REPORT ON THE

REANALYSIS OF SAFETY-RELATED PIPING SYSTEMS

SURRY POWER STATION, UNIT 1

JUNE 5, 1979

WEBSTER

REPORT ON THE
REANALYSIS OF SAFETY-RELATED PIPING SYSTEMS
SURRY POWER STATION, UNIT 1
VIRGINIA ELECTRIC AND POWER COMPANY

JUNE 5, 1979

Stone & Webster Engineering Corporation
Boston, Massachusetts

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY, RICHMOND, VIRGINIA 23261

June 5, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 453
PSE&C/CMRjr:mc

Docket Nos.: 50-280
50-281

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Dear Mr. Denton:

REPORT ON THE REANALYSIS OF SAFETY
RELATED PIPING SYSTEMS
SURRY POWER STATION UNIT 1

The Nuclear Regulatory Commission Order to Show Cause of March 13, 1979 required that certain piping systems associated with Surry Power Station Units 1 and 2 be reanalyzed using an appropriate piping code to account for seismic loads. We complied with the Order requiring shutdown of the Units within 48 hours.

Since that time, an intense effort has been under way to analyze all affected piping systems in a manner acceptable to the NRC staff and commensurate with our commitment to provide a safe and reliable source of power for our customers. We have had the benefit of numerous discussions with the NRC staff to clarify and amplify their specific concerns with regard to the details of our reanalysis effort. We have been and are totally committed to provide the staff, on an expedited basis, with any information they require for their review of the Surry units.

We believe the culmination of the pipe stress analysis effort is at hand. The analysis to date, while continuing, has shown that the piping systems are impacted only slightly even after a thoroughly rigorous reanalysis. It has been unequivocally demonstrated that the impact on the piping systems is wholly incompatible with the severity of the Commission's Order. It is on this basis that we submit the attached Report and request immediate start up of Surry Power Station Unit 1.

Correspondence with the staff has transmitted a vast amount of information between the parties. A compilation of the transmitted information is tabulated in the attached Report in Appendix G for your convenience and reference.

We would not feel justified in requesting an immediate lifting of the Order if the reanalysis had shown demonstrable and persistent modifications to piping systems. Such has not been the case. For example, of the approximately 74 piping problems to be reanalyzed, 29 have been completed as of June 2, 1979. Results show that no piping of any size will have to be replaced or repaired. Of the approximately 873 total supports to be reanalyzed, 138 analyses have been completed. None of those supports will require modification. In a cursory look at the balance of the supports to be completely reanalyzed, we have so far identified four supports which will require some modification. These modifications include addition of one snubber, shimming of one support and lateral braces for two supports. These modifications are not only minor, they do not even occur because of seismic stress conditions. Modifications are discussed in some detail in Section 5 of the Report. On the basis of these analyses and conservatisms contained in our analysis techniques as explained in our attached Report, we believe we have substantial justification for start up of Surry Unit 1.

As we continue our reanalysis effort, it is possible that other potential support modifications may surface. We will evaluate each of these potential modifications on a case by case basis in accordance with the guidelines delineated in Section 5 of the attached Report. We have several methods available to evaluate the necessity of a potential modification. For those modifications which we deem to be major in nature, we will contact you and solicit your involvement. Such modifications, once identified, will be expedited.

Modifications for the design basis earthquake (DBE) case which are considered to be less than major in nature in accordance with the guidelines in Section 5 of the Report will be made at advantageous times in the operating schedule of the unit.

The following two paragraphs specifically address the two items of your May 25, 1979 letter.

Your letter of May 25 requested information regarding operating basis earthquake (OBE) design requirements. For those supports which meet DBE requirements but do not meet the FSAR OBE design requirements, we have not as yet identified a requirement to reduce the present FSAR OBE design value. We will evaluate the OBE requirement as stipulated in Item 2 of your May 25 letter on a continuing basis for those piping systems which meet DBE design requirements but do not meet OBE design requirements. The basis for evaluation will be amplified response spectra (ARS) compatibility between the DBE and OBE cases. That is, if soil structure interaction is used in a piping system evaluation for the DBE case, it will also be used in the OBE case.

Your May 25 letter also requested information on the capability of piping systems to safely withstand all earthquakes up to and including the DBE. An investigation of the effects of earthquakes smaller than the DBE leads to the conclusion that the effects of the DBE are not exceeded by smaller earthquakes. This investigation will be covered in Section 7 of a detailed report on SSI-ARS to be submitted on or before June 8, 1979. Capability of piping systems can also be addressed in terms of the numerous conservatisms involved in the overall analysis. These are addressed in detail in the attached Report in Section 7.

Enclosure three of your April 2 letter addressed verification of certain computer codes, including the NUPIPE code being used on Surry Units 1 and 2, with standard benchmark problems developed by the staff and Brookhaven National Laboratory. These have all been previously forwarded except for one benchmark problem involving the analysis of a two loop NSSS, the results of which will be forwarded to the staff on or before June 8, 1979 by Stone & Webster Engineering Corporation.

Prolonged discussions have been held with the staff regarding the methodology and use of soil structure interaction in the development of amplified response spectra. A detailed report is presently being prepared to fully describe its use on Surry Units 1 and 2. The report will be submitted on or before June 8, 1979 and will be entitled "Soil Structure Interaction in the Development of Amplified Response Spectra for Surry Power Station, Units 1 and 2."

With the submittal of the SSI-ARS report (on or before June 8, 1979), the submittal of the two loop benchmark problem (on or before June 8, 1979), the submittal of information regarding the status and schedule of IE Bulletin 79-02 (letter dated June 4, 1979, Serial No. 146/030879A), and the information contained in the Report attached to this letter, we believe we have complied with all of the staff's outstanding requests for information. We plan no further submittals, except the final report on the piping analysis, unless subsequent evaluation of the above information by the staff leads to further inquiries. Because of the severe economic consequences of the present shutdown status of the plant, we plan to respond as quickly as possible to any questions the staff may have. However, we believe there is sufficiently detailed information available to the staff from this and past submittals, meetings, and telephone conversations to evaluate quickly and with confidence our request to lift the Order and resume operation of Unit 1.

We believe it to be in the best interests of our customers and the citizens of the Commonwealth of Virginia to minimize this country's dependence on oil. For Surry to be allowed to restart and to function during the coming hot months is commensurate with that goal. To be allowed

to do this requires a commitment to address safety concerns to the satisfaction of both ourselves and the NRC. We believe we have gone the extra mile in the case of the Surry pipe stress reanalysis effort and our findings fully justify our position to start up.

As stated in Section 4 of the Report, all reanalysis of Unit 1 systems will be completed and fully reviewed by Engineering Assurance personnel by October 1, 1979.

The staff's accessibility during our reanalysis effort is gratefully acknowledged and appreciated.

Prompt consideration and affirmation of our proposal would be appreciated.

Very truly yours,

W. C. Spencer

W. C. Spencer
Vice President - Power Station
Engineering and Construction Services

Attachment

TABLE OF CONTENTS

SECTION 1

SUMMARY AND CONCLUSIONS

In response to the Nuclear Regulatory Commission's Order to Show Cause, dated March 13, 1979, a reanalysis is being conducted of safety-related piping systems which were originally dynamically analyzed using the SHOCK2 computer program. This program, which used an earlier load combination methodology, is no longer considered acceptable by the NRC.

This report addresses details of the analysis work, results of pipe and support analyses to date, a discourse on conservatisms, and other topics within the scope of the reanalysis task. It is in support of our effort to restart Unit 1 as discussed in the transmittal letter with this report. The report is regarded as a culmination of all work to date and is in addition to other submittals previously forwarded since the Order to Show Cause. A listing of correspondence with the NRC, through the date of this report, is included in Appendix G for reference.

The seismic reanalysis is based on a piping analysis program, NUPIPE, that uses methodology currently acceptable to the NRC. The results to date indicate that the subject systems will be able to perform their intended

safety functions under the maximum seismic conditions specified in the Final 2.1
Safety Analysis Report. The reanalysis effort has demonstrated the 2.2
conservative nature of the original seismic analysis. The piping systems have 2.3
been found to be impacted only slightly after a thoroughly rigorous
reanalysis. Results also show that no piping of any size will have to be 2.5
replaced or repaired. A few systems may require addition of minor pipe 2.6
support hardware to limit stresses to code allowable values; however, these 2.7
changes are due to reasons other than the algebraic summation process.

In addition to the systems formerly analyzed with SHOCK2, systems which were 2.8
originally analyzed with SHOCK0 and SHOCK1 (predecessors of SHOCK2) are being 2.9
reevaluated to demonstrate existing seismic adequacy.

SECTION 2 1.7

SCOPE OF REANALYSIS 1.9

As described in the NRC Order to Show Cause, March 13, 1979, some piping 1.15
systems in the Surry Power Station, Unit 1, were dynamically analyzed with a 1.16
computer program that is not currently acceptable to the NRC. 1.17

In response to the Order to Show Cause, the following actions were taken: 1.18

1. Safety systems or portions thereof that were dynamically analyzed 1.20
using the computer program SHOCK2 were identified. These are listed 1.22
in Appendix A. The specific piping reanalyzed is shown on the flow 1.23
diagrams in Appendix B.

2. These systems are being reanalyzed using computer programs based on 1.25
methodology currently acceptable to the NRC. These programs are 1.26
discussed in Sections 3 and 4.

3. Results of the reanalysis are compared with code allowable pipe 1.27
stresses, with allowable loads for nozzles/penetrations, and with the 1.28
results of original design loads for pipe supports.

4. In those cases where the reanalysis indicated that stresses or loads 1.29
may be in excess of allowable values, using the newer methodology, 2.1

further examination is initiated and, where required, equipment 2.2
modifications are identified.

In addition to the analyses addressed in the Order to Show Cause, those 2.4
problems originally analyzed with the SHOCKO and SHOCK1 programs were 2.5
identified and are being reanalyzed to demonstrate existing seismic adequacy.
This investigation is discussed in Section 5.3. 2.6

Information regarding safety-related piping that was not originally subjected 2.7
to computer seismic analysis (for example, small diameter pipe done by hand 2.8
calculation) is included in the VEPCO letter of May 24, 1979 responding to the 2.9
NRC letter dated April 2, 1979.

SECTION 3

PIPE STRESS RESULTS

A total of 87 pipe stress problems have been identified for reanalysis.

Of these, 69 problems were originally analyzed by the PSTRESS/SHOCK2 computer program that used algebraic summation and are therefore specifically addressed by the Show Cause Order, 6 were hand calculations, and 12 problems were originally analyzed by various versions of PSTRESS/SHOCK0. This latter program is not specifically addressed by the Show Cause Order but is now considered not equivalent to currently accepted practice. These stress problems are being analyzed by two groups: Stone & Webster Engineering Corporation in Boston, Massachusetts, and Nuclear Services Corporation in Campbell, California, as indicated in the following table:

PIPE STRESS PROBLEMS

	S&W	NSC	TOTAL
SHOCK2	42	27	69

SURRY POWER STATION, UNIT 1

SHOCK1	5	7	12	2.2
HAND CALCULATIONS	6	0	6	2.3
TOTAL	53	34	87	2.4

Status as of June 2, 1979 2.6

Field verified piping isometric drawings provide the basis for program inputs 2.9
for the pipe stress reanalysis of the SHOCK2 problems. The reanalysis is 2.11
conducted using the NUPIPE computer program. NUPIPE calculates intra-modal 2.12
seismic forces using a modified square root of the sum of the squares (SRSS)
technique which is always more conservative than the approved SRSS method, and 2.13
an SRSS technique for inter-modal combination.

Additionally, in some cases, piping is analyzed utilizing amplified response 2.14
spectra (ARS) that are developed using soil structure interaction techniques 2.15
(SSI-ARS). The resultant stresses and loads are used to evaluate piping, 2.16
supports, nozzles, and penetrations. These techniques are discussed in 2.17
Section 7.7. In accordance with the NRC letter of May 25, 1979 to Virginia 2.18
Electric & Power Company (VEPCO), the seismic inertial stresses computed using
the SSI-ARS have been increased by a factor of 1.5 for the DBE condition. 2.19

Of the 69 SHOCK2 problems, 24 have been reanalyzed and approved by the Stone & 2.20
Webster Engineering Assurance Division up to this time. This constitutes a 2.22
sample of 24/69 = 35% of the total SHOCK2 stress problems at Surry 1. Table 2.23
3-1 shows the list of problems to be reanalyzed including the results for
these 24 SHOCK2 problems plus 5 of the total of 6 hand calculations rerun on 2.24
NUPIPE and 4 of the 12 SHOCK1 problems, for a total of 33 completed, accepted 2.25
stress problems. In Table 3-1, the figures for Original Total Stress, at the 2.26
point of maximum total stress in the pipe, and Original Seismic Stress, at the 2.27
same point, are extracted from Table 4.1 of the "Seismic Design Review
Equipment and Piping, Surry Power Station," dated September 15, 1971. The 2.29
original calculations for the seismic design review are no longer available,
and correlations were made to the original stresses in Table 4.1 on the basis 3.1
of the MSK's. In some cases, particularly where hand calculations were used 3.2
(problems 1020A, 1020B, 1020C, 1030, and 1010A), the original stresses are not 3.3
available.

In Table 3-1, the columns for New Total Stress, at the point of maximum total 3.4
stress in the pipe, and New Seismic stress, at the same point, were taken from 3.5
the NUPIPE computer runs with the seismic inertial stress magnified by a
factor of 1.5 for runs using the SSI-ARS, per the NRC letter to VEPCO of 3.6
May 25, 1979. Of the 33 completed problems in Table 3-1, 24 used the SSI-ARS 3.7

and 8 used the original ARS. The stresses after the 1.5 magnification for 3.8
SSI-ARS are below the allowable stress for all 33 completed problems.

The original total and original seismic stresses shown in Table 3-1 were 3.9
computed using the SHOCK2, SHOCK1, or SHOCKO programs or hand calculated for 3.10
the original design conditions. The new total and new seismic stresses were 3.11
computed by the NUPIPE program using different mass models and in some cases 3.12
different ARS's than the original calculations. More importantly, the 3.13
reanalyses were based on field-verified, as-built conditions in 1979, which in
some cases differ significantly from the original design conditions. For this 3.15
reason, the new stresses and the original stresses in Table 3-1 are not
comparable, as they do not necessarily represent the same physical conditions. 3.16

Table 3-2 summarizes the nozzles and penetrations evaluated under the 3.17
reanalysis program. Of a total of 67 nozzles on problems originally analyzed 3.18
by the SHOCK2 program, 24 have been evaluated and found to be acceptable, 9 3.19
are under evaluation, and 34 are problems for which the pipe stress analysis 3.20
is not complete and nozzle loads are not yet available. None of the nozzles 3.21
for which evaluation is complete has been found to be unacceptable. For those 3.22
problems in which the SSI-ARS are used, the seismic inertial nozzle loads have
been increased by a factor of 1.5 per the NRC letter of 25 May 1979. There 3.24
are an additional 30 nozzles in problems which were originally computed by the

SHOCKO program or by hand calculations; of these, 5 have been evaluated to date and all are acceptable. 3.25

The SHOCK2 stress problems include 8 penetrations, of which 2 have been evaluated and found to be acceptable. The remaining six are in problems for which analysis is not complete, consequently the loads on the penetrations are not yet available. 3.27 3.28 4.1

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TABLE 3-1

PIPE STRESS REEVALUATION SUMMARY

NA - Not Available
*Table 4-1 of Seismic Design Review Equipment and Piping Surry Power Station, Sept. 15, 1971 or Subsequent Reanalysis**

Preliminary			Criteria	Pipe Stress (psi)				
Problem No.	System Name	Iso. No.	Line Size (NPS)	Original Total*	Original Seismic*	New Total	New Seismic	Allowable
			SHOCK2 Problems					
555	Low Head Safety Injection	122 D1	10" 12"	29290	NA	11180	5307	30882
1555	Low Head Safety Injection	122 L1	12"	25290	NA	10392	3855	30882
706A	Low Head Safety Injection	122 H1	6"	18451	10707	19830	8439	30769
707A	Low Head Safety Injection	122 J1	6"					
708	Low Head Safety Injection	122 K1	6"					
731A	Low Head Safety Injection	127 E1	8"	22671**	984	21503	13940	24750
731B	Low Head Safety Injection	127 E2	8"	22671**	984	21800	16004	24750
743	Low Head Safety Injection	127 F1	10"	24649**	3738	16119	5496	33750
727	Low Head Safety Injection	127 C1 127 C2	6" 10"					
735	High Head Safety Injection	127 G1 127 G2	4", 6" 8", 10"					
525A/ 1525A	Containment & Recirculation Spray	123 A1	8" 10"	11999	10866	9409	3846	33750
546/ 560	Containment & Recirculation Spray	123 D1 123 E1	8" 10"	28209	24753	31976	16024	32616

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Prob-lem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allow-able	
546/ 5600	Containment & Recirculation Spray	123 F3	8" 10"						2.23 2.24
546/ 5620	Containment & Recirculation Spray	123 F2	8" 10"						2.27 2.28
548C	Containment & Recirculation Spray	123 H2	10"						2.31 2.32
547	Containment & Recirculation Spray	123 C1	8" 10"	20953	5688	21960	19284	31482	2.35 2.36
744/ 754	Containment & Recirculation Spray	123 J1	8"						2.39 2.40
548A	Containment & Recirculation Spray	123 B1	8" 10"						2.43 2.44
548B	Containment & Recirculation Spray	123 H1	10"	28660	26790	23251	18529	32616	2.47 2.48
544	Containment & Recirculation Spray	123 G1 123 G2	10"	13402	6986	6386	3766	28485	2.51 2.52
544A	Containment & Recirculation Spray	123 R2	10"	12853	11256	6814	3556	29970	2.55 2.56
544B	Containment & Recirculation Spray	123 R1	10"	12853	11256	6628	4541	28485	3.1 3.2
751	Containment & Recirculation Spray	123 N1 123 N2	10"	6010	5169	7085	5206	28485	3.5 3.6
562/ 546	Containment & Recirculation Spray	123 F1 123 E2	10"						3.9 3.10
745	Containment & Recirculation Spray	123 K1	8"						3.13 3.14
323A	Main Steam	100 D1	30"	13824	6343	13064	354	27000	3.17

