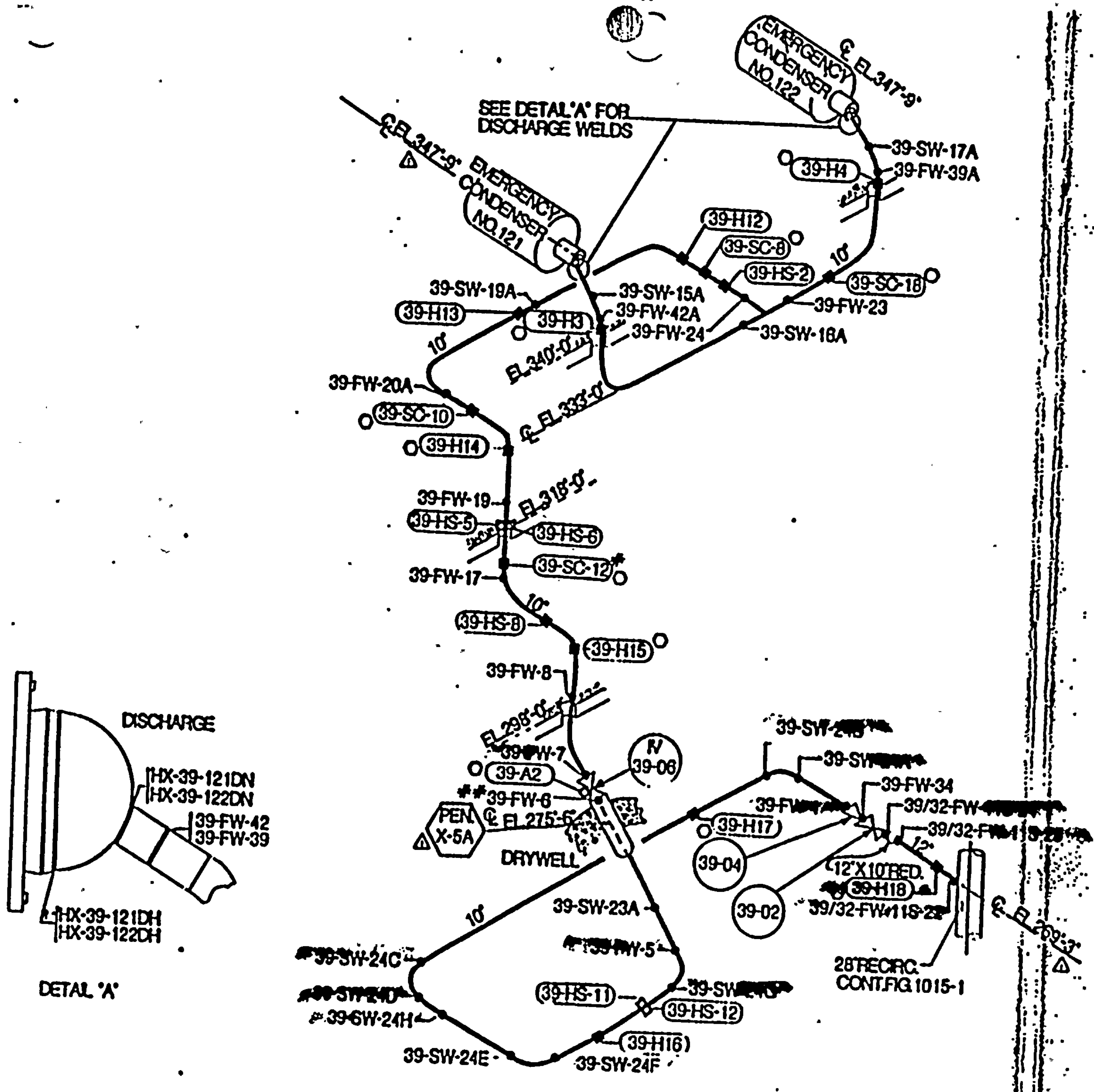
39

1

1013-2

5530 NMP 1


N
SEE DETAIL 'A'
EMERGENCY CONDENSER NO.122
℄ EL 347'-9"
39-H38
39-SW-6B
39-SW-6C
39-H28
39-H34
39-SW-6E
39-H33
12"
39-SW-7G
39-H27
39-SW-7F
39-SW-7E
39-FW-41A
39-FW-40B
39-SW-7B
℄ EL 347'-9"
EMERGENCY CONDENSER NO.121
39-SW-7H
39-H37
'A'
39-SW-7C
39-HS-34
39-HS-32
39-HS-33
39-SC-7
39-HS-31
EL 340'-0"
39-SW-5A
39-SW-5C
39-SW-5B
39-SW-5D
39-H29
39-HS-29
39-HS-30
℄ EL 334'-5"
39-FW-36
39-HS-28
39-SC-9
CONT. ABOVE RIGHT
39-H30
CONT. BELOW LEFT
NOTES:
● -PIPE WELD (FIELD OR SHOP)
□ -HANGER
O -INTEGRALLY WELDED ATTACHMENT
#-RETIRED IN PLACE
##-INACCESSIBLE WELD
39-HS-27
39-SW-4A
EL 318'-0"
39-HS-26
39-HS-25
39-SC-1
39-HS-24
N 39-08
N 39-10
PEN. X-38
EL 308'-4"
12"X10" RED.
39-SW-3A
39-FW-3
39-A1
39-SW-32
39-FW-1
39-FW-2
39-SW-1B
DETAIL 'A'
SUCTION
39-FW-41
39-FW-40
39-HX-39-121SN
39-HX-39-122SN
39-HX-39-121SH
39-HX-39-122SH
39-HX-39-121S
39-HX-39-122S
10"
39-H31
NMPC REF. DWG.
C-26043-C SHT.2 REV.3
39-H32
NOZZLE N5B
SAFE END
REACTOR
39-FW-EC4
39-FW-EC3
39-FW-EC2
39-FW-EC1
℄ EL 305'-10"
nes
EMERGENCY CONDENSER SUPPL
WELD AND HANGER
39
1
1013-3
5530-NMP.1



N.M.P.C. REF.DWG. C-26843-C SHT.3 REV.3

NOTES:
- ● - PIPE WELD (SHOP OR FIELD)
- ▫ - HANGER
- O - INTEGRALLY WELDED ATTACHMENT
- # - RETIRED IN PLACE
- ## - INACCESSIBLE WELD

nes

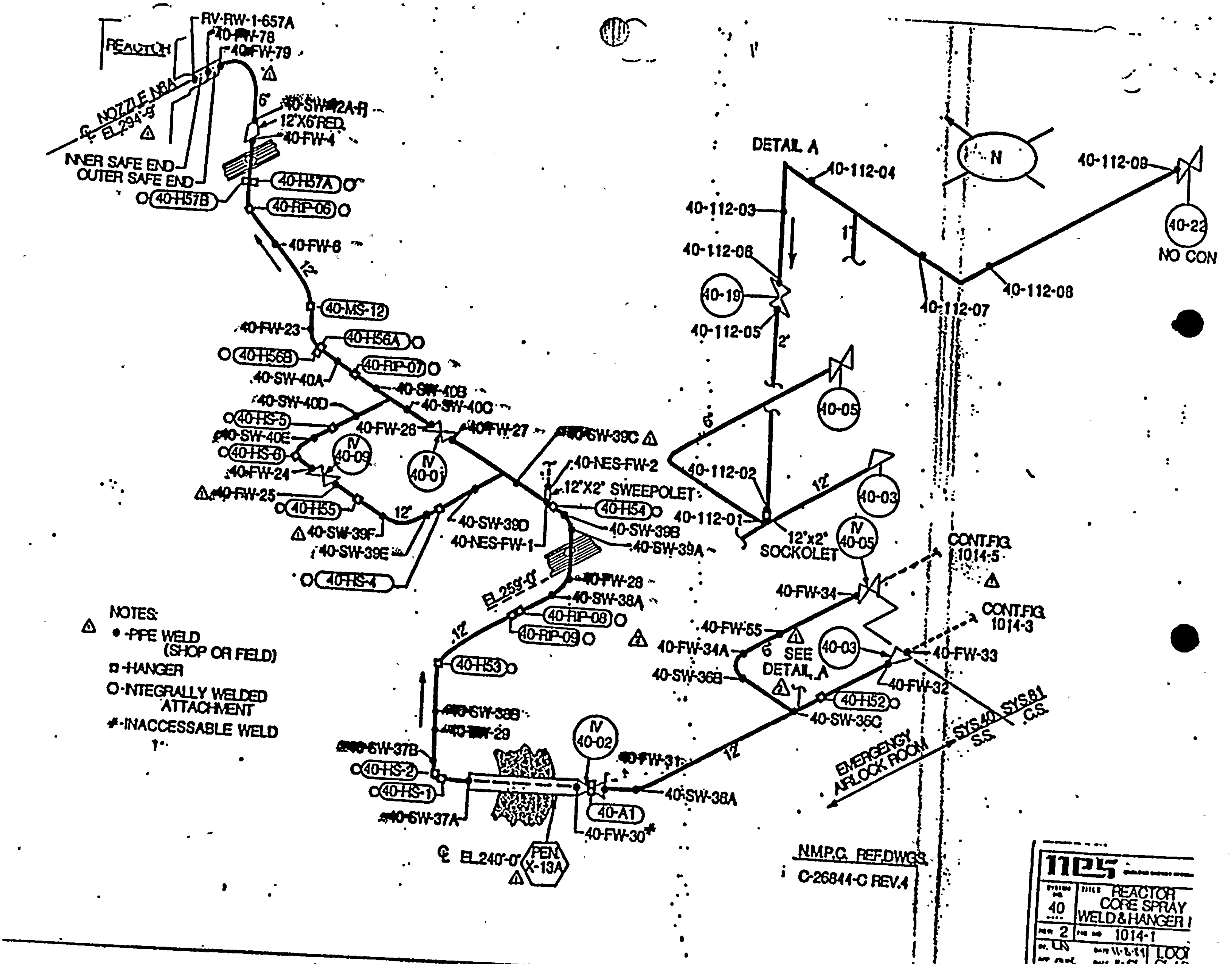
EMERGENCY CONDENSER RETURI WELD & HANGER MAI