SURRY POWER STATION, UNIT 1

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Problem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allowable	
322A	Main Steam	101 D1	30"	13031	5548	11532	400	27000	3.19
334A	Main Steam	102 D1	30"	18635	11082	15407	463	27000	3.21
346	Main Steam	103 A1	30"						3.23
323B	Feedwater	100 G1	14"	15829	590	12923	8061	27000	3.25
322B	Feedwater	101 G1	14"	17927	13521	15965	1796	27000	3.27
334B	Feedwater	102 G1	14"	16025	12281	16145	9828	27000	3.29
417	Auxiliary Feedwater	118 A1 118 A2	3"	8568	NA	26769	14036	27000	3.32 3.33
607	Auxiliary Feedwater	118 G1 118 G2	4" 6"	18681	NA	18331	5467	27000	3.36 3.37
636	Pressurizer Spray & Relief	125 A1	4"						3.40 3.41
630	Pressurizer Spray & Relief	124 A1 124 A2	3", 4" 6", 12"						3.44 3.45
540	Residual Heat Removal	117 B1	3", 4" 6", 12"						3.48 3.49
508	Residual Heat Removal	117 A1 117 A2	10", 12", 14"						3.52 3.53
465	Service Water	119 A1 119 A2 119 A3 119 A4	24"	19101	18285	7778	5826	21600	3.56 3.57 3.58 4.1
488/ 480	Component Cooling	112 C 112 A1	18"						4.4 4.5
507/ 481	Component Cooling	112 F1 112 B1	8" 18"						4.8 4.9

SURRY POWER STATION, UNIT 1

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Problem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allowable	
614	Component Cooling	112 AE1 112 AE2	12" 18"						4.12 4.13
512	Component Cooling	112 AN1	18"						4.16
603A	Component Cooling	112 S1	18"						4.18
766	Component Cooling	112 AR 112 T	8"						4.21 4.22
605A	Component Cooling	112 AA1 112 AA2	3", 6", 18"						4.25 4.26
605B	Component Cooling	112 AA1 112 AA2	3", 6", 18"						4.29 4.30
509A	Component Cooling	112 G1	8", 12", 18", 24"						4.33 4.34
612	Component Cooling	112 AK1	18"						4.37
1512	Component Cooling	112 J	18"						4.39
2529	Component Cooling	112 AH	3", 6", 8", 10", 14", 18"						4.42 4.43 4.44
2526	Component Cooling	112 AJ	2'2", 6", 8", 10"						4.47 4.48
2527	Component Cooling	112 AL	4", 6", 8", 10"						4.51 4.52
527A	Component Cooling	112 T1	4", 6", 8", 10", 14"						4.55 4.56 4.57
517	Component Cooling	112 M1 112 M2 112 M3	4", 6", 8", 10", 14", 18"						5.2 5.3 5.4

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Prob-lem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allow-able	
603B	Component Cooling	112 S1	18"						5.7
526A	Component Cooling	112 L3	6", 8"						5.9
526B	Component Cooling	112 L1	6", 8"						5.11
526C	Component Cooling	112 L1 112 L2 112 L3	6", 8"						5.14 5.15 5.16
527B	Component Cooling	112 T2	4", 6", 8", 10", 14"						5.19 5.20 5.21
527D	Component Cooling	112 T3	4", 6", 8", 10", 14"						5.24 5.25 5.26
509B	Component Cooling	112 G2	8", 12", 18", 24"						5.29 5.30
509C	Component Cooling	112 G3	8", 12", 18", 24"						5.33 5.34
509D	Component Cooling	112 G4	8", 12", 18", 24"						5.37 5.38
CV1	Containment Vacuum	137 A1	8"	25750**	1029	17554	1209	27000	5.41
746	3" HP Steam	131 A1 131 B2 131 C3	3" 4"						5.44 5.45 5.46
CF1	Fire Protection	144 A1	2", 6", 12"						5.49 5.50
CF2	Fire Protection	144 B1	1 1/2", 2", 16"						5.53 5.54
1040	Diesel Muffler Exhaust	143 A1	24"						5.57

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Problem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allowable	
Other Problems (Hand Calculations and SHOCKO/1)									6.3
1000A	Low Head Safety Injection	127 J1	2", 6"						6.6 6.7
1010A	Low Head Safety Injection	127 J2	2" 6"	NA	NA	24709	5423	33750	6.10 6.11
1020A	Low Head Safety Injection	127 J3	2" 6"	NA	NA	28401	6168	33750	6.14 6.15
1020B	Low Head Safety Injection	127 J4	6"	NA	NA	12305	5587	33750	6.18 6.19
1020C	Low Head Safety Injection	127 J5	6"	NA	NA	22453	1270	33750	6.22 6.23
1030	Service Water	1119 A1	24"	NA	NA	3092	1421	21600	6.26
537	Low Head Safety Injection	122 A1 122 A2	4", 6", 10", 12"	19247	13944	22928	17042	25789	6.29 6.30
755	Containment & Recirculation Spray	123 P1	12"	3950	2235	2400	867	33750	6.33 6.34
756	Containment & Recirculation Spray	123 Q1	12"	2077	1230	2638	1205	33750	6.37 6.38
611	Auxilary Feedwater	118 L1	4", 6"						6.41
554	Residual Heat Removal	117 C1	6"	16627	12375	25955	21992	32238	6.44 6.45
606	Component Cooling	112 AB1	12"						6.48
613	Component Cooling	112 AD1	3", 4", 8", 6"						6.51 6.52
502	Component Cooling	112 D1	18"						6.55

TABLE 3-1 (Cont)

PIPE STRESS REEVALUATION SUMMARY

Preliminary

Problem No.	System Name	Iso. No.	Line Size (NPS)	Pipe Stress (psi) Original Total*	Original Seismic*	New Total	New Seismic	Allowable	
506	Component Cooling	112 E1 112 E2 112 E3	18"						6.58 7.1 7.2
747	Spent Fuel Cooling	128 A1	12"						7.5
748	Spent Fuel Cooling	128 C1	12", 16"						7.7
749	Spent Fuel Cooling	128 B1	12"						7.9

Legend: 7.12

Allowable Stress = 1.8 S_h 7.14

Total Stress = $S_{LP} + S_{DW} + 1.5\, S_{DBEI} + S_{DBEA}$ 7.16

Seismic = $1.5\, S_{DBEI} + \cancel{S_{DBEA}}$ 7.18

TABLE 3-2

SURRY POWER STATION, UNIT 1

NOZZLE AND PENETRATION SUMMARY

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment
		SHOCK2 PROBLEMS			
Low Head Safety Injection					
555	1	1	0	0	SSI-ARS
1555	1	1	0	0	SSI-ARS
706A	0	N/A	N/A	N/A	SSI-ARS
707A	0	N/A	N/A	N/A	
708	0	N/A	N/A	N/A	
731A	0	N/A	N/A	N/A	Original ARS
731B	0	N/A	N/A	N/A	Original ARS
743	0	N/A	N/A	N/A	
727					
High Head Safety Injection					
735	3	*	*		Incomplete
Containment Recirculation Spray					
525A/1525A	0	N/A	N/A	N/A	SSI-ARS
546/560	1	1	0	0	SSI-ARS

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
546/5600	0	N/A	N/A	N/A		2.21
546/5620	0	N/A	N/A	N/A		2.23
548C	1	0	0	0		2.25
547	0	N/A	N/A	N/A	Original ARS	2.27
744/754	1	0	1		Incomplete	2.29
548A	0	N/A	N/A	N/A	SSI-ARS	2.31
548B	1	1	0	0	SSI-ARS	2.33
544	2/2	2/2	0/0	0/0	SSI-ARS	2.35
544A	2	2	0	0	SSI-ARS	2.37
544B	2	1	1	0	Original ARS	2.39
751	2	2	0	0	Original ARS	2.41
562/546	0	N/A	N/A	N/A		2.43
745	1	0	1		Incomplete	2.45
Main Steam						2.48
323A	1/1	1/*	0/*	0/*	SSI-ARS	2.50
322A	1/1	1/*	0/*	0/*	SSI-ARS	2.53
334A	1/1	1/*	0/*	0/*	SSI-ARS	2.55
346	0/0	N/A	N/A	N/A	Incomplete	2.57
Feedwater						3.2
323B	1/1	1/*	0/*	0/*	SSI-ARS	3.4
322B	1/1	1/*	0/*	0/*	Original ARS	3.7

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
334B	1/1	1/*	0/*	0/*	SSI/ARS	3.9
Aux. Feedwater						3.12
417	0	N/A	N/A	N/A	SSI-ARS	3.14
607	3	2	1	0	SSI-ARS	3.17
Pressurizer Spray & Relief						3.20 3.21
636	1	0	1	0	Incomplete	3.23
630	5	*	*		Incomplete	3.25
Residual Heat Removal						3.28 3.29
540	0	N/A	N/A	N/A		3.31
508	4	0	4	0	Incomplete	3.33
Service Water						3.36
465	4	4	0	0	SSI-ARS	3.38
Component Cooling						3.41
488/480	4	*	*		Incomplete	3.43
507/481	4	*	*		Incomplete	3.46
614	0	N/A	N/A			3.48
512	0	N/A	N/A			3.50
603A	1	0	0	0	Incomplete	3.52
766	2	*	*		Incomplete	3.54
605A	2	*	*		Incomplete	3.56

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
605B	0	N/A	N/A	N/A		3.58
509A	3	*	*		Incomplete	4.2
612	2	*	*		Incomplete	4.4
1512	0	N/A	N/A	N/A		4.6
2529	0	N/A	N/A	N/A		4.8
2526	0	N/A	N/A	N/A		4.10
2527	0	N/A	N/A	N/A		4.12
527A	2	*	*		Incomplete	4.14
517	4	*	*		Incomplete	4.16
603B	0	N/A	N/A	N/A		4.18
526A	0	N/A	N/A	N/A		4.20
526B						4.22
526C	0	N/A	N/A	N/A		4.24
527B	0	N/A	N/A	N/A		4.26
527D	0	N/A	N/A	N/A		4.28
509B	0	N/A	N/A	N/A		4.30
509C	0	N/A	N/A	N/A		4.32
509D	0	N/A	N/A	N/A		4.34
Containment Vacuum						4.37 4.38
CV1	1	1	0	0	SSI-ARS	4.40

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
3" HP Steam						4.43
746	1	*	*		Incomplete	4.45
Fire Protection						4.48
CF-1	0	N/A	N/A	N/A		4.50
CF-2	0	N/A	N/A	N/A		4.52
Diesel Muffler Exhaust						4.55 4.56
1040	0	N/A	N/A	N/A		4.58

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
OTHER PROBLEMS (HAND CALCULATIONS AND SHOCK0/1)						5.4
Low Head Safety Injection						5.7 5.8 5.9
1000A	0	N/A	N/A	N/A		5.11
1010A	0	N/A	N/A	N/A	SSI-ARS	5.13
1020A	0	N/A	N/A	N/A	SSI-ARS	5.15
1020B	0	N/A	N/A	N/A	SSI-ARS	5.17
1020C	0	N/A	N/A	N/A	SSI-ARS	5.19
Service Water						5.22
1030	4	4	0	0	SSI-ARS	5.24
Low Head Safety Injection						5.27 5.28 5.29
537	1	*	*		Incomplete	5.31
Containment & Recirculation & Spray						5.41 5.42 5.43
755	1	0	1	0	Incomplete	5.45
756	1	1	0	0	Original ARS	5.48
Aux. Feedwater						5.51
611	3	0	3	0	Incomplete	5.53
Residual Heat Removal						5.56 5.57
554	0	N/A	N/A	N/A	SSI-ARS	6.1

TABLE 3-2 (Cont)

System and Prob. No.	Total No. of Nozzles/ Penetrations	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
Component Cooling						6.4 6.5
606	0	N/A	N/A	N/A		6.7
613	3	*	*		Incomplete	6.10
502	4	*	*		Incomplete	6.12
506	5	*	*		Incomplete	6.14
Spent Fuel Cooling						6.17 6.18
747	4	*	*		Incomplete	6.20
748	2	*	*		Incomplete	6.23
749	2	*	*		Incomplete	6.25

*Stress analysis not complete; loads not available
N/A not applicable

SECTION 4

PIPE SUPPORT RESULTS

Table 4-1 summarizes the pipe supports evaluated in the reanalysis program. There are 846 supports on lines originally analyzed by SHOCK2; of these, 117, about 1/7, have been evaluated and found acceptable. A support is considered acceptable if all the reaction components are lower in magnitude than the reactions for which the support was originally designed. If some reaction component is greater than the original design reaction, the support is reanalyzed using the new reactions. Of the total SHOCK2 supports, 300 are being reevaluated at this time. An additional 429 supports are in problems for which stress analysis is not yet complete and hence for which support reactions are not available.

In cases where SSI/ARS was used, the DBE seismic inertial reactions on supports are multiplied by 1.5.

TABLE 4-1

PIPE SUPPORTS STATUS SUMMARY

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment
		SHOCK2 PROBLEMS			
Low Head Safety Injection					
555	14	5	9		
1555	9	5	4		
706A	14	2	12		
707A	8		8		
708	25	2	23		
731A	4	1	3		
731B	4	1	3		
743	5		5		
727	5				
High Head Safety Injection					
735	30				
Containment & Recirc. Spray					
1525A	16	11	5		
525A	15	13	2		
546/560	11		11		
546/5600	11		11		

TABLE 4-1 (Cont)

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment
546/5620	11		11		2.19
548C	11		11		2.21
547	12	3	9		2.23
744/754	5		5		2.25
548A	1	1	0	0	2.27
548B	22	1	21		2.29
544	12	6	6		2.31
544A	5	4	1		2.33
544B	5	3	2		2.35
751	0	0	0	N/A	2.37
562/546	3				2.39
745	8	1	7		2.41
Main <u>Steam</u>					2.43 2.44
323A	15	15	0	0	2.46
322A	3	3	0	0	2.48
334A	2	2	0	0	2.50
346	50				2.52
<u>Feedwater</u>					2.54
323B	10	10	0	0	2.56
322B	5	3	2		2.58
334B	3	3	0	0	3.2

TABLE 4-1 (Cont)

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
Aux. Feedwater						3.5
417	26	12	14			3.7
607	12	4	8			3.9
Pressurizer						3.12
Spray & Relief						3.13
636	29		29			3.15
630	21					3.17
Residual Heat						3.19
Removal						3.20
540	30		30			3.22
508	22		22			3.24
Service Water						3.26
465	4		4			3.28
Component Cooling						3.30
488/480	24					3.32
507/481	20					3.34
614	14					3.36
512	5					3.38
603A	4					3.40
766	5					3.42
605A	8					3.44
605B	7					3.46

TABLE 4-1 (Cont)

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
509A	21					3.48
612	3					3.50
1512	7					3.52
2529	42					3.54
2526	13					3.56
2527	29					3.58
527A	8					4.2
517	13					4.4
603B	4					4.6
526A	10					4.8
526B	9					4.10
526C	17					4.12
527B	11					4.14
527D	16					4.16
509B	9					4.18
509C	8					4.20
509D						4.22
Containment Vacuum						4.24 4.25
CV1	13		13			4.27
3" HP Steam						4.30
746	7					4.32

TABLE 4-1 (Cont)

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
Fire Protection						4.36
CF.1	5					4.38
CF.2	1					4.40
Diesel Muffler Exhaust						4.43 4.44
1040	15	6	9			4.46
OTHER PROBLEMS (HAND CALCULATIONS AND SHOCKO/1)						4.49
Low Head Safety Injection						4.52 4.53 4.54
1000A	4	2	2			4.56
1010A	2		2			4.58
1020A	8	7	1			5.2
1020B	8	6	2			5.4
1020C	11	6	5			5.6
Service Water						5.9
1030	4	4	0	0		5.11
Low Head Safety Injection						5.15
537	28		28			5.17
Containment & Recirculation Spray						5.21 5.22
755	3		3			5.24
756	3		3			5.26

TABLE 4-1 (Cont)

System and Prob. No.	Total No. of Supports	No. Acceptable Evaluation (Complete)	No. Under Evaluation	Modifications or Additions Required	Comment	
Aux. Feedwater						5.30
611	8		8			5.32
Residual Heat Removal						5.36 5.37
554	7	7				5.39
Component Cooling						5.42
606	3					5.44
613	10					5.46
502	16					5.48
506	17					5.50
Spent Fuel Cooling						5.52 5.53
747	4					5.55
748	6					5.57
749	6					6.1

SECTION 5 1.10

SCHEDULE FOR COMPLETION 1.12

5.1 PIPE STRESS AND SUPPORT REANALYSIS 1.15

All reanalysis of Unit 1 systems will be completed and reviewed by Engineering 1.16
Assurance personnel by October 1, 1979. This includes all pipe stress, 1.18
equipment nozzle, penetration, and pipe support evaluations. All required 1.19
modifications will be identified in advance of this date.

5.2 HARDWARE MODIFICATIONS 1.20

Stress analysis of a sample of safety-related piping has progressed to the 1.21
point where some stresses which would exceed code allowable levels have been 1.22
identified. Some have been resolved by the use of more detailed or refined 1.23
modeling techniques or by use of soil structure interaction amplified response 1.24
spectra (SSI-ARS). Others have been designated to be corrected by physical 1.25
hardware additions rather than pursue further analysis. Generally this is 1.26
done where the addition of restraints or damping devices (snubbers) would be
easier and less time-consuming than it would be to continue calculational 1.27

procedures. Most of these modifications are due to differences between 1.28
as-built and original design. In addition to the basic verification for 1.29
SHOCK2, new information is being incorporated in calculations to upgrade the 2.1
analyses where important changes to the input have occurred or where
additional data have been generated since the original analysis. This 2.3
produces some stresses that exceed code allowables. These cases are also 2.4
being corrected at this time even though they are not part of the Show Cause
Order.

Those hardware modifications now identified inside the containment in areas 2.5
which are not accessible because of radiation levels during plant operation 2.6
will be performed prior to startup of Unit 1. 2.7

Those hardware modifications outside the containment which are presently 2.8
identified will be performed within 30 days. Any additional modifications 2.10
outside the containment which may be determined in the process of completion
of the stress analysis will be completed within 30 days of the decision to 2.11
modify hardware.

If any further modifications are determined to be needed inside the 2.12
containment, a detailed evaluation of the severity of the condition will be 2.13

prepared and a proposal made to the NRC in order to determine whether shutdown 2.14
is warranted or whether modification at the next outage is reasonable.

This evaluation would consist for example of comparing calculated stresses 2.15
with real yield stress; determining whether a hanger would deflect within 2.16
acceptable limits or actually be damaged; and review for redundancy and/or 2.17
isolability of a line. Other aspects of an evaluation program will be 2.18
determined on a case-by-case basis.

Several modifications are currently identified even though the analysis and 2.19
evaluation stage is not yet complete. These modifications are described in 2.21
the following paragraphs for the problem identified. Because final design of 2.22
each of these modifications is incomplete, they are currently included under
the heading of "No. Under Evaluation" in Table 4-1. 2.23

Problem No. 743 - Low Head Safety Injection. In order to meet the allowable 2.25
stress for the DBE condition, a box-type restraint on this line requires
shimming to close an existing gap between the restraint and the pipe. This 2.27
shim is necessary for the pipe support to function as a lateral restraint as
well as a vertical restraint. The addition of this shim is considered to be a 2.28
very minor modification and its addition reflects the intent of the original 2.29
design. The analysis of the support itself is incomplete. 3.1

Problem 548A - Containment Recirculation Spray System. The field-verified, 3.3
as-built condition for this line differs from the condition used in the
original analysis. This problem was within code allowable stresses using the 3.4
SSI-ARS but exceeds the DBE allowable when the 1.5 magnification factor is 3.5
applied to the seismic inertial stress, per the NRC letter of May 25, 1979.
This condition will require the addition of a seismic snubber to the piping 3.6
system. This modification is considered minor and is attributable to the 3.7
difference between the as-built and the original design conditions. The 3.9
design of the snubber is incomplete.

Problems 731A and 731B Low Head Safety Injection. These two problems each 3.11
contain a box type support. Based on preliminary analysis using the original 3.12
FSAR ARS, these pipe supports will deflect excessively under DBE loading 3.13
conditions. Although these supports have not been reanalyzed using the SSI- 3.14
ARS loads, it is evident that these supports will require a structural brace 3.15
to be added. The design of the required modification is incomplete. 3.16

5.3 REVIEW OF SHOCK1 PROGRAM 3.19

Twelve Surry 1 pipe stress problems were computed by versions of the program 3.20
PSTRESS/SHOCK1. This program performed intermodal combination by the so- 3.23
called "Navy method," which consists of the absolute sum of the largest 3.24

magnitude modal response and the square root of the sum of the squares of the remaining modes. Intra-modal responses due to multi-directional earthquake excitation were not calculated because SHOCK1 only produced responses parallel to a given earthquake component excitation (i.e., the responses were considered uncoupled). For this reason, the SHOCK1 code is not considered consistent with current analysis techniques. The Navy method, being more conservative than a straightforward square root of the sum of the squares of the modal responses, generally provides more than adequate conservatism. Comparative results calculated for Maine Yankee Atomic Power Station[1] have led to the conclusion that SHOCK1 is suitably conservative to ensure that the piping systems meet the allowable stress levels.

A listing of the latest version of the SHOCK1 program was sent to the NRC (letter from S&W to Mr. Denton, NRC, dated April 6, 1979). However, no safety systems at Surry 1 are known to have been analyzed using this version of the program, and so no verification of this program for Surry 1 has been done. The 12 problems cited were analyzed using earlier versions of the program, now called SHOCK0, for which no listings are now available. Comparative analyses given in Appendix E, using the NUPIPE program together with similar studies made for the Maine Yankee Atomic Power Station, show that SHOCK0 produced stress results comparable to accepted programs and provides assurance that the FSAR criteria are met. The studies and reanalyses performed to date reaffirm

that the Seismic Category I piping is conservatively designed to withstand the 4.18
effects of the design basis earthquake. 4.19

Four of the 12 pipe stress problems originally calculated by SHOCKO/SHOCK1 4.20
have been recomputed using NUPIPE and field-verified, as-built conditions. 4.21
These four problems, numbers 755, 756, 537, and 554, are summarized in 4.22
Table 3-1 given earlier. The SSI-ARS was used only on Problem No. 554, and 4.24
the seismic inertial stress was multipled by 1.5. In all four cases the 4.25
computed maximum total stress is less than the allowable stress and in two
cases very much less than the allowable stress. Also, for comparison 4.27
purposes, two additional SHOCKI/SHOCKO stress problems were rerun on NUPIPE
without the use of field-verified data; consequently these do not represent 4.28
the results from the Surry 1 reanalysis effort. The mass models used in these 5.1
studies were the same as those used originally. Several default program 5.2
values, particularly assumed support stiffnesses, were different in the NUPIPE 5.3
reruns so that the mathematical models were not identical, although they are 5.4
as similar as would be expected in normal production work. It is believed 5.6
that these comparative examples fairly illustrate the results that would be
obtained by production stress analysts using the two programs. In addition to 5.8
the two Surry 1 examples given here, the Maine Yankee Atomic Power Station[1]
verified the SHOCK1 program against NUPIPE and compared 10 SHOCKO problems 5.9
with SHOCK1 runs, and 3 SHOCKO problems with NUPIPE runs, with essentially 5.10

similar results. Appendix E gives a description of each of the two Surry 5.11
problems, the respective isometric drawings, stress plots, and stress 5.12
comparisons by node point.

Because the SHOCKO program is not equivalent to current practice, the Virginia 5.13
Electric and Power Company has decided that all Surry 1 pipe stress problems 5.14
originally analyzed by PSTRESS/SHOCKO or SHOCK1 will be reanalyzed using the 5.15
benchmarked accepted program NUPIPE, using field-verified data. In view of 5.16
the conservative nature of the SHOCKO stresses as determined by the
comparative analyses, and in view of the determination by the NRC Staff in the 5.17
matter of Maine Yankee Atomic Power Station[2], Docket No. 50-309, that 5.18
SHOCKO/SHOCK1 did not use the algebraic summation method, reanalysis of the
12 SHOCKO/SHOCK1 problems will be given a lower priority than the reanalysis 5.19
of the SHOCK2 problems. The Virginia Electric and Power Company commits to 5.20
reanalyze all the remaining SHOCKO problems and to provide the results to the 5.21
NRC Staff by October 1, 1979.

[1] S&W Report, "Verification of SHOCK1 Program for Maine Yankee Atomic Power 5.24
Station," April 19, 1979.

(2) NRC "Safety Evaluation by the Office of Nuclear Reactor Regulation" 5.25
May 24, 1979

SECTION 6 1.10

HIGH ENERGY LINE BREAKS 1.12

Of the high energy lines addressed in Appendix D of the FSAR, only the main 1.15
steam lines outside the containment are included in this stress reanalysis. 1.16

Each of the main steam lines has two terminal break locations, one at the 1.17
containment penetration and the other at the main steam manifold. These 1.19
terminal breakpoints are predetermined and are not changed as a result of the
stress reanalysis.

Two intermediate break locations were originally determined based upon maximum 1.20
primary plus secondary stresses. Upon initial inspection, it is felt that the 1.21
reanalysis will not significantly affect their location.

For piping downstream of the main steam manifold, no seismic analysis is 1.23
required.

In summary, the reanalysis will not create any appreciable change to 1.24
Appendix D of the FSAR.

This will be verified with main steam line calculations after the stress 1.25
reanalysis is completed and the highest stress points reviewed by the same 1.26
procedure used in the original break analysis. If any break location changes 1.27
are noted, they will be incorporated into the existing station inspection 1.28
program within 90 days.

SECTION 7

CONSERVATISMS

There are many conservatisms inherent in the seismic design of the Surry Power Station, Unit 1. Some of these are:

- Elastic dynamic analyses are performed using conservatively low damping values. This point is discussed in greater detail in Appendix F.

- Multiple-directional seismic input, with each horizontal component having equal intensity, is considered in design of plants. Actual earthquakes are typically stronger in one direction.

- In the design of structures and equipment, it is convenient to assure that all elements of the structure or equipment are designed to stress levels well below the actual strength of the materials, so that any permanent deformation is very small. This approach obviates the need for complex and costly inelastic analyses. Inelastic behavior would significantly reduce structural response prior to failure. From the standpoint of functionability, piping systems and

components can withstand the deformation and usually even the failure 1.29
of some supports due to redundancy, i.e., the existence of multiple 2.1
load paths.

- Stress limits, whether elastic or inelastic, are based upon material 2.2
behavior under static loading conditions. Since dynamic loads 2.3
contain a limited amount of energy, the margin (between the stress
limits and failure) under dynamic loads is greater than under static 2.4
loads if elastically calculated peak response is compared to the
stress limits with strain rate effects neglected. 2.5

- Pipe and structural support members are selected from standard 2.6
available sections, and consequently have generally greater strength 2.7
than the minimum requirements by the analysis. Hence, the computed 2.9
stresses are often lower than the allowable stresses. This 2.10
difference constitutes an additional margin in the actual plant.

- Computed pipe stresses are magnified by code stress intensification 2.11
factors. As discussed in Appendix F, these stress intensification 2.12
factors are primarily fatigue factors, and are not strictly 2.13
applicable to the seismic DBE condition.

- Piping in general is conservatively designed, even when no dynamic 2.14
seismic analysis is performed. Fossil-fueled power plants, 2.16
refineries, and process plants have survived major earthquakes in 2.17
California, Alaska, Guatemala, and other locations with little or no
piping damage. This experience includes earthquakes considerably 2.18
larger than the DBE for Surry Unit 1. The experience with piping 2.19
performance in earthquakes is reviewed in detail in a report included 2.20
here as Appendix F.

In addition to the conservatisms listed above, which are inherent in any 2.22
design of nuclear facilities, there are additional conservatisms specific to 2.23
the Surry units. These conservatisms are not theoretical concepts, but indeed 2.24
are real and existing margins of safety. To quantify these conservatisms is 2.26
difficult, but this in no way negates the sound conservative premise on which 2.27
the Surry reanalysis effort is based. These additional conservatisms are 2.28
discussed below.

7.1 STRESS LIMITS 3.9

The analyses and reanalyses of Seismic Category I piping systems are based 3.10
upon the conservative stress limit of $1.8S_h$ under the limiting faulted or DBE 3.11
loading conditions. The present ASME Section III Code specifies the piping 3.13

stress limit to be $2.4S_h$ under the Faulted DBE Condition. In July 1978, the 3.15
NUREG/CR-0261 report* used the limit moment theory to address the Code rules,
and it was established that gross plastic deformation may occur when primary 3.16
stress exceeds 1.5 to 2.0 times the yield strength (S_y) of piping material, 3.17
but for stresses below these values, functional capability was maintained. 3.18
For Surry Power Station, Unit 1, the majority of carbon steel piping material 3.19
is of SA-106 Grade B steel. Using the lower limit of $1.5S_y$ from NUREG/CR- 3.21
0261 and representative properties of SA-106 Grade B steel, the added margin 3.22
of conservatism is the ratio ($1.5S_y/1.8S_h$), which ranges from 1.4 at 650°F to
1.94 at 100°F.

The Surry 1 reanalysis calculations have included the seismic stress due to 3.27
anchor displacements in the DBE condition. Inclusion of the anchor movement 3.29
stresses was not required by ANSI Code B31.1, used for the original design, 4.1
and is not required by current 1979 codes, for the faulted DBE condition.
Addition of this stress component is a significant conservatism. 4.2

* E.C. Rodabaugh and S.E. Moore, "Evaluation of the Plastic Characteristics of Piping Products in Relation to Code Criteria," NUREG/CR-0261, July 1978

7.2 SYSTEM REDUNDANCIES 4.4

All systems essential for shutdown or for mitigation of a design basis 4.5
accident (DBA) contain redundant flow paths and driving equipment. These 4.8
redundant paths and equipment are analyzed as separate problems with complete
input verification, engineering analysis, and engineering review. Therefore, 4.10
a common mode problem is precluded. Even identically designed redundant 4.11
systems may not always experience similar seismic excitation due to different
mounting locations with structural filtering effects. Thus, even a postulated 4.13
loss of a redundant component will not mean a loss of function for the system.

7.3 SAFETY SYSTEMS 4.15

All safety systems, not just a sampling or a portion of redundant systems, 4.16
which were originally seismically analyzed with the SHOCK2 program have been 4.17
included in this reanalysis. The systems involving the check valves whose 4.18
weight has been revised (see Appendix D) have also been completed. Therefore 4.19
pipe stress reanalysis has been completed for the systems in this reanalysis
program that interface with the reactor coolant system. The auxiliary feed 4.21
system and the steam supply to the steam driven auxilary feed pump have also
been reanalyzed.

The new weights for safety injection accumulator discharge motor-operated 4.23
valves supplied by Velan have ben reviewed. The problems were rerun with 4.25
higher weights and no additional changes to stresses resulted.

7.4 FIELD VERIFICATION OF AS-BUILT CONDITIONS 4.27

To ensure that the pipe stress reanalysis is performed as accurately as 4.28
possible, field verification of as-built conditions has been performed. The 5.2
field verification produced detailed piping isometric drawings upon which
reanalysis is based. This confirmation of input data provides assurance that 5.3
analytical results are correct. All field-verified piping isometrics are 5.4
independently verified by Surry Power Station quality control personnel. 5.5
Licensee Event Reports were filed when discrepancies were found and these 5.6
discrepancies are being corrected. See Section 8. 5.7

7.5 QUALITY ASSURANCE/ENGINEERING ASSURANCE 5.9

A comprehensive and extensive Quality Assurance program has been developed and 5.11
applied to the reanalysis activities. The unusual nature of the evaluation 5.13
and analysis program required the development of new technical procedures for 5.14
use in performing evaluation and analysis work. A detailed project procedure 5.15
was developed that includes provisions for design control, document control, 5.16

and interface controls. Each new procedure developed received a full review 5.17
by the S&W Engineering Assurance (EA) staff; project procedures required 5.18
approval by EA.

The normal Quality Assurance auditing activity has been extensively expanded 5.19
to provide the highest possible degree of confidence in the reanalysis work. 5.20
Instead of auditing on a sampling basis, each pipe stress/pipe support problem 5.21
package is subjected to a detailed EA inspection according to a written EA 5.22
inspection plan. The EA inspection plan is sufficiently broad in scope to 5.23
include all significant technical attributes in addition to the usual 5.24
programmatic attributes inspected in audit plans. Each problem package is 5.26
inspected at the completion of the stress evaluation and again at the 5.27
completion of support/nozzle/equipment evaluations. This type of inspection 5.28
activity, by increasing the confidence which can be placed on task group
output, is considered to confirm the high degree of conservatism inherent in 5.29
the engineering work.

In addition to the stress and support package inspections, EA has confirmed 6.1
the quality of key input information used in the stress/support evaluation. 6.2
The as-built documentation effort has been extensively audited by S&W EA and 6.3
VEPCO QC. Each as-built document created was subjected to this joint S&W 6.4
EA/VEPCO QC inspection, conducted at the plant site, to confirm the actual as- 6.5

built configuration. Finally, the development of the SSI-ARS will be 6.7
subjected to a comprehensive, technical audit. The effort to confirm the 6.8
accuracy of the as-built configuration and the amplified response spectra adds 6.9
additional confidence to the quality of the reanalysis effort.

7.6 USE OF AMPLIFIED RESPONSE SPECTRA 6.11

The peaks in amplified floor response spectra are broadened to account for 6.13
variation in material properties and approximations in modeling. Peak 6.15
broadening is intended to reflect a range of uncertainty in the precise
location of the resonant peak of the response curve and not to indicate that 6.16
multiple peak resonant response is likely within the broadened range. What, 6.18
in fact, exists is a "family" of resonant curves, each having only one point
of maximum resonant response. 6.19

It would be more precise to analyze systems and components for a number of 6.20
unbroadened spectra which are members of the broadened family of possible 6.21
amplified response spectra. Since there can only be one single peak frequency 6.22
for a given system, the use of peak broadened floor response spectra as 6.23
practiced in seismic design is a conservative analytical expediency that 6.24
results in an additional margin of safety for systems, components, and
supports.

In addition, when a piping system has multiple supports, the system is 6.25
analyzed with a response spectrum that envelops the spectrum at each support 6.26
that results in an additional margin of safety for systems, components, and 6.27
supports.

Since pipe runs generally extend over a range of elevations from beginning to 6.28
end of run, and since the magnitude of acceleration associated with each ARS 6.29
increases with elevations in a structure, the ARS applied in the analysis of
each run is selected coincident with the higher elevations along each piping 7.1
run.

7.7 DEVELOPMENT OF SOIL-STRUCTURE INTERACTION AMPLIFIED RESPONSE 7.4
SPECTRA (SSI-ARS) 7.5

Reevaluation of piping systems for induced earthquake loading has employed 7.8
conservatisms in the development of ARS based on SSI and the application of 7.11
resulting loads to the pipe stress analysis. These conservatisms involve the 7.13
methodology of developing ARS, the range in soil property variations in 7.14
developing amplified spectra, the application of particular ARS to each piping 7.15
run, the effects of three dimensional input, and an increase (bumping) in the 7.16
inertia forces applied to each pipe run after computer calculation of stress
and support loads.

The methodology used in ARS based on SSI is based upon a layered elastic media model for soil and a lumped mass model for the structure. Analysis using these models involves (a) the calculation of frequency-dependent stiffness at the surface of a layered medium using the program REFUND, (b) modification of a specified surface motion to account for embedment of the structure, (c) the application of kinematic interaction principles to modify translation input specified at the surface to both a translation and rotational motion at the base of the rigid structure foundation using the program KINACT, and (d) analysis of the structural model supported on frequency-dependent springs using the program FRIDAY. The resulting ARS developed from this methodology were compared with ARS developed using a detailed finite element representation of the underlying soil medium with a lumped mass representation of the containment structure using the program PLAXLY. The amplified values of acceleration computed using the REFUND/KINACT/FRIDAY method are generally 30 to 100 percent larger than values computed using the more rigorous PLAXLY approach.

Variations in soil properties have generally been accounted for by developing ARS using mean values of soil moduli and damping ratio values adjusted for strain levels associated with earthquakes, and peak spreading the resulting ARS.

Substantial conservatism is also gained by the application of a 50 percent 8.9
increase in the seismic inertia forces acting on each piping system above 8.10
those forces computed using the ARS developed from the REFUND/KINACT/FRIDAY 8.11
approach. The 50 percent increase in inertial seismic forces is a requirement 8.12
of the NRC letter of May 25, 1979. The 50 percent increase accounts for a 8.13
wide range of possible input parameters and, in fact, adds again conservatisms 8.14
already accounted for by using peak spreading techniques.

Substantial elaboration of the techniques described in this section on the use 8.15
of soil structure interaction techniques will be documented in a subsequent 8.16
submittal to the NRC staff on or about June 8, 1979.