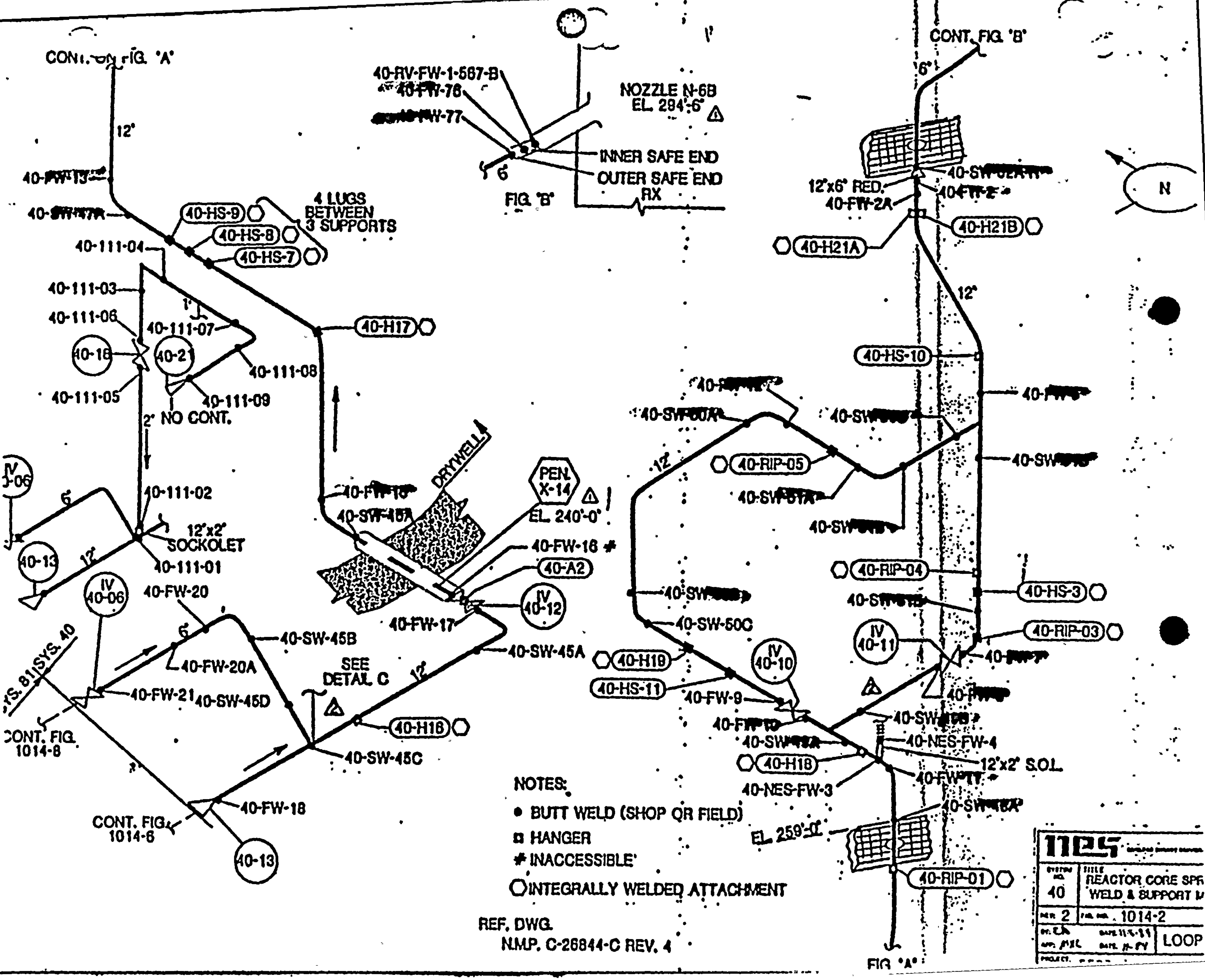
39

1

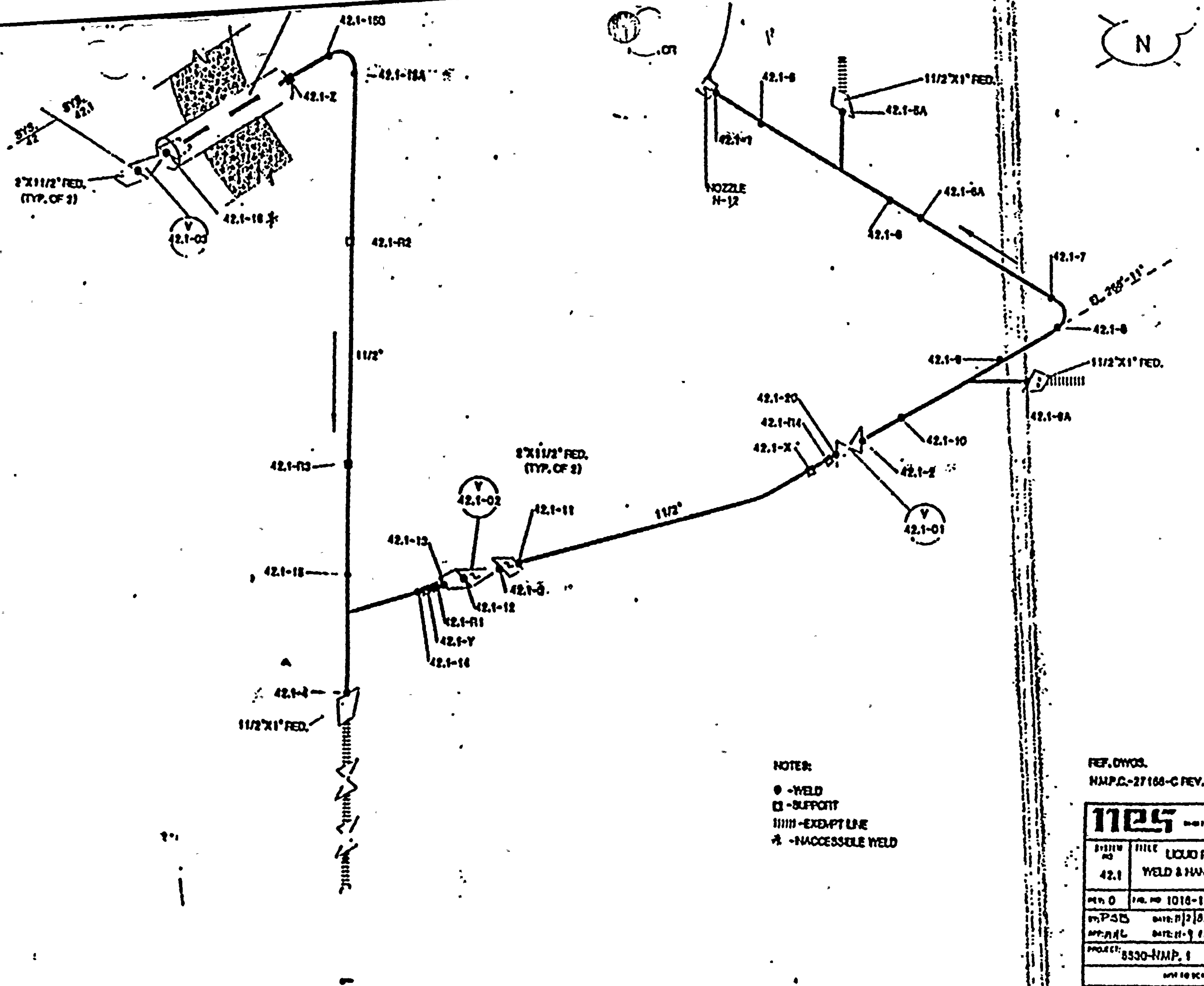
1013-4

5530-NMP1


REACTOR
RV-RW-1-657A
40-FW-78
40-FW-79
NOZZLE N5A
EL 294'-9"
INNER SAFE END
OUTER SAFE END
6"
40-SW-12A-B
12"X6" RED.
40-FW-4
40-H57A
40-H57B
40-RP-06
40-FW-6
12"
40-MS-12
40-FW-23
40-H56A
40-H56B
40-RP-07
40-SW-40A
40-SW-40B
40-SW-40C
40-SW-40D
40-HS-5
40-SW-40E
40-HS-6
40-FW-24
40-FW-25
40-FW-26
40-FW-27
IV 40-09
IV 40-01
40-H55
40-SW-39F
40-SW-39E
40-HS-4
40-SW-39D
40-NES-FW-1
40-SW-39C
40-NES-FW-2
12"X2" SWEEPOLET
40-H54
40-SW-39B
40-SW-39A
EL 259'-0"
40-FW-28
40-SW-38A
40-RP-08
40-RP-09
40-H53
40-SW-38B
40-FW-29
40-SW-37B
40-HS-2
40-HS-1
40-SW-37A
IV 40-02
40-FW-31
40-SW-38A
40-A1
40-FW-30
EL 240'-0"
PEN. X-13A
NOTES:
● -PIPE WELD (SHOP OR FIELD)
□ -HANGER
○ -INTEGRALLY WELDED ATTACHMENT
-INACCESSABLE WELD
DETAIL A
40-112-04
40-112-03
40-112-06
40-19
40-112-05
1"
2"
40-112-07
40-112-08
40-112-09
40-22
NO CON
N
40-05
6"
40-112-02
40-112-01
12"x2" SOCKOLET
40-03
IV 40-05
CONT.FIG. 1014-5
40-FW-34
CONT.FIG. 1014-3
40-FW-55
40-FW-34A
6 SEE DETAIL A
40-03
40-FW-33
40-SW-36B
40-FW-32
40-H52
40-SW-36C
EMERGENCY AIRLOCK ROOM
SYS.40 S.S.
SYS.81 C.S.
NMP.C. REF.DWGS.
C-26844-C REV.4
REACTOR CORE SPRAY WELD & HANGER
40
2
1014-1