7.8 SEISMIC EVENT PROBABILITY 8.18

The seismic hazard at the Surry Power Station site is small. Calculations 8.20
incorporating the effects of potential seismic zones far from the site as well 8.22
as random local seismicity indicate that the annual risk of equaling or
exceeding the DBE of 0.15 g is about 1.1×10^{-5}, which corresponds to a hazard 8.23
of 4.5×10^{-4} over the 40 year life of the plant. 8.24

During any one month, there is a hazard of 4.5×10^{-4} of equaling or exceeding 8.25
an earthquake with a peak acceleration of 0.04 g. Thus, the chances are very 8.27

slight that the plant will experience any significant shaking due to an 8.28
earthquake during the period that systems are being checked and any corrective
action taken. The chances of experiencing the DBE are extremely small. 8.29

The calculations leading to these conclusions use methods developed by 9.1
Prof. C. Allin Cornell of M.I.T. and Dr. Robin K. McGuire of the U.S. 9.2
Geological Survey. The seismicity data are derived from historical records 9.3
and are developed into seismic zones by the methods of Dr. Edward Chiburis of 9.4
Weston Observatory. The analytical techniques explicitly account for 9.5
randomness of occurrence of earthquakes in space and time and for uncertainty 9.6
in attenuation relations.

SECTION 8

LICENSEE EVENT REPORTS AND RESOLUTIONS

The pipe supports addressed in Licensee Event Report LER79-010/03L-0 will be shimmed or restored to meet the requirements of the stress analyses performed on the various systems. Final as-built verifications will be performed to demonstrate correspondence of the stress analysis models with hardware and system arrangements prior to startup.

Licensee Event Report LER79-004/01T-0 describes the computer code nonconservatism which is being restored by this report. The variance in valve weights from those originally given by the valve vendor are being included in this effort also. (These valve weights are also the subject of IE Bulletin 79-04.)

APPENDIX A 1.9

SYSTEMS AFFECTED 1.11

The reanalysis included those safety-related lines originally computer-analyzed with the SHOCK2 program. The systems line numbers, the associated computer problem numbers, and the flow diagram numbers are listed below.

The figure numbers refer to the FSAR drawings, and the Surry, Unit 1, FM and FB drawings included in Appendix B.

System	Line No.	Problem No.	Figure No.
Pressurizer	4"-RC-15-1502	636	4.2.1-1
Spray & Relief	4"-RC-14-1502		4.2.1-2
	12"-RC-36-602	630	4.2.1-2
	6"-RC-39-1502		
	6"-RC-42-602		
	6"-RC-38-1502		
	6"-RC-41-602		
	6"-RC-37-1502		
	6"-RC-40-602		
	4"-RC-34-1502		
	3"-RC-35-1502		
	6"-RC-20-602		
	3"-RC-61-1502		
	6"-RC-62-602		
Low Head Safety	12"-RC-23-1502	555	4.2.1-1
Injection	12"-SI-46-1502		6.2.2.1-2
	12"-RC-22-1502	1555	4.2.1-1
	12"-SI-45-1502		6.2.2.1-2
	6"-RC-16-1502	706A	4.2.1-1
	6"-SI-49-1502		6.2.2.1-2
	6"-RC-21-1502	707A	4.2.1-1
	6"-SI-50-1502		6.2.2.1-2
	6"-RC-18-1502	708	4.2.1-1
	6"-SI-48-1502		6.2.2.1-2
	6"-SI-143-1502		
	6"-SI-49-1502		
	6"-SI-50-1502		
	8"-SI-92-153	731A,B	6.2.2.1-1
	8"-SI-14-153		
	10"-SI-6-153	743	6.2.2.1-1

System	Line No.	Problem No.	Figure No.	
	6"-RC-17-1502	1000A	4.2.1-1	2.11
	2"-SI-74-1502		6.2.2.1-2	2.12
	6"-SI-145-1502			2.13
	6"-SI-48-1502	727	6.2.2.1-1	2.16
	6"-SI-49-1502			2.17
	6"-SI-153-1502			2.18
	10"-SI-148-153			2.19
	10"-SI-149-153			2.20
	10"-SI-150-153			2.21
	10"-SI-151-153			2.22
	10"-SI-16-153			2.23
	10"-SI-13-153			2.24
	6"-RC-19-1502	1010A	4.2.1-1	2.26
	2"-SI-85-1502		6.2.2.1-2	2.27
	6"-SI-144-1502			2.28
	6"-RC-20-1502	1020A,B,C	4.2.1-1	2.30
	2"-SI-75-1502		6.2.2.1-2	2.31
	6"-SI-153-1502			2.32
	6"-SI-15-1502			2.33
	6"-SI-145-1502			2.34
High Head Safety Injection	8"-SI-7-152	735	6.2.2.1-1	2.36
	8"-SI-102-152			2.37
	10"-SI-6-153			2.38
	8"-SI-17-152			2.39
	6"-SI-18-152			2.40
	6"-SI-19-152			2.41
	6"-SI-78-152			2.42
	6"-CH-201-152			2.43
	6"-CH-202-152			2.44
	6"-CH-203-152			2.45
	8"-CH-204-152			2.46
	8"-CH-206-152			2.47
	8"-CH-17-152			2.48
	6"-CH-72-152			2.49
	6"-CH-18-152			2.50
	6"-CH-19-152			2.51
	4"-CH-112-152			2.52
Containment and Recirculation Spray	8"-CS-23-153	525A	FM-101A	2.54
	8"-CS-22-153			2.55
	10"-CS-3-153			2.56
	10"-CS-4-153			2.57
	10"-CS-4-153	547	FM-101A	3.1

System	Line No.	Problem No.	Figure No.	
	8"-CS-34-153			3.2
	8"-RS-21-153	546/560	FM-101A	3.4
	10"-RS-4-153			3.5
	8"-CS-34-153	744(754)	FM-101A	3.7
				3.8
				3.9
	10"-CS-3-153	548A,B	FM-101A	3.11
	8"-CS-33-153			3.12
	10"-RS-3-153			3.13
	10"-RS-10-153	544, 544A,B	FM-101A	3.15
	10"-RS-9-153			3.16
	10"-RS-1-153			3.17
	10"-RS-2-153			3.18
	10"-RS-10-153	751	FM-101A	3.20
	10"-RS-9-153			3.21
	10"-RS-12-153	562	FM-101A	3.23
	8"-RS-23-153			3.24
	8"-CS-33-153	745	FM-101A	3.26
	10"-RS-3-153	546/5600	FM-101A	3.27
	8"-RS-20-153			3.28
	10"-RS-11-153	546/5620	FM-101A	3.30
	8"-RS-22-153			3.31
	10"-RS-11-153	548C	FM-101A	3.33
Residual	4"-RC-197-153	540	4.2.1-2	3.37
Heat	4"-RH-15-152		9.3-1	3.38
Removal	12"-RH-19-602			3.39
	10"-RH-16-1502			3.41
	3"-RH-13-602			3.42
	6"-RH-14-602			3.43
	10"-RH-23-602			3.44
	10"-RH-17-1502			3.45
	14"-RH-1-1502	508	9.3-1	3.47
	14"-RH-18-602		4.2.1-1	3.48
	14"-RH-2-602			3.49
	10"-RH-4-602			3.50
	10"-RH-5-602			3.51

System	Line No.	Problem No.	Figure No.	
	12"-RH-6-602			3.52
	12"-RH-12-602			3.53
	10"-RH-8-602			3.54
	10"-RH-9-602			3.55
	12"-RH-19-602			3.56
	10"-RH-7-602			3.57
	10"-RH-10-602			3.58
Component Cooling Water	18"-CC-227-121	488/480	9.4-4	4.3
	18"-CC-228-121			4.4
	18"-CC-229-121			4.5
	18"-CC-230-121			4.6
	18"-CC-5-121			4.7
	18"-CC-220-121			4.8
	18"-CC-6-121			4.9
	18"-CC-237-121	507/481	9.4-4	4.12
	18"-CC-236-121			4.13
	18"-CC-235-121			4.14
	8"-CC-311-151			4.15
	18"-CC-225-121			4.16
	18"-CC-232-121			4.17
	18"-CC-233-121			4.18
	18"-CC-234-121			4.19
	18"-CC-231-121			4.20
	18"-CC-7-121	614	9.4-4	4.22
			9.4-3	4.23
	12"-CC-27-121		9.4-6	4.24
	18"-CC-16-121	1512	9.4-1	4.26
			9.4-4	4.27
	18"-CC-15-121	512	9.4-1	4.29
			9.4-4	4.30
			9.4-6	4.31
	18"-CC-16-121	603A,B	9.4-1	4.33
	18"-CC-8-121			4.34
	8"-CC-69-151	766	9.4-3	4.36
	8"-CC-67-151			4.37

System	Line No.	Problem No.	Figure No.	
	8"-CC-61-151	527A,B,D	9.4-3	4.40
	4"-CC-61-151			4.41
	6"-CC-62-151			4.42
	8"-CC-75-131			4.43
	6"-CC-81-121			4.44
	10"-CC-81-121			4.45
	14"-CC-67-121			4.46
	18"-CC-10-121			4.47
	18"-CC-10-121	605A,B	9.4-1	4.50
	6"-CC-105-151			4.51
	3"-CC-107-151			4.52
	3"-CC-112-151			4.53
	6"-CC-207-151			4.54
	18"-CC-17-121			4.55
	4"-CC-108-151			4.56
	4"-CC-113-151			4.57
	18"-CC-8-121	509A,B,C,D	9.4-1	5.1
	18"-CC-237-121		9.4-2	5.2
	18"-CC-236-121		9.4-4	5.3
	18"-CC-235-121			5.4
	8"-CC-311-151			5.5
	8"-CC-32-151			5.6
	6"-CC-32-151			5.7
	4"-CC-32-151			5.8
	1 1/2"-CC-30-151			5.9
	6"-CC-286-151			5.10
	24"-CC-235-121			5.11
	18"-CC-226-121			5.12
	18"-CC-10-121			5.13
	12"-CC-27-121			5.14
	24"-CC-226-121			5.15
	18"-CC-9-121			5.16
	10"-CC-89-151			5.17
	6"-CC-89-151			5.18
	6"-CC-97-151			5.19
	18"-CC-14-121	612	9.4-4	5.21
			9.4-3	5.22
	14"-CC-72-121			5.23
	10"-CC-72-121			5.24
	4"-CC-148-151			5.25
	6"-CC-146-151			5.26
	18"-CC-14-121	517	9.4-3	5.28
	14"-CC-72-121		9.4-5	5.29
	14"-CC-70-121		9.4-6	5.30

System	Line No.	Problem No.	Figure No.	
	18"-CC-70-151			5.31
	4"-CC-66-151			5.32
	8"-CC-66-151			5.33
	6"-CC-64-151			5.34
	10"-CC-72-121			5.35
	18"-CC-17-151			5.36
	8"-CC-70-151			5.37
	8"-CC-71-151			5.38
	8"-CC-78-151	526A,B,C	9.4-2	5.41
	18"-CC-17-121			5.42
	6"-CC-85-151			5.43
	6"-CC-93-151			5.44
	10"-CC-104-121			5.45
	6"-CC-101-151			5.46
	4"-CC-144-151	2527	9.4-5	5.49
	6"-CC-145-151			5.50
	8"-CC-144-151			5.51
	10"-CC-143-151			5.52
	6"-CC-181-151	2526	9.4-6	5.54
	2 1/2"-CC-184-151			5.55
	6"-CC-173-151			5.56
	10"-CC-181-151			5.57
	6"-CC-165-151			5.58
	8"-CC-156-151			6.1
	18"-CC-14-121			6.2
	3"-CC-127-151	2529	9.4-3	6.4
	14"-CC-143-121		9.4-5	6.5
	18"-CC-7-121			6.6
	10"-CC-161-121			6.7
	6"-CC-161-151			6.8
	8"-CC-153-151			6.9
Service Water	24"-WS-33-10	1030	9.9-1	6.12
	24"-WS-34-10			6.13
	24"-WS-35-10			6.14
	24"-WS-36-10			6.15
	24"-WS-26-10	465	9.9-1	6.18
	24"-WS-28-10			6.19
	24"-WS-30-10			6.20
	24"-WS-32-10			6.21
Main Steam	30"-SHP-1-601	323A	10.3-1	6.23

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System	Line No.	Problem No.	Figure No.	
	30"-SHP-2-601	322A	10.3-1	6.25
	30"-SHP-3-601	334A	10.3-1	6.27
	30"-SHP-1-601	346	10.3-1	6.29
	30"-SHP-2-601			6.30
	30"-SHP-3-601			6.31
High Pressure Steam	4"-SHP-25-601	746	10.3-1	6.34
	4"-SHP-26-601			6.35
	4"-SHP-27-601			6.36
	3"-SHP-28-601			6.37
	3"-SHP-29-601			6.38
	3"-SHP-30-601			6.39
	3"-SHP-32-601			6.40
	3"-SHP-33-601			6.41
	3"-SHP-34-601			6.42
	3"-SHP-35-601			6.43
Feedwater	14"-WFPD-17-601	323B	10.3.5-2	6.47
	14"-WFPD-13-601	322B	10.3.5-2	6.49
	14"-WFPD-9-601	334B	10.3.5-2	6.51
Auxiliary Feedwater	3"-WAPD-9-601	417	10.3.5-2	6.54
	3"-WAPD-11-601			6.55
	3"-WAPD-13-601			6.56
	3"-WAPD-10-601			6.57
	3"-WAPD-12-601			6.58
	3"-WAPD-14-601			7.1
	6"-WAPD-1-601			7.2
	6"-WAPD-2-601			7.3
	6"-WAPD-1-601	607	10.3.5-2	7.10
	6"-WAPD-2-601			7.11
	6"-WAPD-3-601			7.12
	6"-WAPD-4-601			7.13
	4"-WAPD-5-601			7.14
	4"-WAPD-6-601			7.15
	4"-WAPD-7-601			7.16
	4"-WAPD-8-601			7.17
Containment Vacuum	8"-CV-8-151	CV-1	FM-102A	7.19
				7.20
Fire Protection	See Figure	CF-1	FB-3D	7.22
	See Figure	CF-2		7.23

Diesel Muffler	See Figure	1040	FB-25L	7.25
Exhaust				7.26

In addition to the SHOCK2 computer calculations, the above list includes 7.30
five hand calculations which were reanalyzed because of incorrect VELAN check 7.31
valve weights (IE Bulletin No. 79-04). 7.32

Nonsafety-related SHOCK2 problems were not reanalyzed, except for the fire 7.34
protection and diesel muffler exhaust.

The following is a listing of safety-related lines that were analyzed with the 7.35
SHOCKO/SHOCK1 programs. These problems are also identified on the flow 7.37
diagrams in Appendix B.

System	Line No.	Problem No.	Figure No.	7.40 7.41
Low Head Safety	12"-SI-47-1502	537	4.2.1-1	7.44
Injection	12"-RC-24-1502		6.2.2.1-2	7.45
Containment and	12"-CS-1-153	755	FM-101A	7.49
Recirculation	12"-CS-2-153	756	FM-101A	7.50
Spray				7.51
Residual	6"-RH-20-152	554	9.3-1	7.55
Heat				7.56
Removal				7.57
Component	12"-CC-33-121	606	9.4-3	8.3
Cooling	12"-CC-34-121			8.4
Water				8.5
	4"-CC-37-151	613	9.4-4	8.8
	3"-CC-38-151			8.9
	6"-CC-37-151			8.10
	6"-CC-287-151			8.11
	24 & 18"-CC-225-121	481	9.4-4	8.13
	18"-CC-232-121			8.14
	24 & 18"-CC-233-121			8.15
	18"-CC-234-121			8.16
	18"-CC-1-121	502	9.4-4	8.18
	18"-CC-2-121			8.19
	18"-CC-3-121			8.20
	18"-CC-4-121			8.21
	18"-CC-223-121			8.22
	18"-CC-227-121			8.23
	18"-CC-228-121			8.24
	18"-CC-229-121			8.25
	8"-CC-314-151			8.26

	8"-CC-287-151			8.27
	18"-CC-14-121	506	9.4-4	8.37
	18"-CC-15-121			8.38
	18"-CC-16-121			8.39
	18"-CC-17-121			8.40
	18"-CC-19-121			8.41
	6"-CC-20-151	539	9.4-4	8.43
	6"-CC-22-151			8.44
	6"-CC-222-151			8.45
	6"-CC-21-151			8.46
Fuel Pit Cooling	12"-FP-4-152	747	9.5-1	8.49
	12"-FP-5-152			8.50
	12"-FP-3-152			8.51
	12"-FP-32-152	748	9.5-1	8.54
	12"-FP-33-152			8.55
	16"-FP-18-152			8.56
	12"-FP-1-152	749	9.5-1	8.58
	12"-FP-2-152			9.1
Auxiliary Feedwater	6"-WCMU-5-151	611	10.3.5-2	9.4
	6"-WCMU-6-151			9.5
	6"-WCMU-7-151			9.6
	4"-WCMU-9-151			9.7
	4"-WCMU-10-151			9.8
	6"-WCMU-11-151			9.9
	6"-WCMU-8-151			9.10

APPENDIX B 1.10

FLOW DIAGRAMS - 1.12
IDENTIFICATION OF PROBLEMS 1.13
REANALYZED 1.14

SURRY POWER STATION, UNIT 1

FLOW DIAGRAMS 1.10

Title	Figure No.	1.13
Reactor Coolant - Sheet 1	4.2.1-1	1.16
Reactor Coolant - Sheet 2	4.2.1-2	1.17
Safety Injection - Sheet 1	6.2.2.1-1	1.18
Safety Injection - Sheet 2	6.2.2.1-2	1.19
Chemical and Volume Control	9.1-2	1.20
Containment & Recirculation Spray	11448-FM-101A	1.21
Residual Heat Removal	9.3-1	1.22
Component Cooling - Sheet 1	9.4-1	1.23
Component Cooling - Sheet 2	9.4-2	1.24
Component Cooling - Sheet 3	9.4-3	1.25
Component Cooling - Sheet 4	9.4-4	1.26
Component Cooling - Sheet 5	9.4-5	1.27
Component Cooling - Sheet 6	9.4-6	1.28
Fuel Pit Cooling	9.5-1	1.29
Service Water	9.9-1	1.30
Main Steam	10.3-1	1.31
Feedwater	10.3.5-2	1.32
Containment Vacuum	11448-FM-102A	1.33
Fire Protection	11448-FB-3D	1.34
Diesel Muffler Exhaust	11448-FB-25L	1.35