CONT. ON FIG. 'A'
12"
40-[redacted]
40-SW-[redacted]
40-111-04
40-HS-9
40-HS-8
40-HS-7
4 LUGS BETWEEN 3 SUPPORTS
40-111-03
40-111-06
40-111-07
1"
40-18
40-21
40-111-08
40-111-05
40-111-09
2" NO CONT.
40-H17
40-RV-FW-1-567-B
40-FW-76
[redacted]-FW-77
6
FIG. 'B'
NOZZLE N-6B
EL. 284'-6"
INNER SAFE END
OUTER SAFE END
RX
IV 40-06
6"
40-111-02
12"x2" SOCKOLET
40-111-01
40-13
12"
40-[redacted]
40-SW-45A
DRYWELL
PEN. X-14
EL. 240'-0"
40-FW-16 #
40-A2
IV 40-12
40-FW-17
40-FW-20
6"
40-SW-45B
40-FW-20A
SEE DETAIL C
12"
40-SW-45A
SYS. 81
SYS. 40
40-FW-21
40-SW-45D
CONT. FIG. 1014-8
40-H16
40-SW-45C
40-FW-18
CONT. FIG. 1014-6
40-13
NOTES:
● BUTT WELD (SHOP OR FIELD)
▫ HANGER
INACCESSIBLE
◯ INTEGRALLY WELDED ATTACHMENT
REF. DWG.
N.M.P. C-26844-C REV. 4
CONT. FIG. 'B'
6"
40-SW-[redacted]
12"x6" RED.
40-[redacted]
40-FW-2A
40-H21B
40-H21A
N
12"
40-HS-10
40-[redacted]
40-SW-[redacted]
40-SW-[redacted]
12"
40-RIP-05
40-SW-[redacted]
40-SW-[redacted]
40-SW-[redacted]
40-RIP-04
40-SW-[redacted]
40-HS-3
40-SW-[redacted]
40-SW-50C
IV 40-11
40-RIP-03
40-[redacted]
40-H19
IV 40-10
40-HS-11
40-FW-9
40-[redacted]
40-[redacted]
40-SW-[redacted]
40-NES-FW-4
40-SW-[redacted]
40-H18
12"x2" S.O.L.
40-[redacted]
40-NES-FW-3
40-SW-[redacted]
EL. 259'-0"
40-RIP-01
FIG. 'A'
NES
SYSTEM NO. 40
TITLE REACTOR CORE SPR
WELD & SUPPORT M
REV. 2
DWG. NO. 1014-2
LOOP

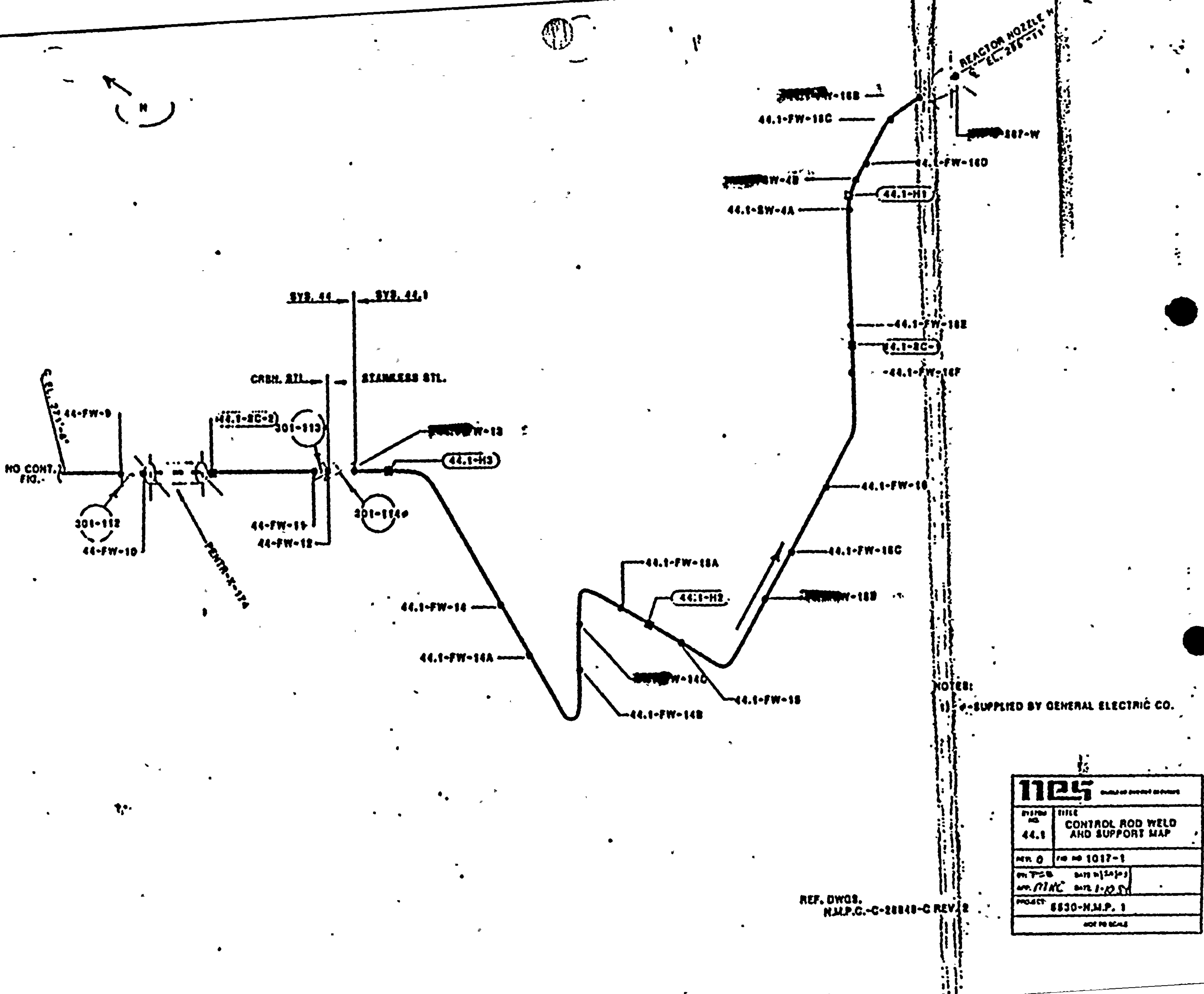


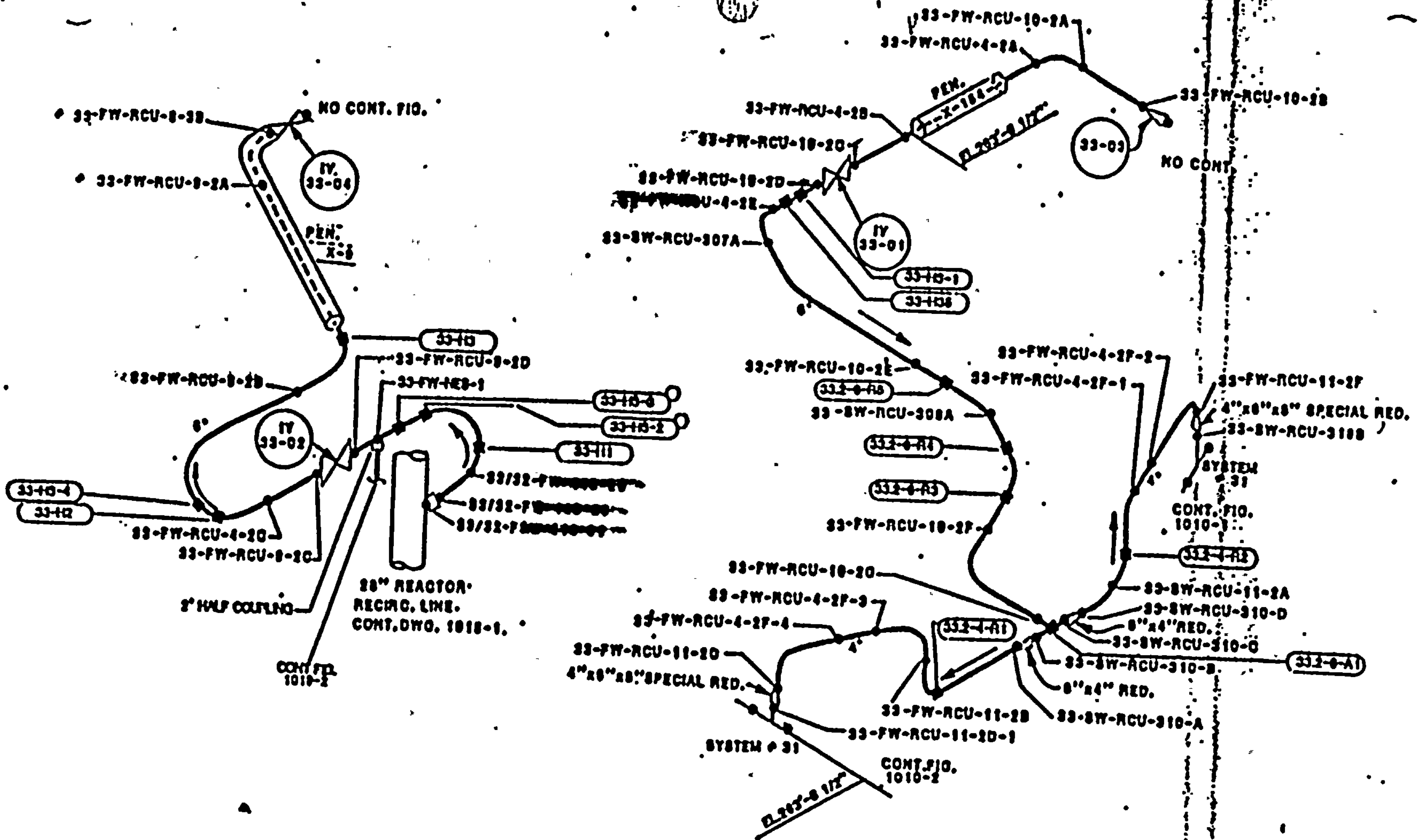
NOTES:

- ● -WELD
- ◻ -SUPPORT
- IIIIII -EXEMPT LINE
- ✱ -INACCESSIBLE WELD

REF. DWGS.
NMP.C.-27166-C REV. 0

NES	
SYSTEM NO. 42.1	TITLE LIQUID POISON WELD & HANGER MAP
REV. 0	DWG. NO. 1016-1
BY PSB	DATE 11/2/83
APP. RAL	DATE 11-9-83
PROJECT 8330-NMP. 1	
NOT TO SCALE	


N
[redacted]W-18B
44.1-FW-18C
[redacted]287-W
44.1-FW-18D
[redacted]FW-4B
44.1-H1
44.1-SW-4A
44.1-FW-18E
44.1-SC-
44.1-FW-18F
SYS. 44
SYS. 44.1
CRBN. STL.
STAINLESS STL.
44-FW-9
44.1-SC-2
301-113
[redacted]W-13
44.1-H3
NO CONT. FIG.
301-112
44-FW-10
44-FW-11
44-FW-12
301-114⊕
44.1-FW-18
44.1-FW-16C
44.1-FW-15A
44.1-H2
[redacted]W-15B
44.1-FW-14
44.1-FW-14A
[redacted]W-14C
44.1-FW-15
44.1-FW-14B
NOTES:
1) ⊕-SUPPLIED BY GENERAL ELECTRIC CO.
REF. DWGS.
N.M.P.C.-C-28848-C REV. 2
nes
SYSTEM NO. 44.1
TITLE CONTROL ROD WELD AND SUPPORT MAP
REV. 0
FIG. NO. 1017-1
PROJECT 5530-N.M.P. 1
NOT TO SCALE

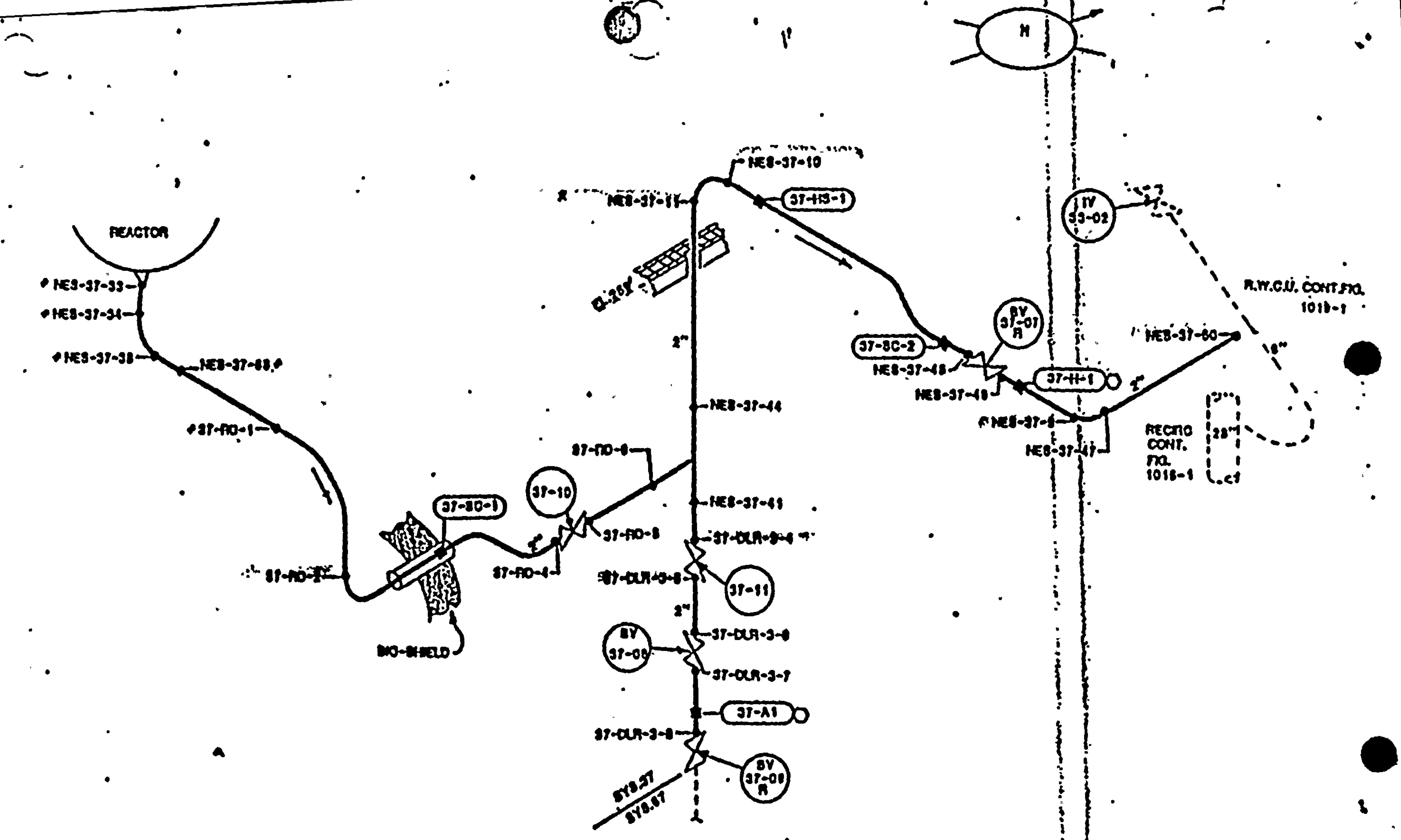


NOTE
INACCESSABLE WELD
O-INTEGRALLY WELDED ATTACHMENT

REF. DWGS.

C-28882-C SHT. 1 REV. 4
C-28838-C REV. 4
C-28882-C SH. 2 REV. 3

NES	
33	REACTOR CLEAN-UP WELD & HANGER MAP
REV 1	FIG. NO. 1018-1
BY PSD	DATE 12/21/84
APP	DATE 1-9-85
PROJECT 8830 NMP 1	



REF.DWGS.

N.M.P.C.-C26882O SH1 REV.4
SH2 REV.3

○ INTEGRALLY WELDED ATTACHMENT
♦ INACCESSABLE WELD

nes

37	REACTOR DRAIN WELD & HANGER MAP
1	1019-2
	CLASS
	8530-NMP. 1

VI CONCLUSIONS

CONCLUSIONS

- FIFTEEN CONTAINMENT PENETRATIONS HAVE SIMILAR GUARD PIPE CONFIGURATIONS.

- EIGHT PENETRATIONS ARE SUBJECT TO PIPE BREAK LOADS IN EXCESS OF DESIGN IF DOUBLE-ENDED PIPE BREAKS INSIDE THE GUARD PIPE ARE ASSUMED.

- MODERN LEAK-BEFORE-BREAK ANALYSES OF THE FOUR AFFECTED SYSTEMS WERE PERFORMED FOR POSTULATED BREAKS OUTSIDE THE DRYWELL. THESE ANALYSES HAVE BEEN REVIEWED AND REVISED FOR THE PORTIONS OF THE PIPING INSIDE THE DRYWELL AT THE PENETRATIONS. RESULTS SHOW:

  -- ADEQUATE MARGIN AGAINST UNSTABLE RUPTURE EXISTS FOR LARGE, POSTULATED THROUGH-WALL FLAWS (90° OF CIRCUMFERENCE).

  -- THE LEAK RATE FROM SUCH FLAWS WOULD EXCEED 1 GPM.

- DRYWELL LEAKAGE MONITORING CAPABILITY EXISTS TO DETECT LEAKS SMALLER THAN 1 GPM.

OTHER MITIGATING FACTORS

- EC LINE BREAK ANALYSIS/CONSEQUENCES

- NEXT REFUELING OUTAGE SCHEDULE

- MUST SHUT DOWN BY 3/30 IN ACCORDANCE WITH ENVIRONMENTAL QUALIFICATION EXTENSION

- CONSERVATIVE REACTOR COOLANT SYSTEM LEAKAGE TECHNICAL SPECIFICATIONS

- EFFECTS OF EMERGENCY CONDENSER STEAM LINE BREAK ON NINE MILE POINT UNIT 1 LIMITED SCOPE PROBABILISTIC SAFETY ASSESSMENT

- ISI PROGRAM

VII REQUEST FOR LICENSE AMENDMENT

A. DOCKET INFORMATION

B. NRC APPROVAL SCHEDULE

PICKARD, LOWE AND GARRICK, INC.
2260 UNIVERSITY DRIVE
NEWPORT BEACH, CA 92660

NEWPORT BEACH, CALIFORNIA
TELEPHONE 714 650-8000
TELEX 3718953

WASHINGTON, D.C.
TELEPHONE 202 296-8633

January 22, 1986
NMPC-830-PLG-42

Mr. David K. Greene
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, NY 13202

Dear Dave:

EMERGENCY CONDENSER STEAM LINE BREAKS AT THE CONTAINMENT PENETRATION AT NINE MILE POINT UNIT 1 (NMP-1)

As you requested, the following discussion summarizes observations on the impact on core melt frequency of emergency condenser steam line breaks at the containment penetration with concurrent penetration failure. These observations are based on the Limited Scope Probabilistic Safety Assessment (PSA) for NMP-1 and our understanding of the plant.