FIG. 4.2.1-1
OCT. 15, 1970

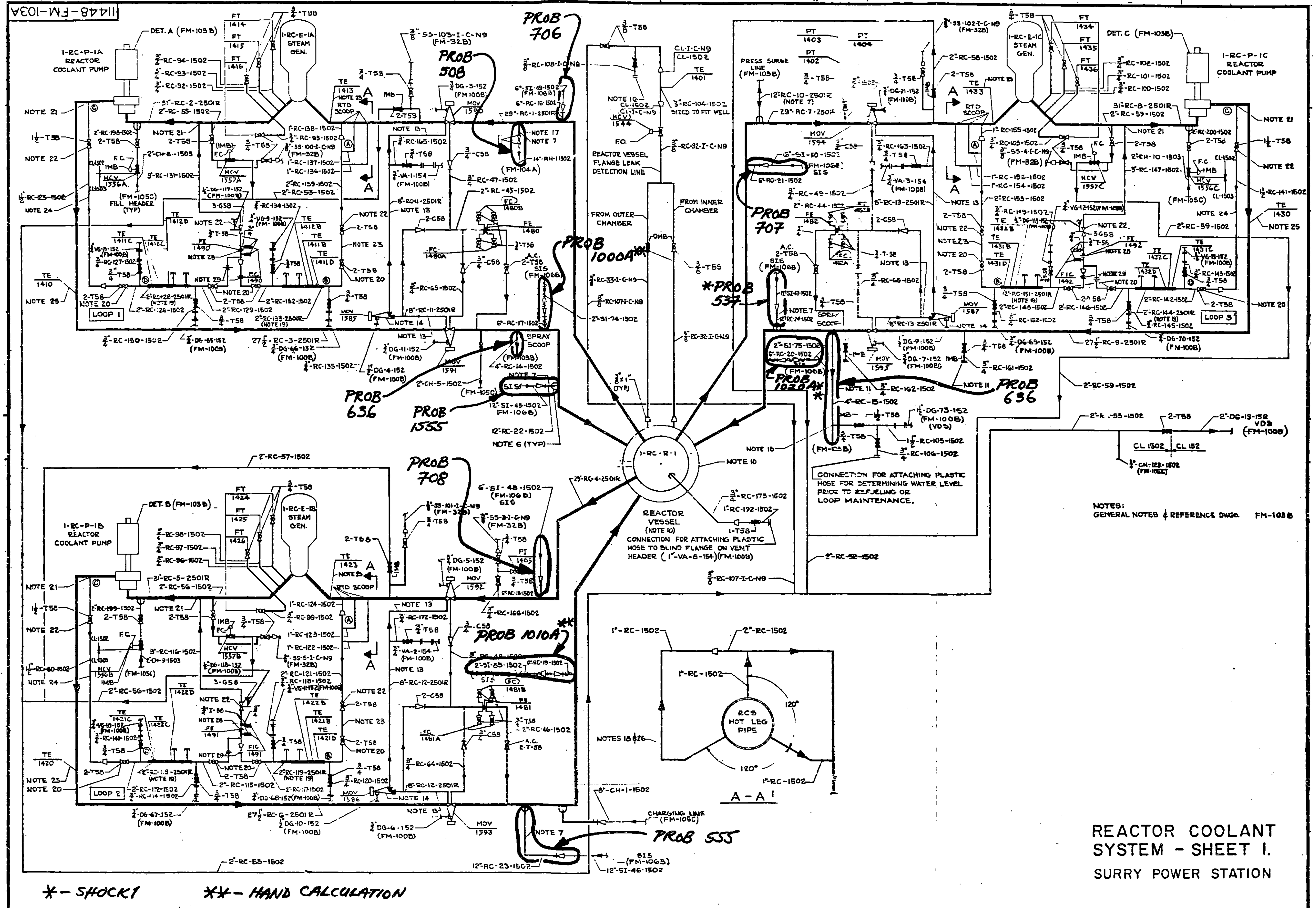
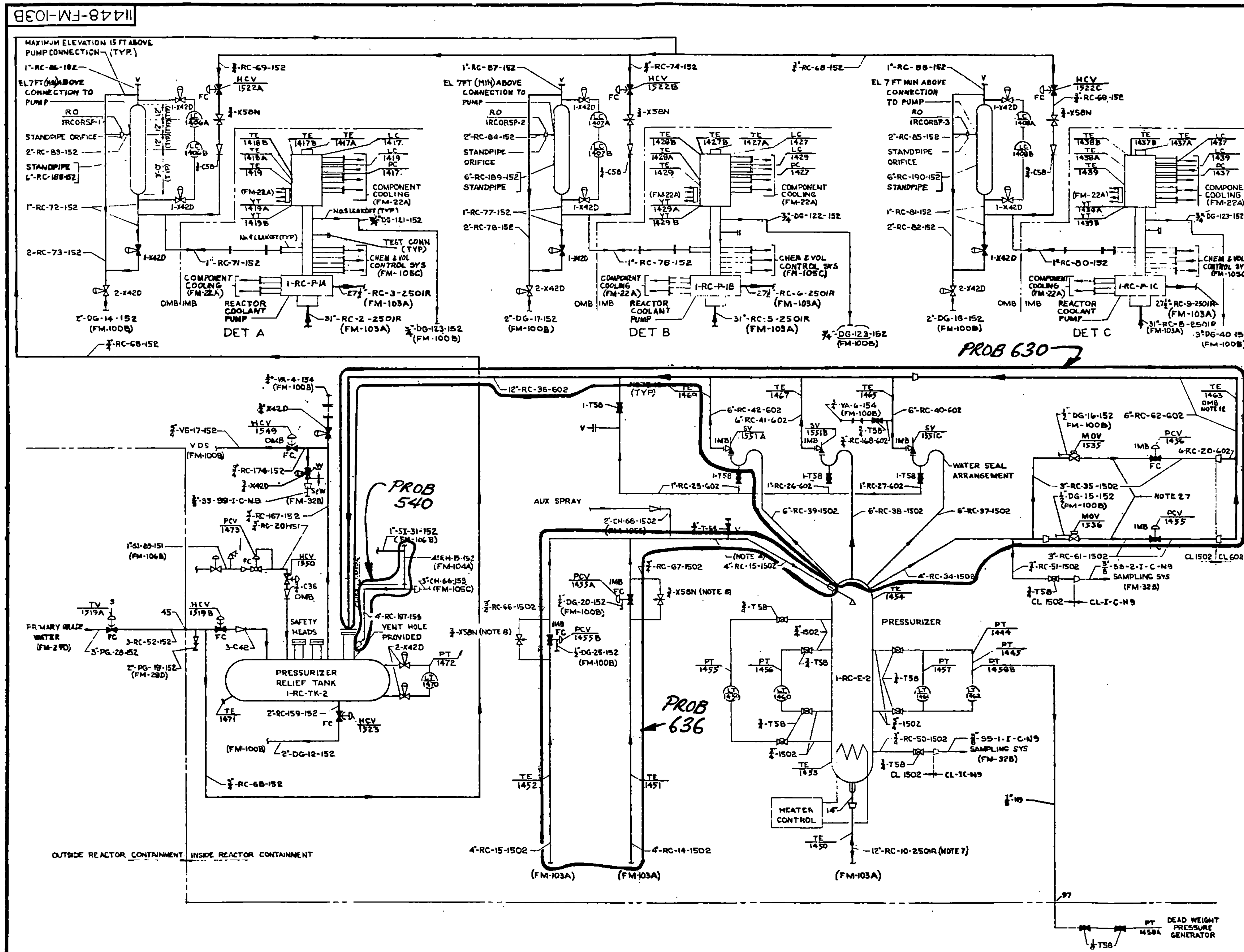


FIG. 4.2.1-2
OCT. 15, 1970



NOTES:

1. 27 1/2", 29" and 31" REFER TO INSIDE DIAMETER.
2. ALL CLASS 2501R PIPING SUPPLIED BY WESTINGHOUSE FOR DETAILS SEE W SPEC. G676341, 0676342 AND 0676343. ALL OTHER CLASSES - PIPING SPEC. NUS-20.
3. ALL HIGH PRESSURE VENTS TO BE BLANKED OFF WITH 1,500 LB FF 316SS BLIND FLANGE DURING NORMAL OPERATION.
4. ARRANGE SO THAT SPRAY VALVES WILL HAVE A WATER SEAL
5. GLOBE VALVES NORMALLY INSTALLED WITH FLOW UNDER SEAT EXCEPTIONS ARE VALVES DOWNSTREAM OF VALVES HCV-RC1557 A,B,&C
6. INDICATES CONNECTION WITH THERMAL SLEEVE.
7. SCHEDULE 140 PIPE.
8. PROVIDE EXTENSION STEM FOR ADJUSTMENT DURING NORMAL OPERATION.
9. SYMBOLS - PIPING SPECIFICATIONS.
10. ACCESS FOR INSPECTION OF REACTOR COOLANT PIPE TO REACTOR VESSEL NOZZLE WELDS TO BE PROVIDED.
11. TEST CONNECTIONS FOR LOOP STOP VALVE ANALYSIS
12. PLACE DETECTORS AT BOTTOM OF PIPE.
13. SCHEDULE 120 PIPE.
14. 0.25" HOLE THROUGH VALVE PLUG.
15. LOCATE BELOW ELEVATION OF REACTOR VESSEL FLANGE.
16. LOCAL COLLECTION POINT FOR DETERMINING LEAKAGE RATE.
17. LOCATE CONNECTION 45° BELOW HORIZONTAL CENTER LINE.
18. HOT LEG BYPASS LINE SCOOPS - ALL LOCATED IN SAME CROSS-SECTIONAL PLANE (SEE SECTION A-A).
19. RTD MANIFOLD AND RTD's SUPPLIED AS PACKAGE BY WESTINGHOUSE (MANIFOLD APPROXIMATELY 20" LONG).
20. LOCATE VALVES APPROXIMATELY 9" FROM MANIFOLD
21. LOCATE CONNECTION ON UPPER 180° OF PIPE CIRCUMFERENCE
22. LOCATE VALVE ABOVE CENTER LINE ELEVATION OF REACTOR VESSEL NOZZLES.
23. TOTAL LENGTH OF HOT LEG 2" SECTION UPSTREAM OF RTD MANIFOLD A TO B TO BE A MAXIMUM OF 4'-6 1/4". ALL BYPASS LOOP PIPING AND RTD MANIFOLDS SHALL HAVE REMOVABLE INSULATION UP TO LOOP ROOT VALVES
24. TOTAL LENGTH OF COLD LEG SECTION UPSTREAM OF RTD MANIFOLD C TO D TO BE A MAXIMUM OF 6'-6".
25. RTD INSTALLED IN WELL
26. PARALLEL PIPE PATHS SHOULD BE APPROXIMATELY EQUIVALENT LENGTHS. FLOW PATH LENGTH OF EACH 1" PIPING SECTION SHOULD NOT EXCEED 12'-0".
27. PROVIDE WATER SEAL FOR RELIEF VALVES
28. LOCATE ORIFICE IN BYPASS RETURN LINE WITH MINIMUM OF 20 PIPE DIAMETERS OF UPSTREAM STRAIGHT RUN.
29. FLANGES INSTALLED FOR INSERTION OF FLOW LIMITING ORIFICES IF REQUIRED. BLANK ORIFICE PLATE WILL BE DRILLED IN FIELD AS REQUIRED.
30. DRAIN LINES TO BE SLOPED DOWN FROM LOOP SEALS TO CONNECT WITH COMMON RELIEF LINE TO PRESSURIZER RELIEF TANK.

LEGEND

A.C. - INDICATED MANUAL VALVE POSITION MAINTAINED BY ADMINISTRATIVE CONTROL, WHICH IS CONSIDERED EQUIVALENT TO A LOCKED VALVE
IMB - INSIDE } MISSILE BARRIER AND/OR SECONDARY SHIELD
OMB - OUTSIDE } MISSILE BARRIER AND/OR SECONDARY SHIELD
FO - FAIL OPEN
FC - FAIL CLOSED
D - LOCAL DRAIN
V - LOCAL VENT
S - TRIPPED BY SAFETY INJECTION SIGNAL
- DIAPHRAGM VALVE WITH AIR OPERATOR
- NEEDLE VALVE
- HEAT TRACING

REFERENCE DWGS:

COMPONENT COOLING SYSTEM	FM-22A&B
BORON RECOVERY SYSTEM	FM-29A&D
SAMPLING SYSTEM	FM-32A
VENT & DRAIN SYSTEM	FM-100B
REACTOR COOLANT SYSTEM	FM-103A
RESIDUAL HEAT REMOVAL	FM-104A
CHEM & VOLUME CONTROL SYS	FM-105B&C
SAFETY INJECTION SYSTEM	FM-106B

REACTOR COOLANT SYSTEM - SHEET 2.

SURRY POWER STATION

FIG. 6.2.2.1-1
FEB. 11, 1972

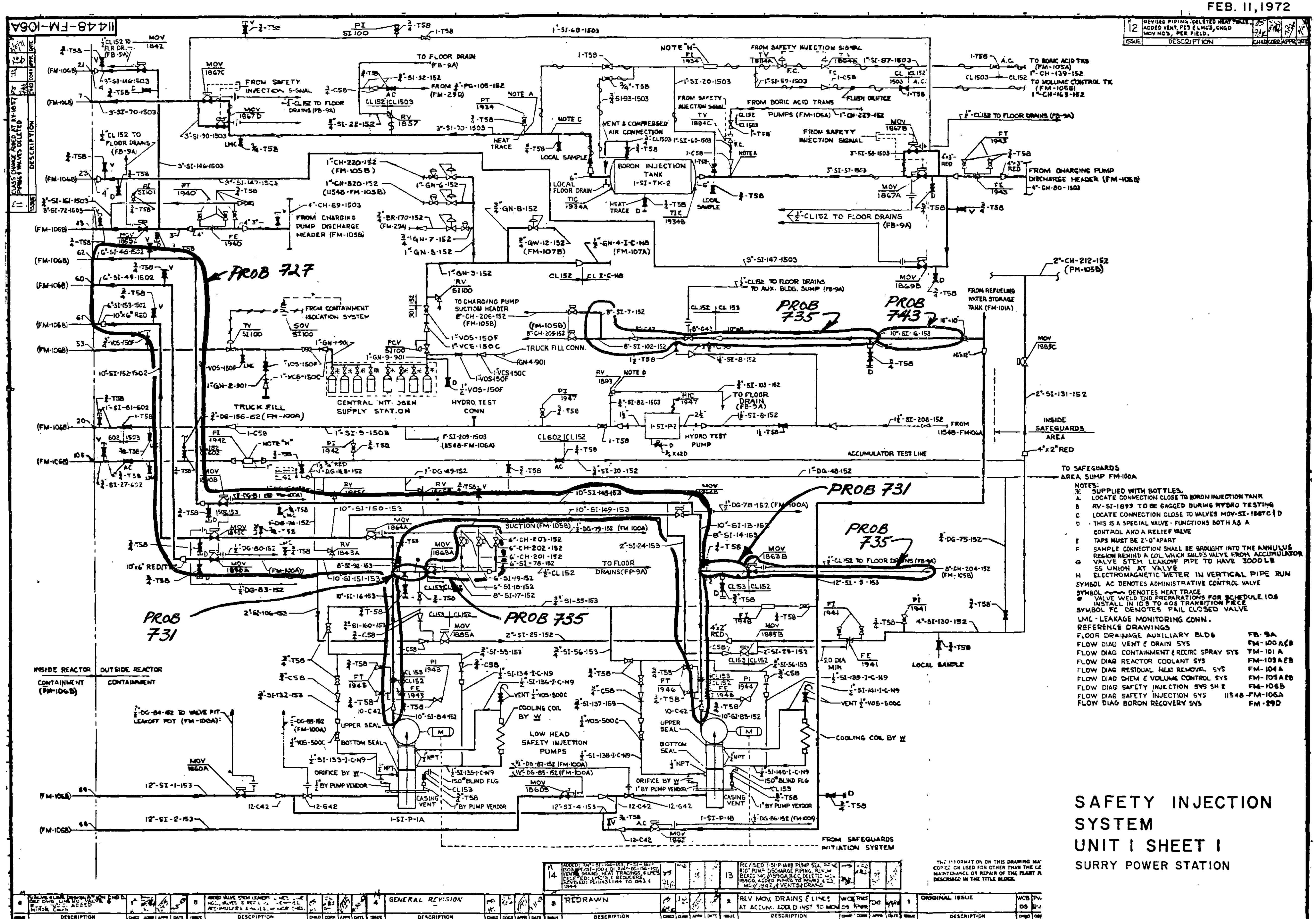


FIG. 6.2.2.1-2
FEB. 11, 1972

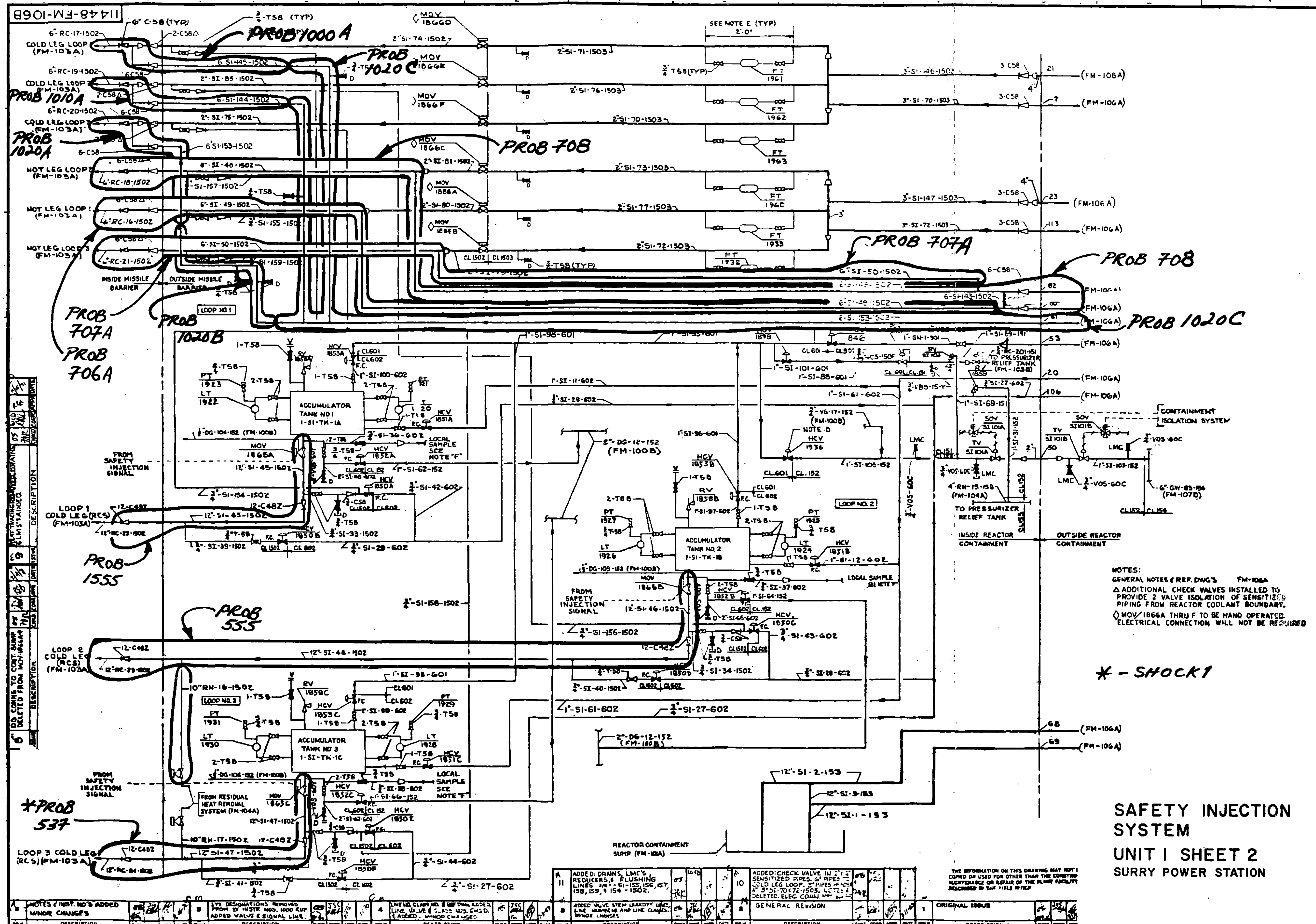
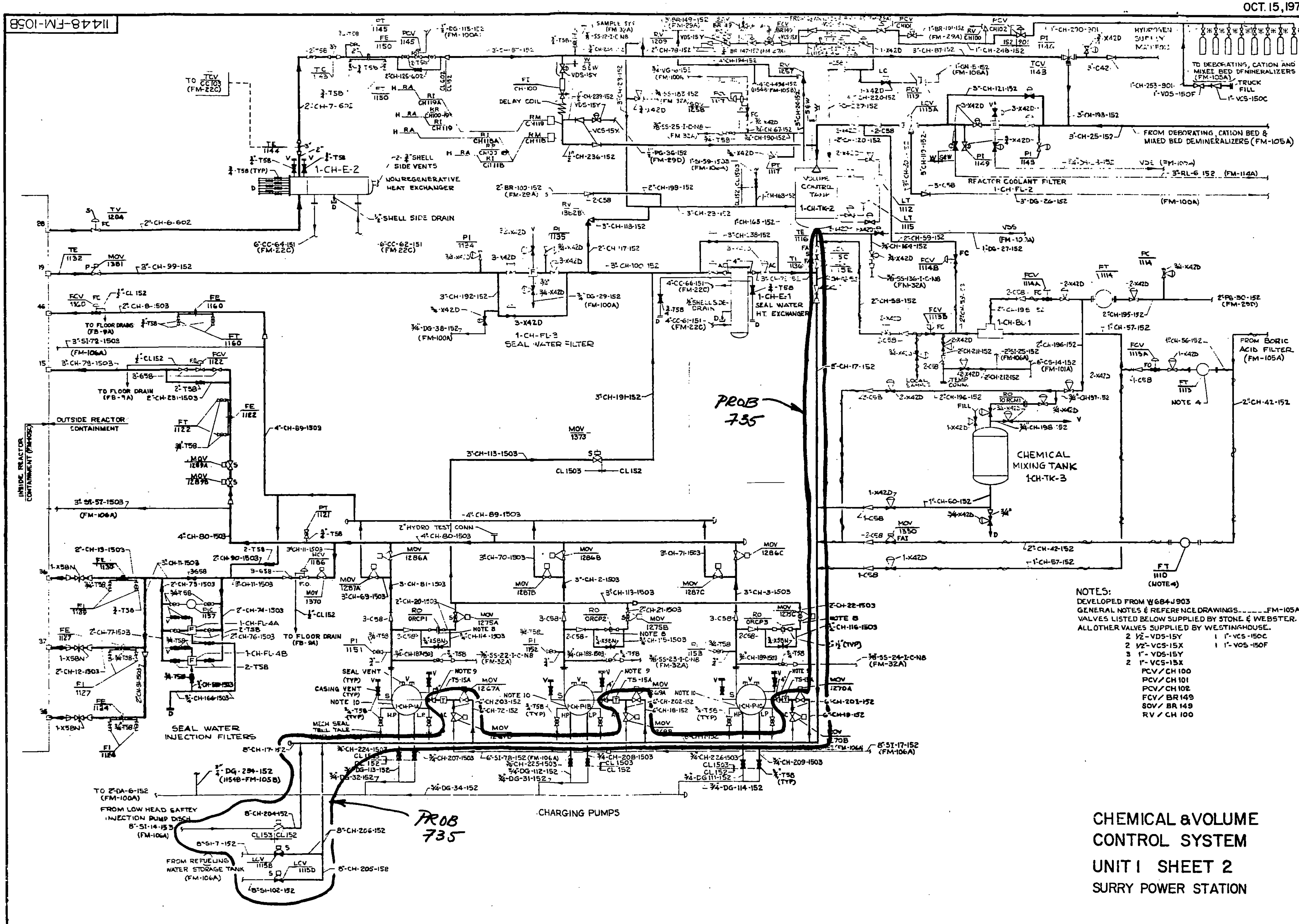


FIG. 9.1-2
OCT. 15, 1970



NOTES:
DEVELOPED FROM W684J903
GENERAL NOTES & REFERENCE DRAWINGS FM-105A
VALVES LISTED BELOW SUPPLIED BY STONE & WEBSTER.
ALL OTHER VALVES SUPPLIED BY WESTINGHOUSE.

2 1/2"-VDS-15Y	1 1"-VCS-150C
2 1/2"-VCS-15X	1 1"-VOS-150F
3 1"-VDS-15Y	
2 1"-VCS-15X	
PCV/CH 100	
PCV/CH 101	
PCV/CH 102	
FCV/BR 149	
SOV/BR 149	
RV/CH 100	

CHEMICAL & VOLUME CONTROL SYSTEM
UNIT I SHEET 2
SURRY POWER STATION

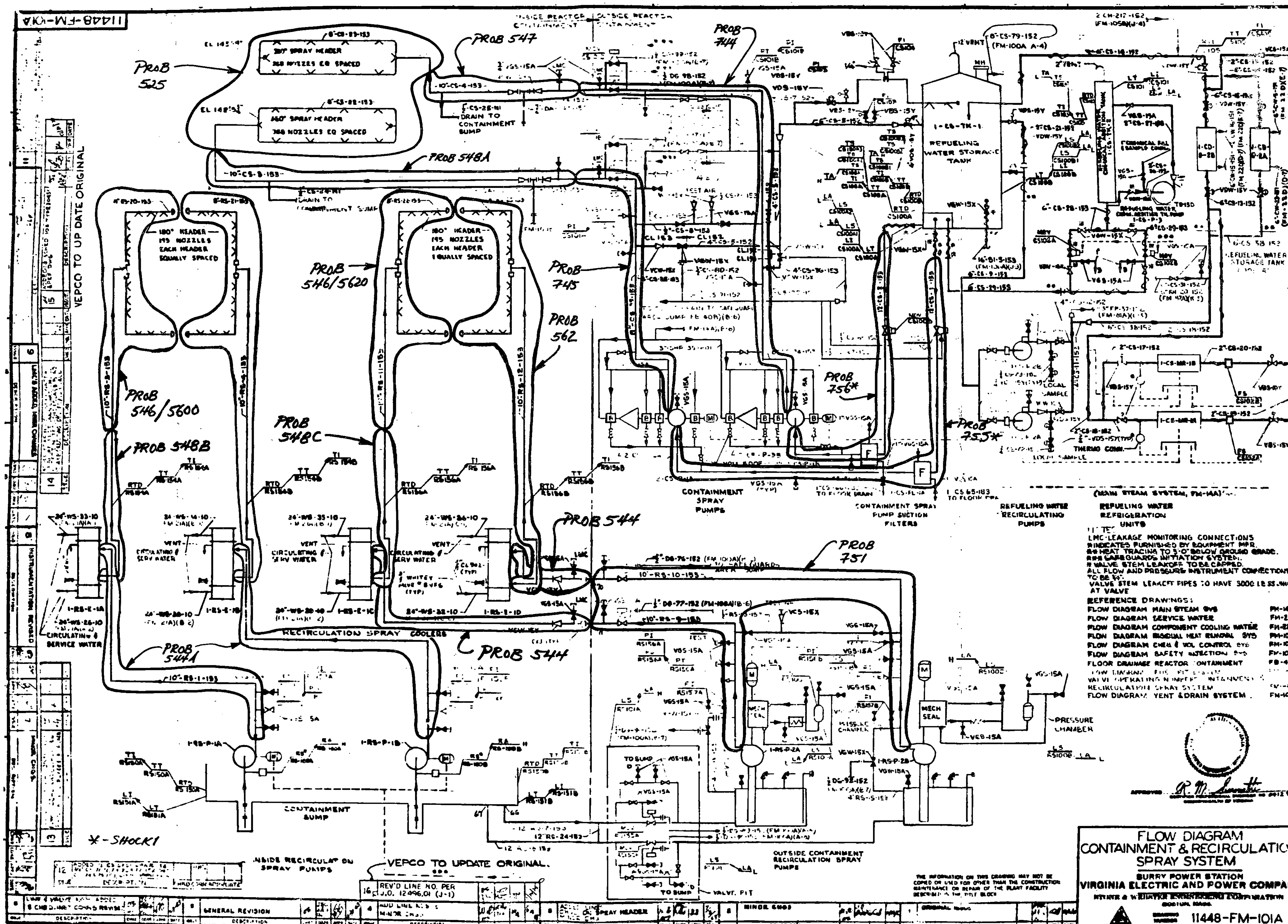
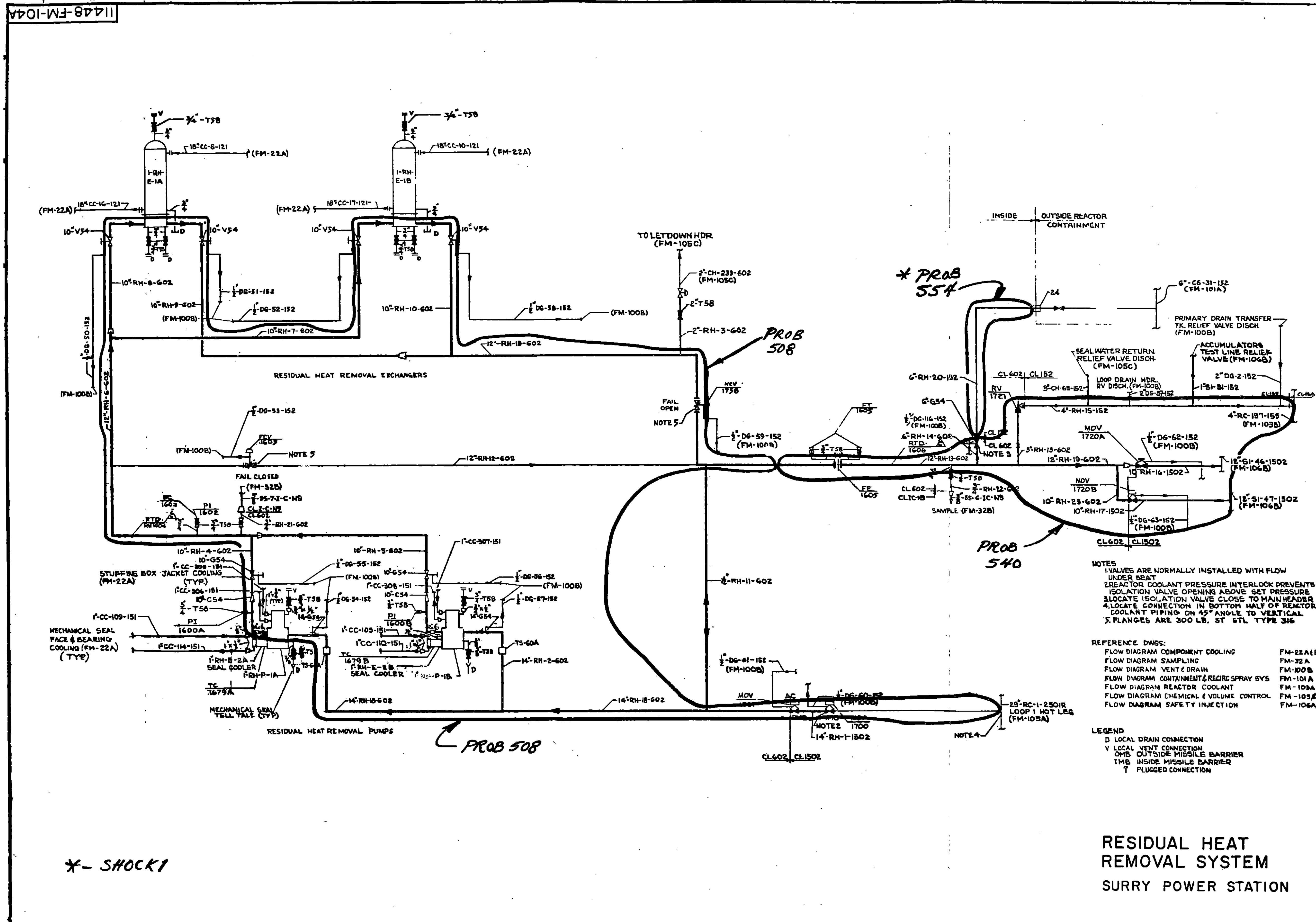

11448-FM-101A
PROB 525
PROB 547
PROB 744
PROB 548A
PROB 546/5620
PROB 745
PROB 562
PROB 546/5600
PROB 548B
PROB 548C
PROB 756*
PROB 755*
PROB 544
PROB 751
PROB 544A
PROB 544
300° SPRAY HEADER
368 NOZZLES EQ. SPACED
360° SPRAY HEADER
368 NOZZLES EQ SPACED
180° HEADER
195 NOZZLES
EACH HEADER
EQUALLY SPACED
DRAIN TO CONTAINMENT SUMP
REFUELING WATER STORAGE TANK
REFUELING WATER STORAGE TANK
CONTAINMENT SPRAY PUMPS
CONTAINMENT SPRAY PUMP SUCTION FILTERS
REFUELING WATER RECIRCULATING PUMPS
REFUELING WATER REFRIGERATION UNITS
RECIRCULATION SPRAY COOLERS
CIRCULATING & SERV WATER
CONTAINMENT SUMP
OUTSIDE CONTAINMENT RECIRCULATION SPRAY PUMPS
INSIDE RECIRCULATION SPRAY PUMPS
PRESSURE CHAMBER
MECH SEAL
THERMO COMM.
LOCAL SAMPLE
TO SUMP
VALVE PIT
VEPCO TO UP DATE ORIGINAL
VEPCO TO UPDATE ORIGINAL.
*-SHOCK1
LMC-LEAKAGE MONITORING CONNECTIONS
REFERENCE DRAWINGS:
FLOW DIAGRAM MAIN STEAM SYS FM-14A
FLOW DIAGRAM SERVICE WATER FM-21A
FLOW DIAGRAM COMPONENT COOLING WATER FM-22D
FLOW DIAGRAM RESIDUAL HEAT REMOVAL SYS FM-101A
FLOW DIAGRAM CHEM & VOL CONTROL SYS FM-108D
FLOW DIAGRAM SAFETY INJECTION SYS FM-106A
FLOOR DRAINAGE REACTOR CONTAINMENT FB-40B
FLOW DIAGRAM VENT & DRAIN SYSTEM FM-100A
REV'D LINE NO. PER J.O. 12496.01 (J-1)
GENERAL REVISION
FLOW DIAGRAM
CONTAINMENT & RECIRCULATION SPRAY SYSTEM
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
11448-FM-101A