1. Unisolable steam line breaks at the containment penetration with concurrent penetration failure (such as the emergency condenser steam line) are included in the class of initiating events called Large Break LOCAs Inside the Containment (Initiating Event B1). The mean frequency for B1 events has been estimated at 1.76×10^{-3} per calendar year (see Table E.6-1 of the NMP-1 PSA). This frequency distribution was developed using the historical data shown in Table E.6-2 of the NMP-1 PSA and Bayesian methods. The historical experience shows that there have been no B1 type events in over 150 calendar years of operation at 21 BWR plants as of May 31, 1982. Additional experience without a B1 event could be as much as 75 calendar years as of December 1985.

2. Using the frequency in Table E.6-1 and the large break LOCA event tree model for NMP-1 (see Figure 4-12 of the NMP-1 PSA), the mean core damage frequency contribution for all B-1 type events is 3.5×10^{-6} events per calendar year as shown in Table 2-3 of the NMP-1 PSA. This contribution is less than 1% of the total mean core damage frequency estimated in the NMP-1 PSA. This total mean core damage frequency was 5.5×10^{-4} per year.

3. It is important to note that the emergency condenser steam lines are only one of the possible contributors to large break LOCAs. Assuming no physical evidence to indicate that the condenser steam lines are more susceptible to failure at NMP-1, we would conclude that the risk contribution from failures in these lines is less than the contribution for all B1 type events.

In summary, the important facts are:

- Large Break LOCAs Initiating Event Frequency = 1.76×10^{-3}/Calendar Year
- Core Damage Frequency Contribution from Large LOCA Initiating Events = 3.5×10^{-6}/Calendar Year
- Total Core Damage Frequency = 5.5×10^{-4}/Calendar Year

We hope this information adequately summarize our telephone discussion and will be useful in your deliberations. Please call if there are any questions.

Very truly yours,

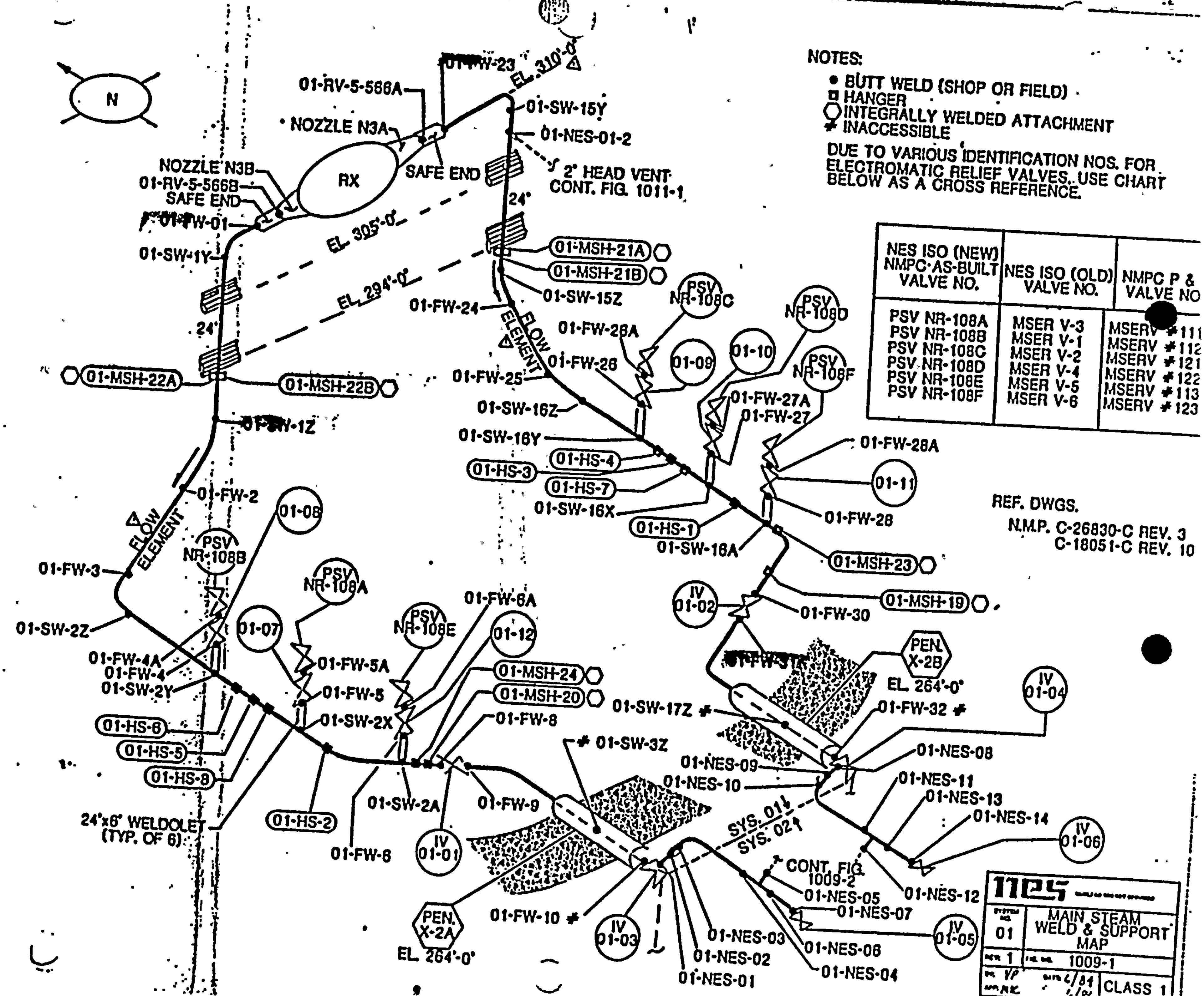
W. C. Gekler

Willard C. Gekler

Attached, please find Inservice Inspection isometrics marked up with a highlighter to indicate ultrasonic and liquid penetrant inspections performed during the last two outages at Nine Mile Point #1. It should be noted that during the end of outage hydro all penetrations are specifically inspected for leakage. Listed below is a synopsis of inspections performed ommiting longitudinal welds that were UT & PT tested.

SYSTEM	PT	UT
Main Steam - 01	6	4
Feedwater - 31	1	2
Rx. Water Cln. Up-33	8	4
Head Spray - 34	20	0
Reactor Drain -37	7	0
Shutdown Cooling -38	6	5
Emergency Cooling - 39	32	3
Core Spray - 40	56	50
Liq. Poison - 42.1	6	0
Control Rod Drive - 44.1	7	6

NOTE: Yellow Highlighter–Liquid Penetrant (PT)
Blue Highlighter–Ultrasonic (UT)



NOTES:

- ● BUTT WELD (SHOP OR FIELD)
- ▫ HANGER
- ⬡ INTEGRALLY WELDED ATTACHMENT
- \# INACCESSIBLE

DUE TO VARIOUS IDENTIFICATION NOS. FOR ELECTROMATIC RELIEF VALVES, USE CHART BELOW AS A CROSS REFERENCE.

NES ISO (NEW) NMPC AS-BUILT VALVE NO.	NES ISO (OLD) VALVE NO.	NMPC P & ID VALVE NO.
PSV NR-108A	MSER V-3	MSERV #111
PSV NR-108B	MSER V-1	MSERV #112
PSV NR-108C	MSER V-2	MSERV #121
PSV NR-108D	MSER V-4	MSERV #122
PSV NR-108E	MSER V-5	MSERV #113
PSV NR-108F	MSER V-6	MSERV #123

REF. DWGS.