FIG. 9.3-1
OCT. 15, 1970



RESIDUAL HEAT REMOVAL SYSTEM
SURRY POWER STATION

FIG. 9.4-1
OCT. 15, 1970

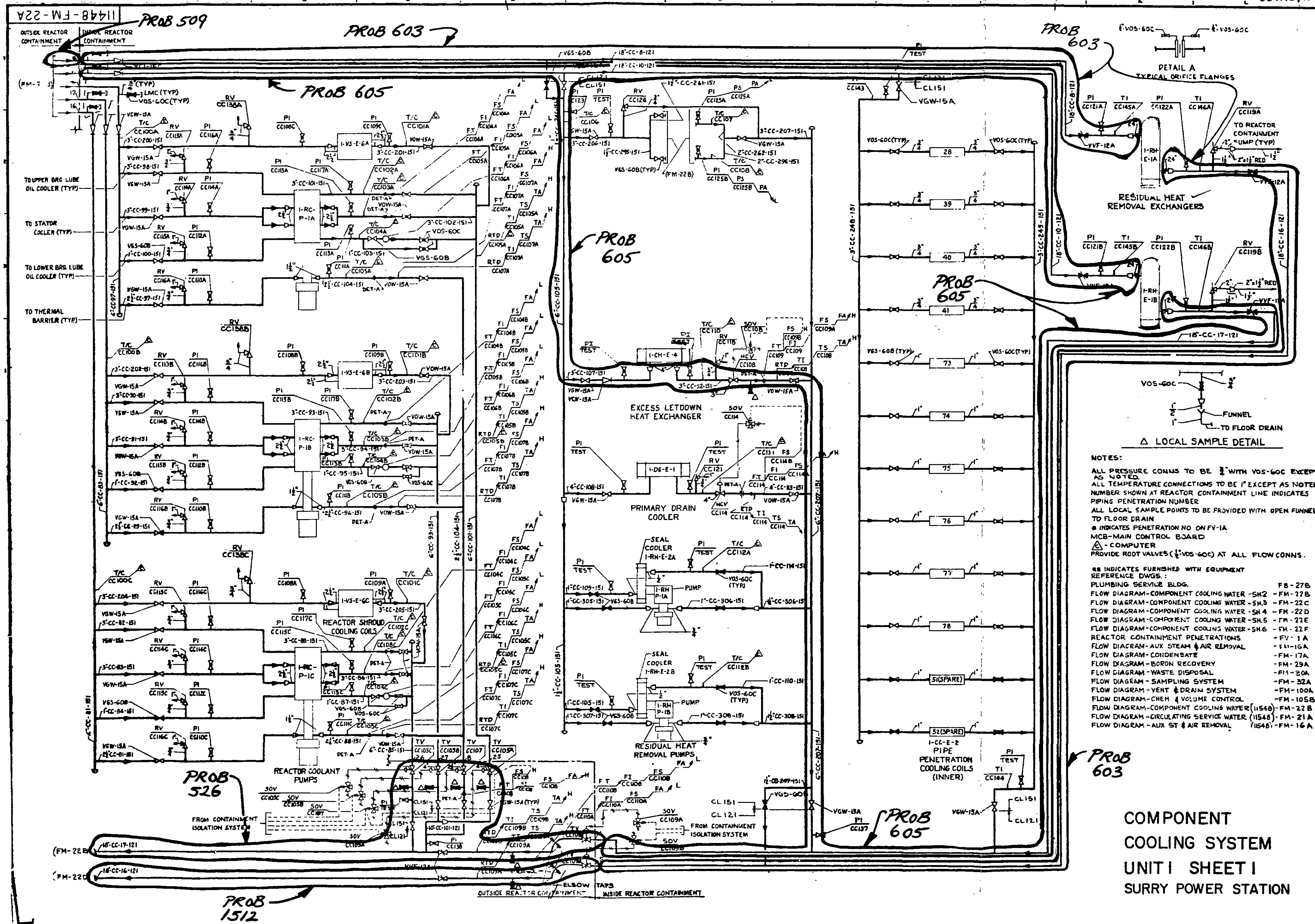
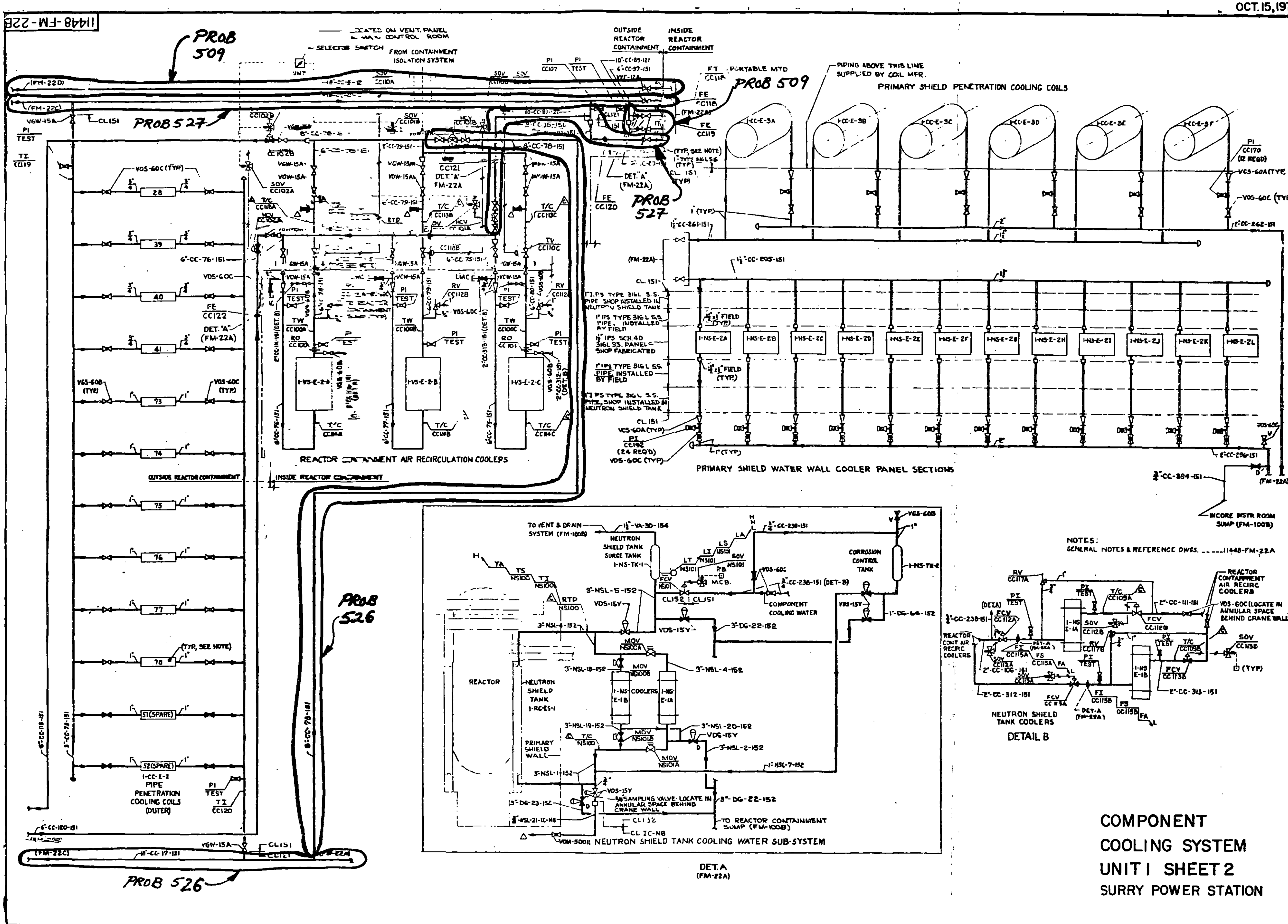
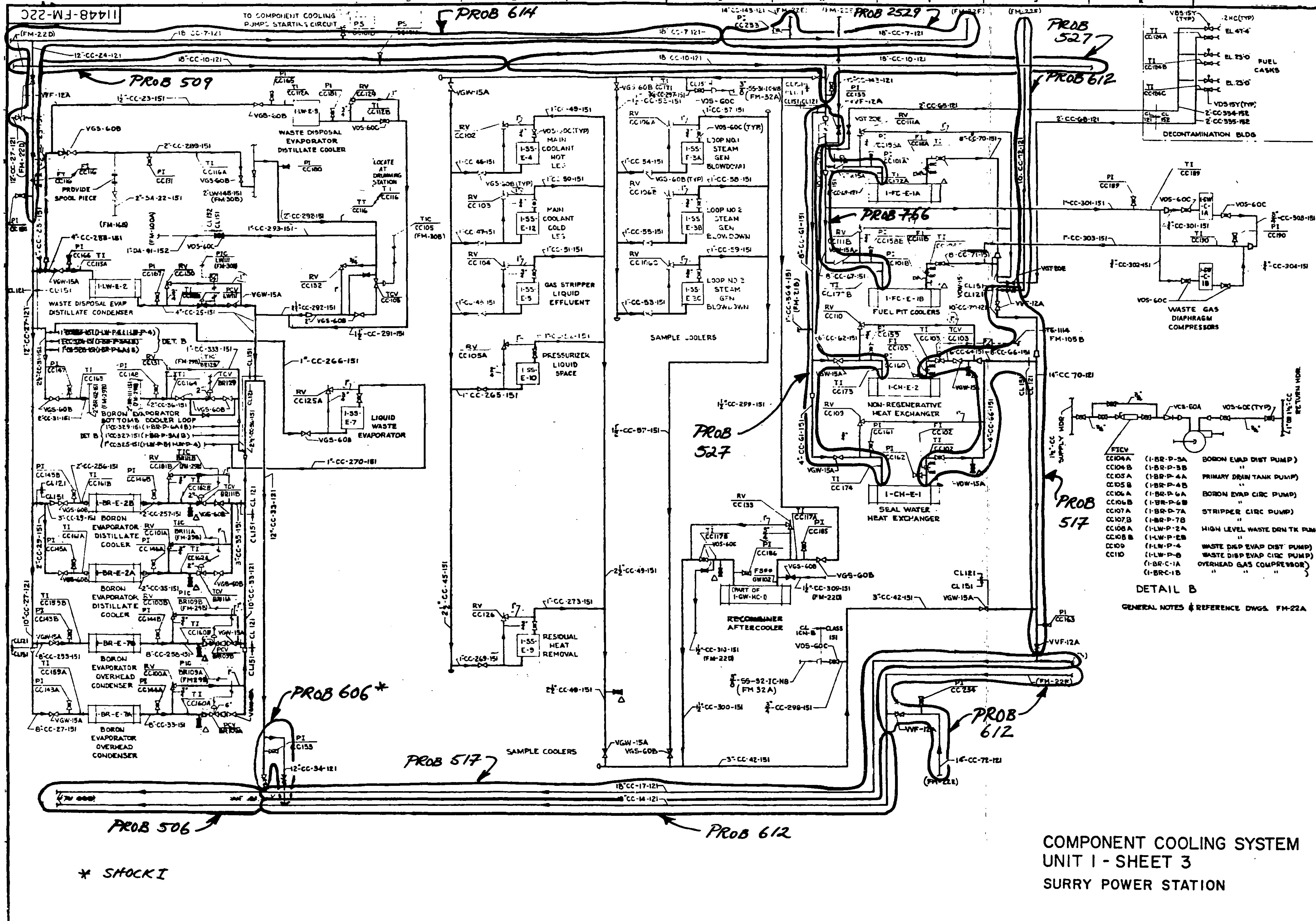