N.M.P. C-26830-C REV. 3
C-18051-C REV. 10

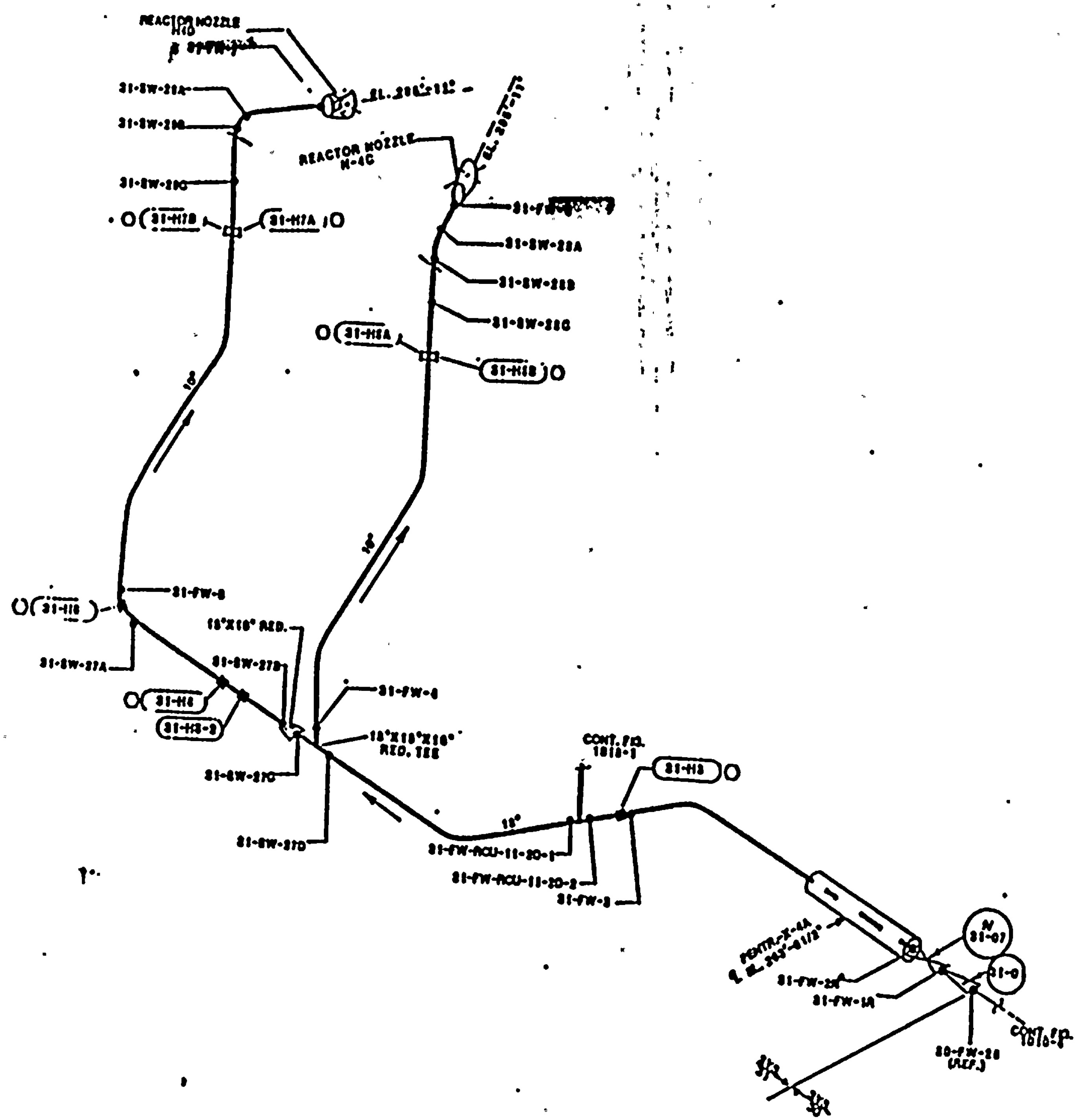
NES

SYSTEM NO. 01

MAIN STEAM WELD & SUPPORT MAP

REV. 1 — FIG. NO. 1009-1

CLASS 1



REF. DWGS.

NMPC C-26838-C REV. 4

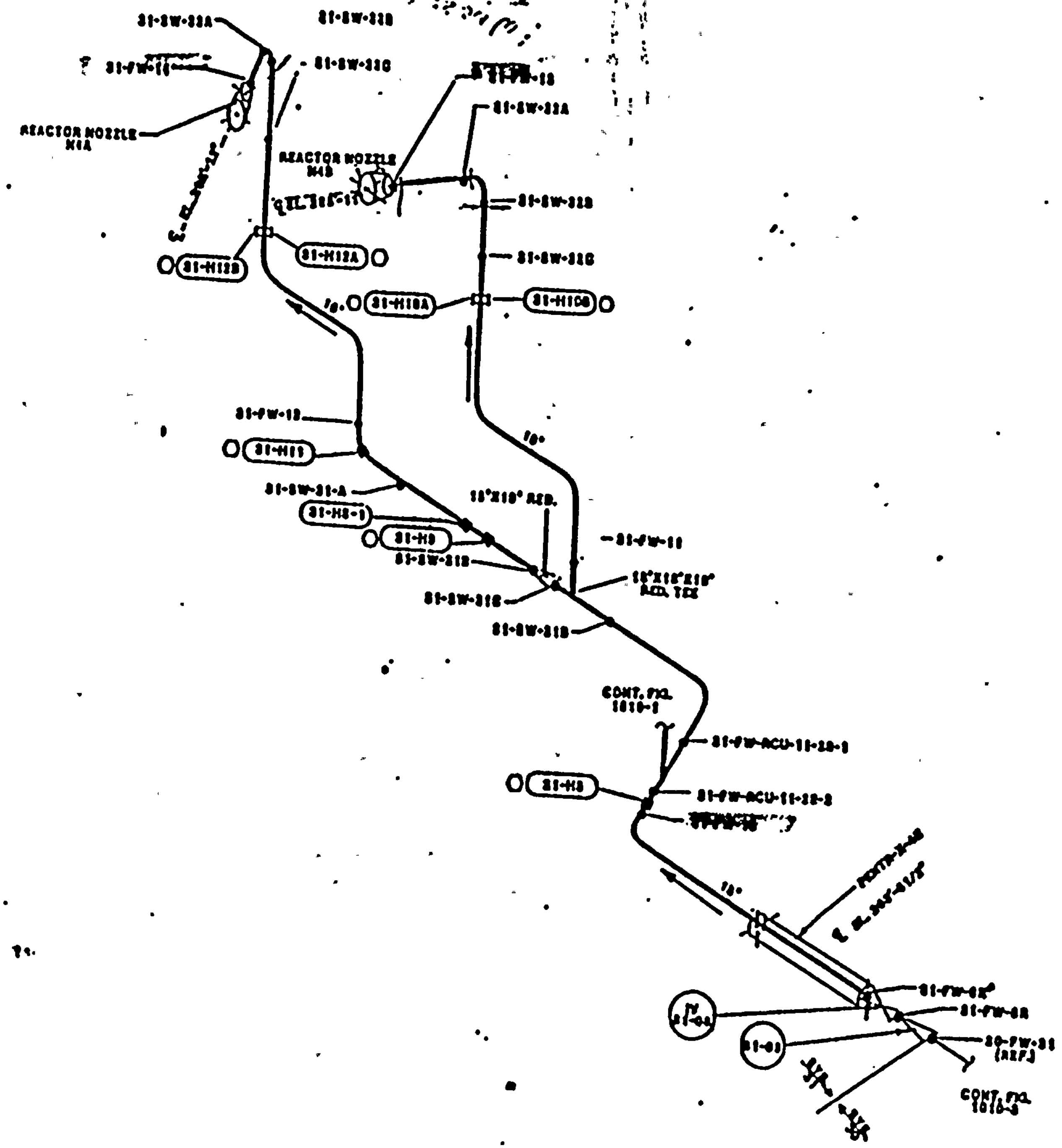
N

NOTES:

- ● INACCESSIBLE WELDS
- ○ INTEGRALLY WELDED ATTACHMENT
- ● BUTT WELD (SHOP OR FIELD)
- ⊡ HANGER

nes	
SYSTEM NO. 31	TITLE CLASS 1 FEEDWATER WELD & HANGER MAP
REV. 0	FIG. NO. 1010-2
DR. PSD	DATE 8/4/93
APP.	DATE
PROJECT 8530-NMP. 1	

NOT TO SCALE



REF. DWGS.

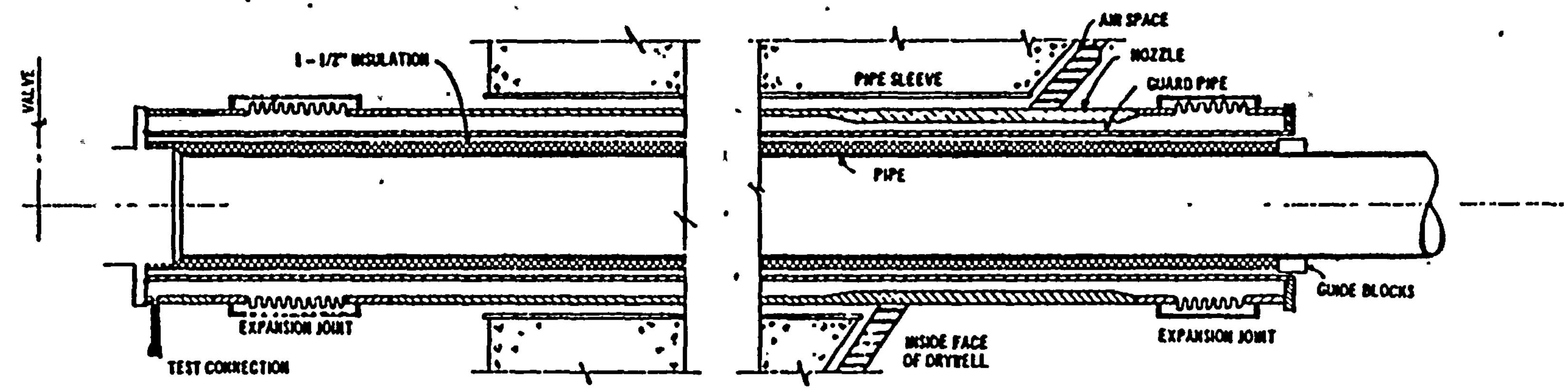
NMPC C-18[illegible]-C REV. 4

NOTES:

- INACCESSIBLE WELD
- ○ INTEGRALLY WELDED ATTACHMENT
- BUTT WELD (SHOP OR FIELD)
- HANGER

nes	
SYSTEM NO. 31	TITLE: CLASS 1 FEEDWATER WELD & HANGER MAP
REV. 0	FIG. NO. 1010-1
PROJECT	8830-N.M.P. 1

TYPICAL HOT PIPE PENETRATION



B. POTENTIAL PENETRATIONS AFFECTED

SYSTEM NAME	SYS NO.	PENETRATION NO.	O.D. PROCESS PIPE DIA.(IN)	I.D. GUARD PIPE DIA.(IN)
MAIN STEAM	01	X-2A & X-2B	24"	28 1/2"
EMERG. COND.(STM)	39	X-3A & X-3B	10 3/4"	15"
FEEDWATER	30/31	X-4A & X-4B	18"	23"
CORE SPRAY	40	X-13A & X-14	12 3/4"	15"
EMERG. COND. (COND.)	39	X-5A & X-5B	10 3/4"	15"
SHUTDOWN COOLING	38	X-7 & X-8	14"	19"
CLEANUP	33	X-9 & X-154	6 5/8"	13"
CONTROL ROD DRIVE	301	X-174	3 1/2"	4 13/16"

III ORIGINAL DESIGN CRITERIA

GUARD PIPE DESIGN

ORIGINAL

- GUARD PIPE PROTECTS DRYWELL STEEL FROM HOT LINE RUPTURE
- HOT FLUID WAS VENTED TO THE DRYWELL

NEW CONSIDERATIONS

- MOMENTUM EFFECTS DUE TO FLOW REVERSAL WITHIN GUARD PIPE
- INCREASED LOADS BY APPROXIMATELY 2

IV CURRENT ANALYSIS AND RESULTS

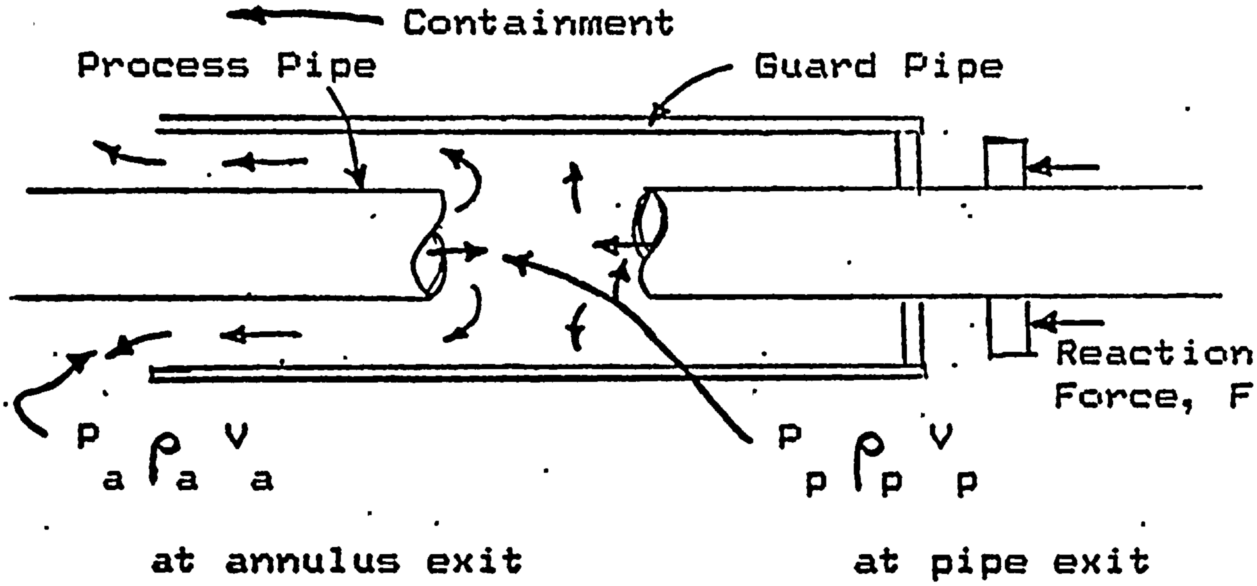
A. SUMMARY OF 15 PENETRATIONS ANALYZED

B. CALCULATIONAL METHOD

PRELIMINARY GUARD PIPE/PENETRATION EVALUATION

PENETRATION	NO.	NORMAL PIPING P/T AT PENETRATION	ORIGINAL DESIGN P.B. LOAD (kips)	CALCULATED ACTUAL P.B. LOAD (kips)	REMARKS
Emer. Cond. - Steam	2	1050 psia, SAT Steam	94	180	Thrust load exceeds design.
Emer. Cond. - Cond.	2	Isolated by Check Valve	—	—	Pipe break not a concern.
Main Steam	2	1050 psia, SAT Steam	240	430	Thrust load exceeds design.
Feedwater	2	1050 psia, 350-550°F Water	125	265	Thrust load exceeds design.
Cleanup Suction	1	1050 psi, 550°F Water	50	61	Thrust load exceeds design.
Cleanup Return	1	1050 psi, 450°F Water	36	66	Thrust load exceeds design.
Core Spray	2	Depressurized	—	—	Pipe break not a concern.
Shutdown Cooling	2	Depressurized	—	—	Pipe break not a concern.
CRD Return	1	Isolated by Check Valve	—	—	Pipe break not a concern.

CALCULATION OF THRUST LOAD



$$F = (P_p + \rho_p V_p^2) A_p + (P_a + \rho_a V_a^2) A_a$$

Fluid pressures, densities, and velocities calculated considering:

* Homogeneous, Steady Flow
* Multi-Node Computer Solution
* Critical Flow where Appropriate
* Upstream Losses Accounted For
* Complete Double-Ended Break Assumed
* Separation of Pipe Ends Exceeds D/4
* Flow from Outside Containment Included when Appropriate

METHOD OF SOLUTION

* Divide Piping into Volumes

* Boundary Fluid Conditions Known

* Volumes Connected by Flow Paths with Resistance and Flow Area

* Solve One Equation for Each Flow Path Either change in Stagnation Pressure:

$$P_{oi} - P_{oj} = F_k W^2$$

Or Choking Condition:

$$W = W_{crit}(P_{oi}, h_o)$$

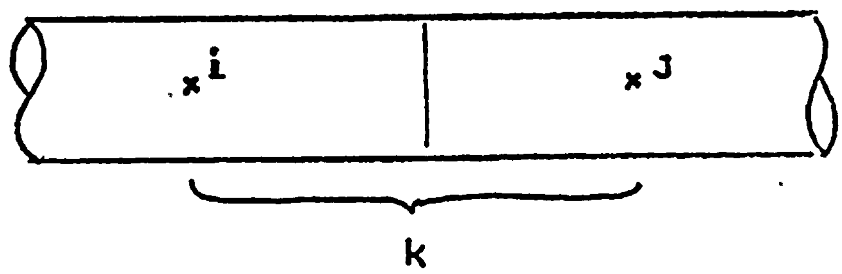
* Unknowns are P_{oi} for Each Internal Volume and W

* Static Conditions and velocity are Calculated from P_{oi}, h_o, and W using Steam Tables:

$$s = s(P_{oi}, h_o) \qquad h_k = h_{ok} - V_k^2/2$$

$$P_k = P_k(h_k, s) \qquad \rho_k = \rho_k(h_k, s)$$

and: $\rho_k V_k A_k = W$



V MITIGATING FACTORS

A. LEAK BEFORE BREAK CRITERIA

B. LEAK DETECTION SYSTEMS

A. LEAK BEFORE BREAK CRITERIA

NMP-1 LEAK-BEFORE-BREAK ANALYSIS RESULTS

- MPR-820, REV. 2, AUGUST 1984
- APPLIED TO PIPING INSIDE DRYWELL

SCOPE OF ANALYSIS

- o IDENTIFY REPRESENTATIVE HIGH ENERGY PIPING SYSTEMS.
- o EVALUATE EXISTING LEAKAGE DETECTION CAPABILITY AT NMP-1 AND ESTABLISH A DETECTABLE LEAK RATE FOR THE REACTOR AND TURBINE BUILDINGS.
- o DEVELOP A THERMO-HYDRAULIC MODEL OF LEAKAGE THROUGH TIGHT CRACKS IN PIPES AND ESTABLISH THE MINIMUM DETECTABLE CRACK LENGTH IN EACH PIPING SYSTEM.
- o PERFORM FINITE-ELEMENT STRESS ANALYSES OF EACH PIPING SYSTEM FOR DEADWEIGHT, PRESSURE AND SAFE SHUTDOWN EARTHQUAKE LOADS.
- o SHOW, USING ELASTIC AND/OR ELASTIC-PLASTIC FRACTURE MECHANICS METHODS, THAT POSTULATED THROUGH-WALL AXIAL OR CIRCUMFERENTIAL CRACKS WILL NOT SHOW SUBSTANTIAL GROWTH UNDER LEVEL D LOADINGS.
- o DEMONSTRATE, WITH ELASTIC-PLASTIC FRACTURE MECHANICS, THAT A 90° CIRCUMFERENTIAL THROUGH-WALL FLAW IS STABLE UNDER EXTREME LOADS.
- o ESTABLISH LEAKAGE MONITORING REQUIREMENTS TO ASSURE THE POSTULATED DETECTABLE LEAK IS DETECTED.

HIGH ENERGY LINE BREAKS IN THE REACTOR AND TURBINE BUILDINGS

SAMPLE PIPING SYSTEMS

- MAIN STEAM
  - FROM ISOLATION VALVES TO THE TURBINE STOP + CONTROL VALVE MANIFOLD
  - CARBON STEEL
  - PIPE SIZES - 16, 18 AND 24 INCH DIAMETER
- HIGH PRESSURE REACTOR FEEDWATER
  - FROM 5TH FEEDWATER HEATER TO ISOLATION VALVES
  - CARBON STEEL
  - PIPE SIZES - 14, 16, AND 18 INCH DIAMETER

- EMERGENCY CONDENSER
  - WESTBANK STEAM SUPPLY AND CONDENSATE RETURN
  - STAINLESS STEEL
  - PIPE SIZES 10 AND 12 INCH DIAMETER
- REACTOR WATER CLEAN-UP
  - FROM ISOLATION VALVES TO THE FIRST HEAT EXCHANGER
  - CARBON STEEL
  - PIPE SIZE - 6 INCH DIAMETER

CRITERIA FOR LEAK-BEFORE-BREAK ANALYSIS

- SHOW ABILITY TO DETECT ONE GPM FLAW UNDER NORMAL CONDITIONS
- SHOW $J < J_{IC}$ FOR ONE GPM + 2T FLAWS UNDER LEVEL D LOADING
- SHOW STABILITY OF 90^{o} FLAWS UNDER LEVEL D LOADING
- SHOW PART THROUGH CRACKS TEND TO BECOME THROUGH CRACKS

FRACTURE MECHANICS METHODS

- LINEAR ELASTIC

  INSTABILITY CONDITION:

  $K \geq K_{IC}$

  INAPPLICABLE TO DUCTILE MATERIALS AT HIGH STRESSES

- ELASTIC-PLASTIC

  INSTABILITY CONDITION:

  $$\left.\frac{DJ}{DA}\right)_{APP} \geq \left.\frac{DJ}{DA}\right)_{MAT}$$

  APPLICABLE TO DUCTILE MATERIALS AT HIGH STRESSES

$$\left.\frac{dM}{d\phi_c}\right|_{crack} < \left.\frac{dM}{d\phi_c}\right|_{piping}$$

TABLE 5

CIRCUMFERENTIAL FLAWS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	TOTAL STRESS @ FLAW(1) (psi)	A_o(2) (in)	J (in-lbs/in^2)	J/J_{IC}(3)	LEAKAGE FLOW (gpm)	Δa(4) (in)
Reactor Cleanup	6.625	0.432	CS	13,947	5.0	210	0.23	1.6	0
Main Steam	16.0	1.031	CS	16,817	10.56	520	0.58	1.8	0
	18.0	1.156	CS	16,388	11.81	537	0.59	1.9	0
	24.0	1.219	CS	17,708	11.34	470	0.52	2.1	0
Reactor Feedwater	14.0	0.937	CS	24,439	6.67	879	0.97	2.5	0
	16.0	1.031	CS	19,832	6.96	375	0.43	2.7	0
	18.0	1.156	CS	25,342	7.41	534(5)	0.59	3.0	0
Emer. Cond. - Condensate	10.75	0.522	SS	21,336	4.44	417	0.42	2.0	0
Emer. Cond. - Steam	12.75	0.622	SS	23,201	7.94	1,317(5)	1.33	1.7	0.17

NOTES:

(1) Total Stress = Bending Stress + Axial Stress + Pressure Stress under Deadweight + Safe Shutdown Earthquake Loading

(2) One gpm flaw size + 2T

(3) Carbon Steel: J_{IC} = 903 in-lb/in^2

Stainless Steel: J_{IC} = 992 in-lb/in^2

(4) Crack growth resulting from applied J.

(5) Calculated with elastic-plastic theory.

TABLE 6

LONGITUDINAL FLAWS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	CIRCUMFERENTIAL PRESSURE STRESS (psi)	A_o (1) (in)	J (in-lbs/in^2)	J/J_{IC} (2)	LEAKAGE FLOW (gpm)
Reactor Cleanup	6.625	0.432	CS	6868	3.36	69	0.07	2.2
Main Steam	16.0	1.031	CS	7097	8.06	130	0.14	3.4
	18.0	1.156	CS	7125	8.01	101	0.11	4.0
	24.0	1.219	CS	9286	7.44	126	0.14	2.7
Reactor Feedwater	14.0	0.937	CS	6794	4.97	36	0.04	4.9
	16.0	1.031	CS	7097	5.16	37	0.04	4.9
	18.0	1.156	CS	7125	5.51	38	0.05	5.2
Emer. Cond. - Condensate	10.75	0.522	SS	9576	3.14	80	0.08	3.4
Emer. Cond. - Steam	12.75	0.622	SS	9527	5.04	Note (3)	Note (3)	2.2

NOTES:

(1) One gpm flaw size + 2T

(2) Carbon Steel: J_{IC} = 903 in-lb/in^2

Stainless Steel: J_{IC} = 992 in-lb/in^2

(3) Plasticity effects precluded a linear elastic calculation. Value of J expected to be similar to condensate line.

TABLE 7

ELASTIC-PLASTIC RESULTS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	APPLIED TEARING MODULUS T (1)	LEVEL D STRESS (psi)	MARGIN TO INSTABILITY (2) (3)
Reactor Cleanup	6.625	0.432	CS	27	13,947	2.10
Main Steam	16.0	1.031	CS	28	16,817	1.75
	18.0	1.156	CS	28	16,388	1.80
	24.0	1.219	CS	31	17,708	1.62
Reactor Feedwater	14.0	0.937	CS	17	24,439	1.29 (4)
	16.0	1.031	CS	10	19,832	1.61 (5)
	18.0	1.156	CS	30	25,342	1.26 (6)
Emer. Cond. - Condensate	10.75	0.522	SS	49	21,336	1.43 (7)
Emer. Cond. - Steam	12.75	0.622	SS	55	23,201	1.32 (8)

NOTES:

(1) Carbon Steel: T_{mat} = 215
Stainless Steel: T_{mat} = 182

(2) Moment required to unstably tear a 90° flawed pipe divided by the equivalent moment resulting in Level D stress in an unflawed pipe.

(3) Unless otherwise indicated, $L/R = \infty$.

(4) L/R = 73. For L/R = 73, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.20.

(5) L/R = 63. For L/R = 63, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.49.

(6) L/R = 110. For L/R 110, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.06.

(7) L/R = 262. For L/R 262, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.15.

(8) L/R = 178. For L/R 178, instability is controlled by plastic collapse rather than unstable tearing. For $L/R = \infty$, Margin = 1.05.

REVISED ELASTIC-PLASTIC RESULTS

SYSTEM	OUTSIDE DIAMETER (in)	WALL THICKNESS (in)	MATERIAL	APPLIED TEARING MODULUS T (1)	OUTSIDE D/W		INSIDE D/W @ PENETRATION	
					LEVEL D STRESS (psi) (3)	MARGIN TO INSTABILITY (2) L/R =	LEVEL D STRESS (psi) (3)	MARGIN TO INSTABILITY (2) L/R =
Reactor Cleanup	6.625	.432	CS	31	13,947	2.10	9,089	3.22
Main Steam	24.0	1.219	CS	30	17,708	1.62	13,974	2.05
Reactor Feedwater	18.0	1.156	CS	66	25,342	1.06	8,534	3.15
Emer. Cond. - Steam	12.75	.622	SS	152	23,201	1.05	11,456	2.13

NOTES:

(1) Carbon Steel: T_{mat} = 215.

Stainless Steel: T_{mat} = 182.

(2) Moment required to unstably tear a 90° flawed pipe divided by the equivalent moment resulting in Level D stress in an unflawed pipe.

(3) Calculated stresses outside drywell are based on conservative, updated seismic criteria; calculated stresses inside drywell are based on original design calculations except for emergency condenser which is based on an updated analysis.

B. LEAK DETECTION SYSTEMS

INFORMATION ON LEAK DETECTION SYSTEM

Type of System	(1) Is System Operable (yes/no)	(2) Leak Rate Sensitivity (gpm)	(3) Time Required To Achieve Sensitivity (hours)	(4) Is System Functional After SSE (yes/no)	(5) Control Room Indications (alarms) (recorders)	(6) Calibration or Testing During Operation (yes/no)	(7) Documentation Reference for (1) Thru (6)
Rate of Rise (Level vs. Time	Yes	For inflows of 1 gpm: 0.2 gpm For inflows of 1-5 gpm: 0.5 gpm	.66 hours .13 hours	No	A Alarm-High Level B Recorder	Yes (Daily checks)	First ad Fifth Supplement to Final Safety Analysis Report and the Technical Specifications
Rate of Rise (rate of change)	Yes	0.25 gpm	0.03 hours	No	A Alarm-High Rate B Recorder	Yes	Niagara Mohawk Procedure - Drywell Floor Drain Rate of Rise Instrument Channel Calibration
Timer (with level sensor)	Yes	5.0 gpm	0.30 hours	No	A Alarm-High Level	Yes	First and Fifth Supplement to Final Safety Analysis Report and the Technical Specifications

PENETRATION DETAILS

SYSTEM NO.	NAME	PENETRATION	DWG. NO.
01	MAIN STEAM	X-2A, X-2B	C-18056-C
31	FEEDWATER	X4A, X-4B	C-18120-C
39	EMER. COND. STEAM SUPPLY	X-3A, X-3B	C-18359-C SHEET 1
39	EMER. COND. CONDENSATE RETURN	X-5A, X-5B	C-18359-C SHEET 2
38	SHUTDOWN COOLING	X-7, X-8	C-18363-C
40	CORE SPRAY	X-13A, X-14	C-18369-C
42.1	LIQUID POISON	X-131	C-18415-C
44.1 (301)	C.R.D. EXHAUST WATER TO REACTOR	X-174	C-18666-C
33	REACTOR WATER CLEAN-UP	X-9	C-18683-C
33	REACTOR WATER CLEAN-UP	X-154	C-18684-C
36 (MISC.)	REACTOR INSTR. AND MISC. PIPING IN DRYWELL	MISC.	C-18461-C
110	REACTOR BLDG. SAMPLING	X-139	C-18347-C

TI APERTURE CARD

8602040483-01 C

ISOMETRIC DWGS.

SYSTEM NO.	NAME	PENETRATION	DWG. NO.
01	MAIN STEAM	X-2A, X2B	C-26830-C
39	EMER. COND. STEAM SUPPLY	X-3A, X-3B	C-26843-C SHT. 1&2
39	EMER. COND. CONDENSATE RETURN	X-5A, X-5B	C-26843-C SHT. 3&4
31	FEEDWATER - EXT. ISO. VALVE TO REACTOR	X-4A, X-4B	C-26839-C
40	CORE SPRAY	X-13A, X-14	C-26844-C
38	SHUTDOWN COOLING	X-7, X-8	C-26847-C SHT. 1
42.1	LIQUID POISON - EXT. ISO. VALVE TO REACTOR	X-131	C-27168-C SHT. 1
44.1 (301)	CRD EXHAUST TO REACTOR	X-174	C-26849-C
33	REACTOR WATER CLEAN-UP FROM VESSEL DRAIN	X-9	C-26852-C SHT. 1
33	REACTOR WATER CLEAN-UP RETURN TO FEEDWATER	X-154	C-26852-C SHT. 1

8602040483-02C

P&ID DWGS.

SYSTEM NO.	NAME	DRAWING NO.	
01	MAIN STEAM	C-18002-C SHT. 1	
31	FEEDWATER	C-18005-C SHT. 2	
40	CORE SPRAY	C-18007-C	
33	REACTOR WATER CLEAN-UP	C-18009-C SHT. 1	
44.1	CRD EXHAUST WATER TO REACTOR	C-18016-C SHT. 1	
39	EMER. COND. STEAM SUPPLY AND CONDENSATE RETURN	C-18017-C	
38	SHUTDOWN COOLING	C-18018-C	

TI APERTURE CARD

8602040483-03 C