FIG. 94-2
OCT. 15, 1970

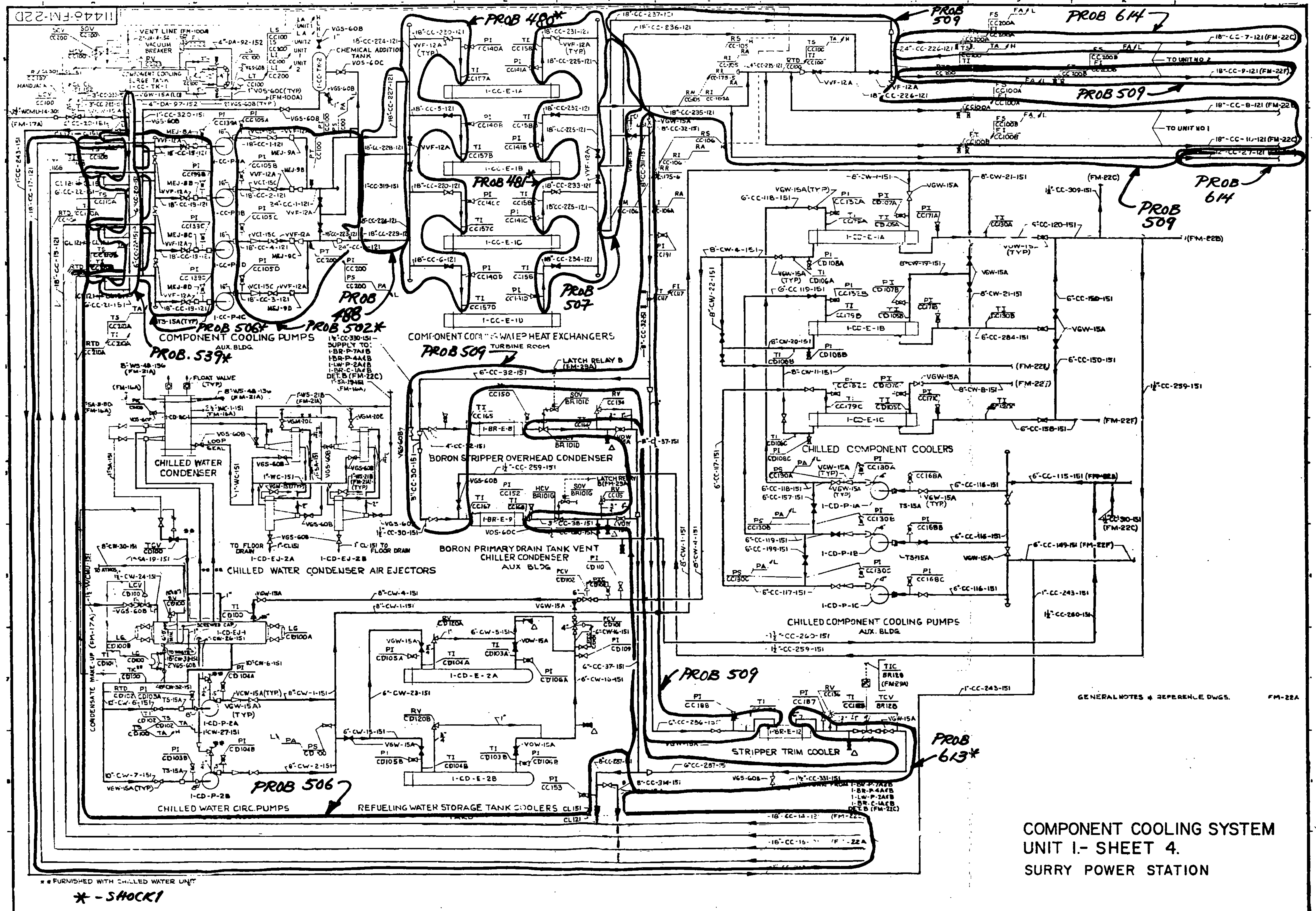




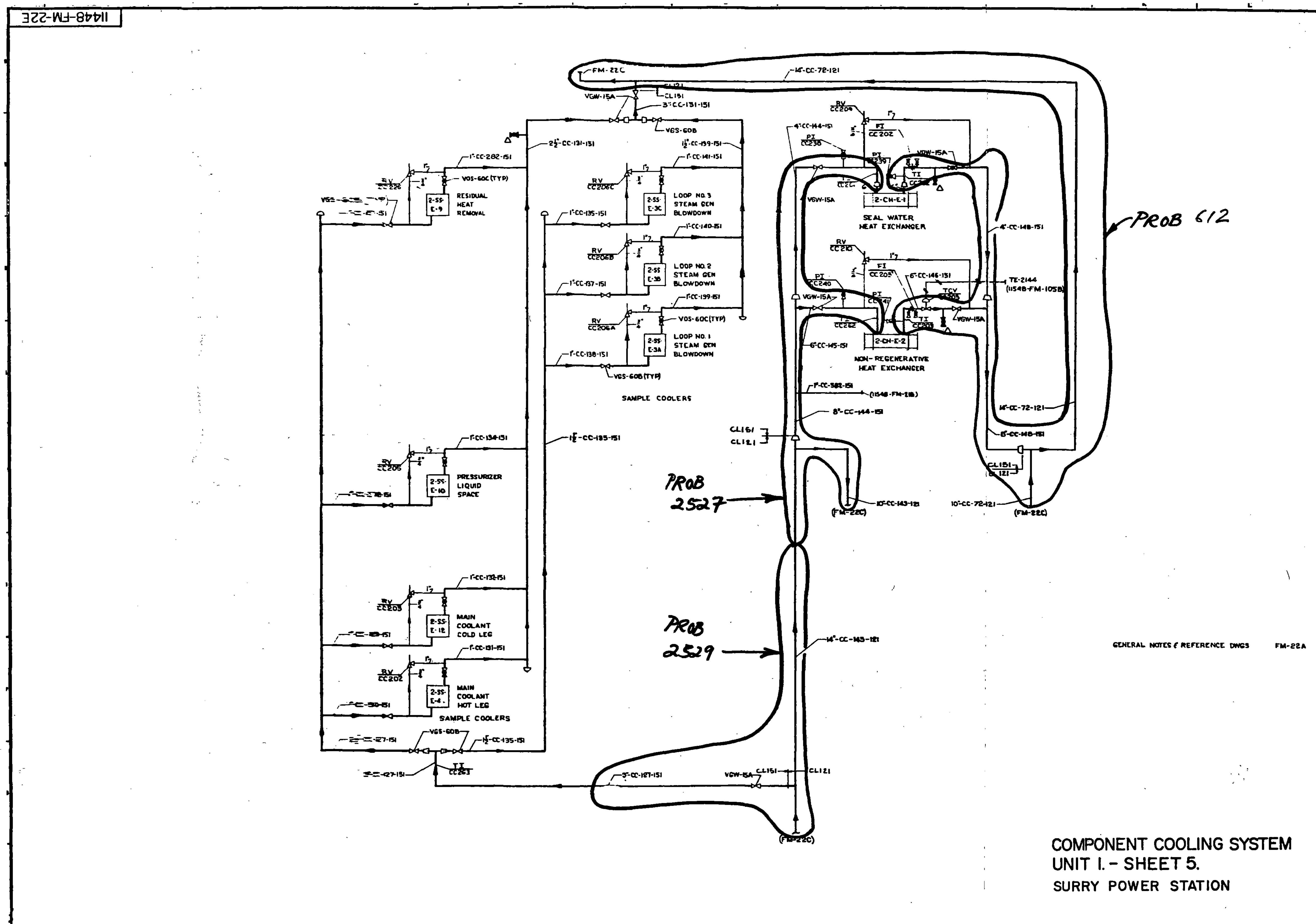
COMPONENT COOLING SYSTEM
UNIT I - SHEET 3
SURRY POWER STATION

* SHOCK I

FIG. 9.4-4
OCT. 15, 1970

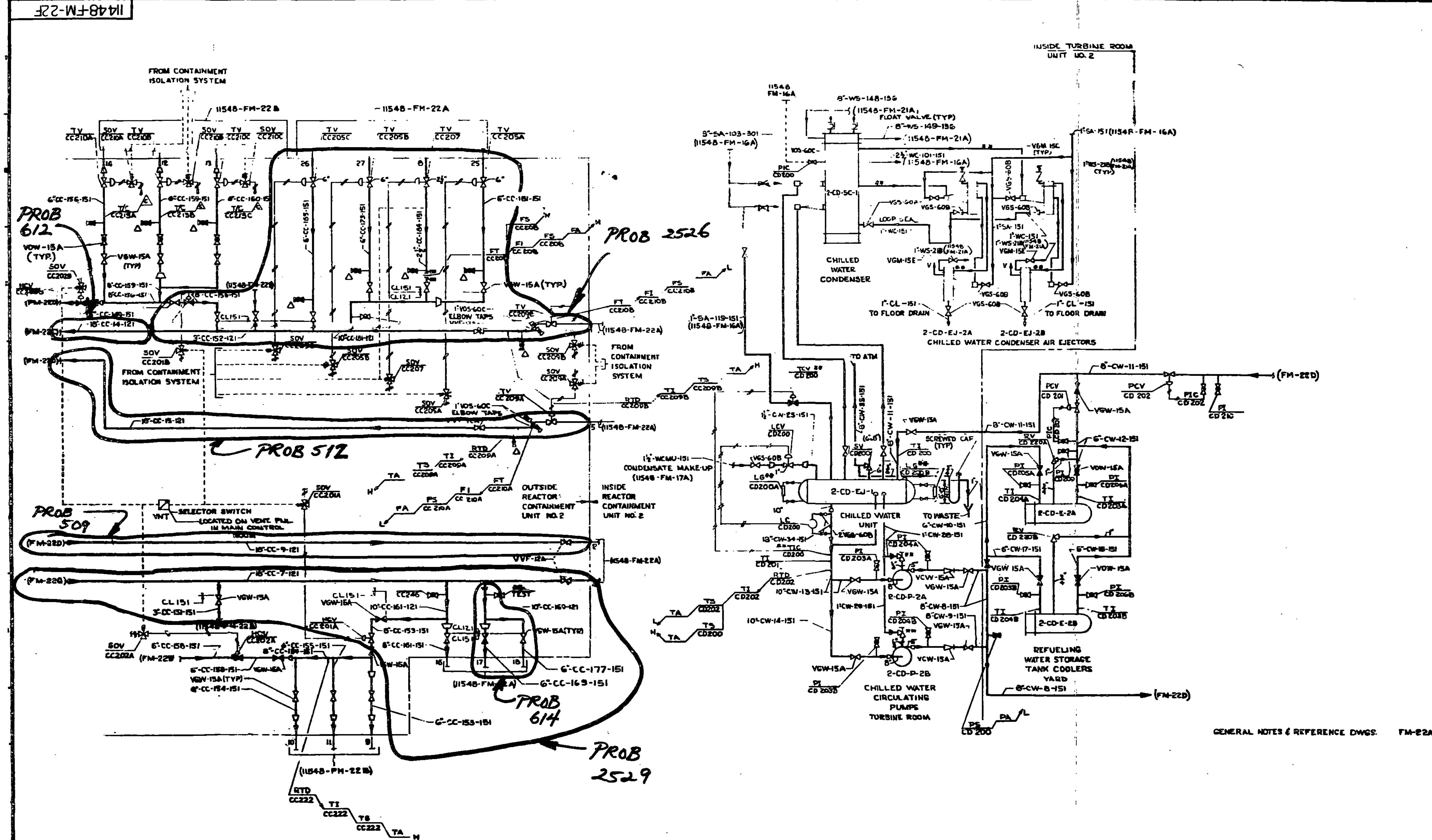


COMPONENT COOLING SYSTEM
UNIT 1.- SHEET 4.
SURRY POWER STATION



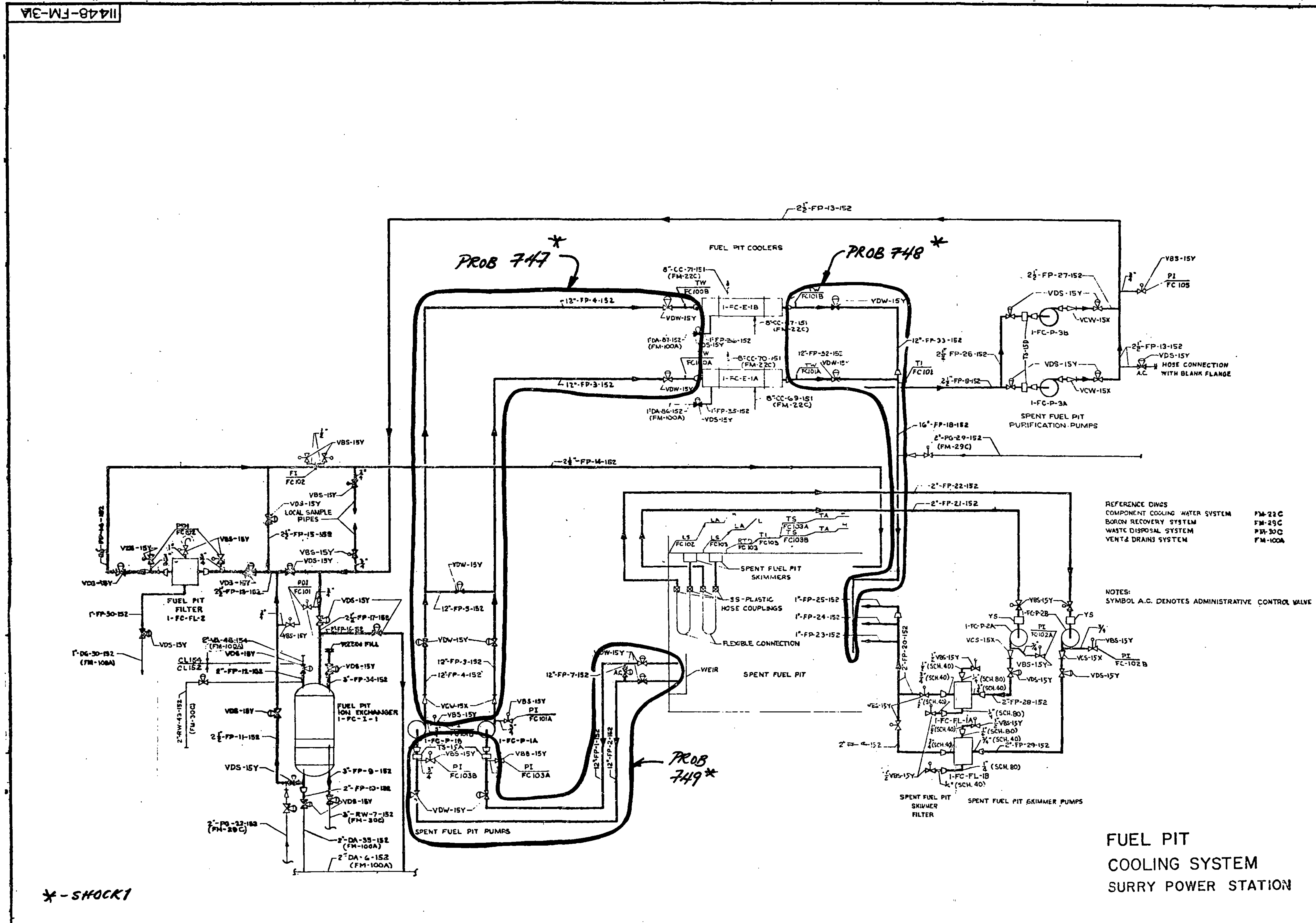
COMPONENT COOLING SYSTEM
UNIT 1. - SHEET 5.
SURRY POWER STATION

FIG. 9.4-6
OCT. 15,1970

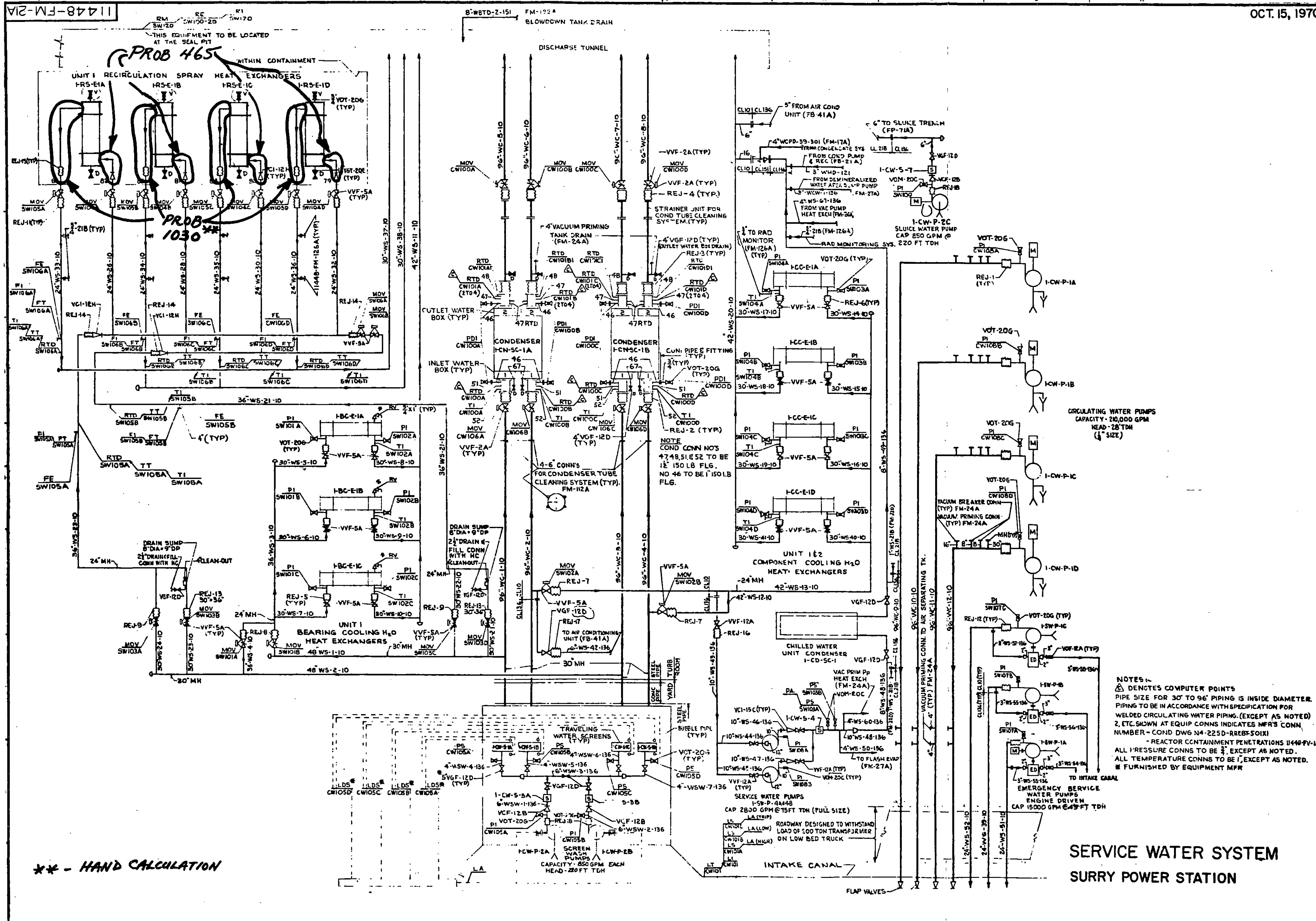


COMPONENT COOLING SYSTEM
UNIT 1 - SHEET 6
SURRY POWER STATION

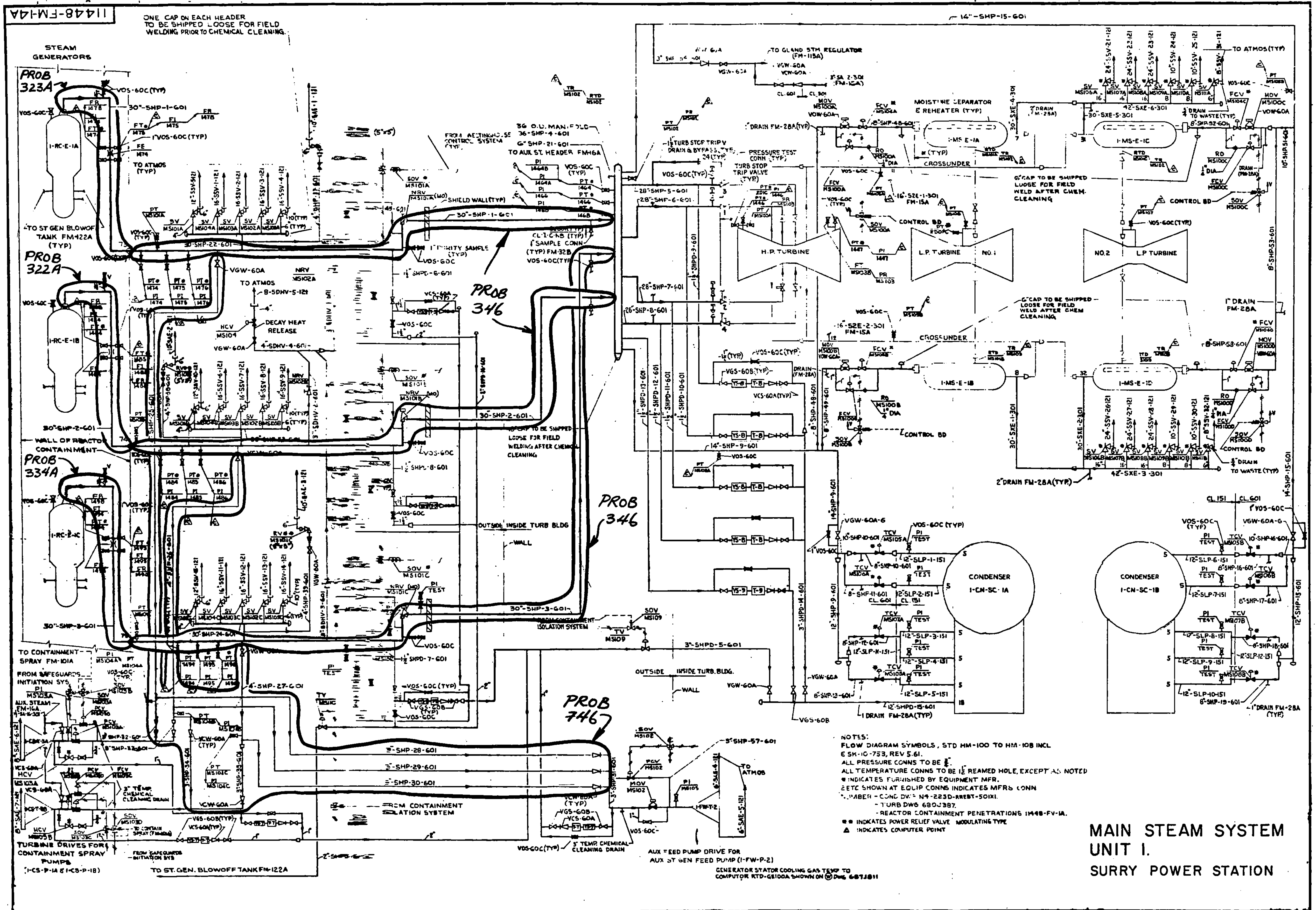
FIG. 9.5-1
OCT. 15, 1970



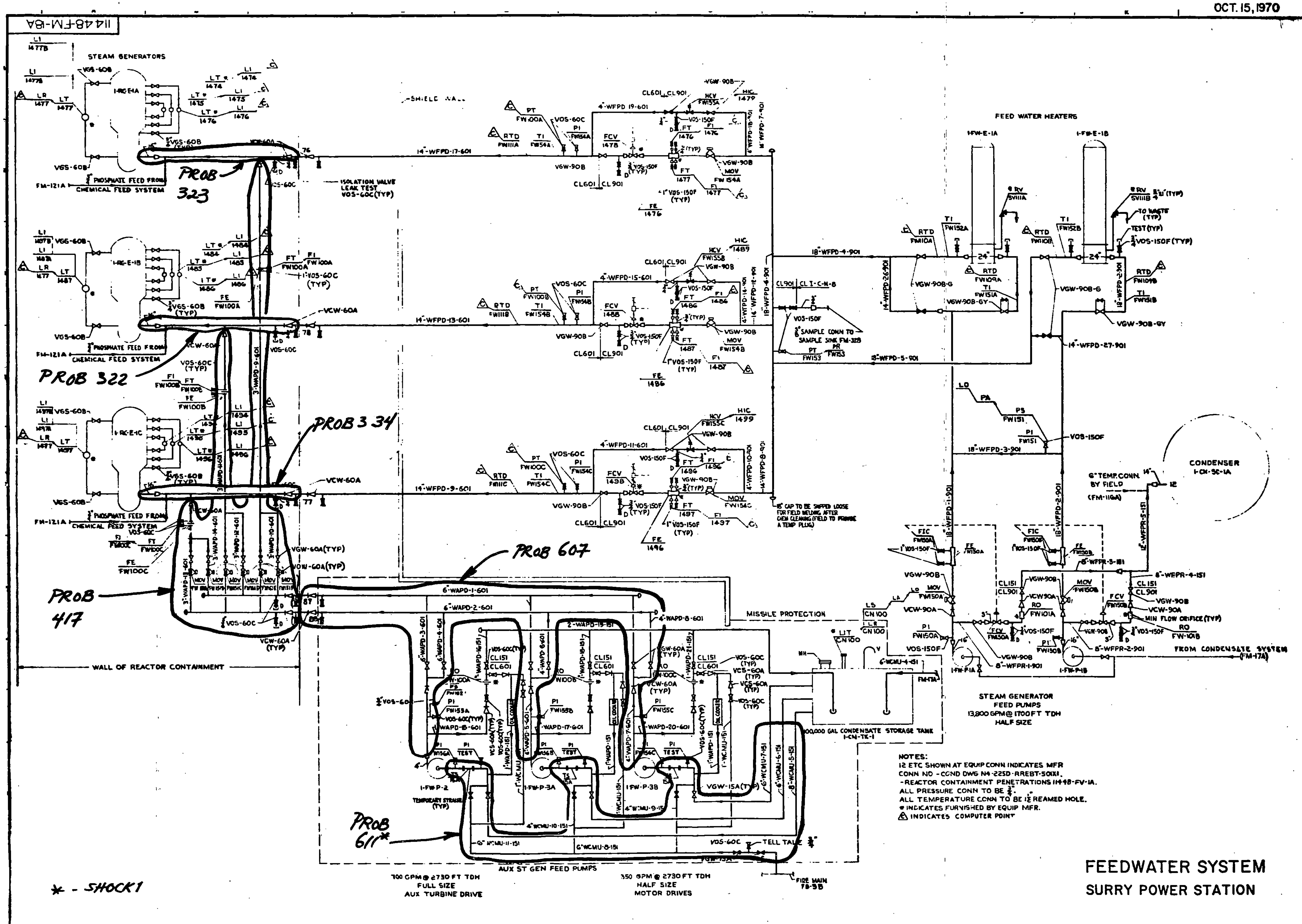
FUEL PIT
COOLING SYSTEM
SURRY POWER STATION

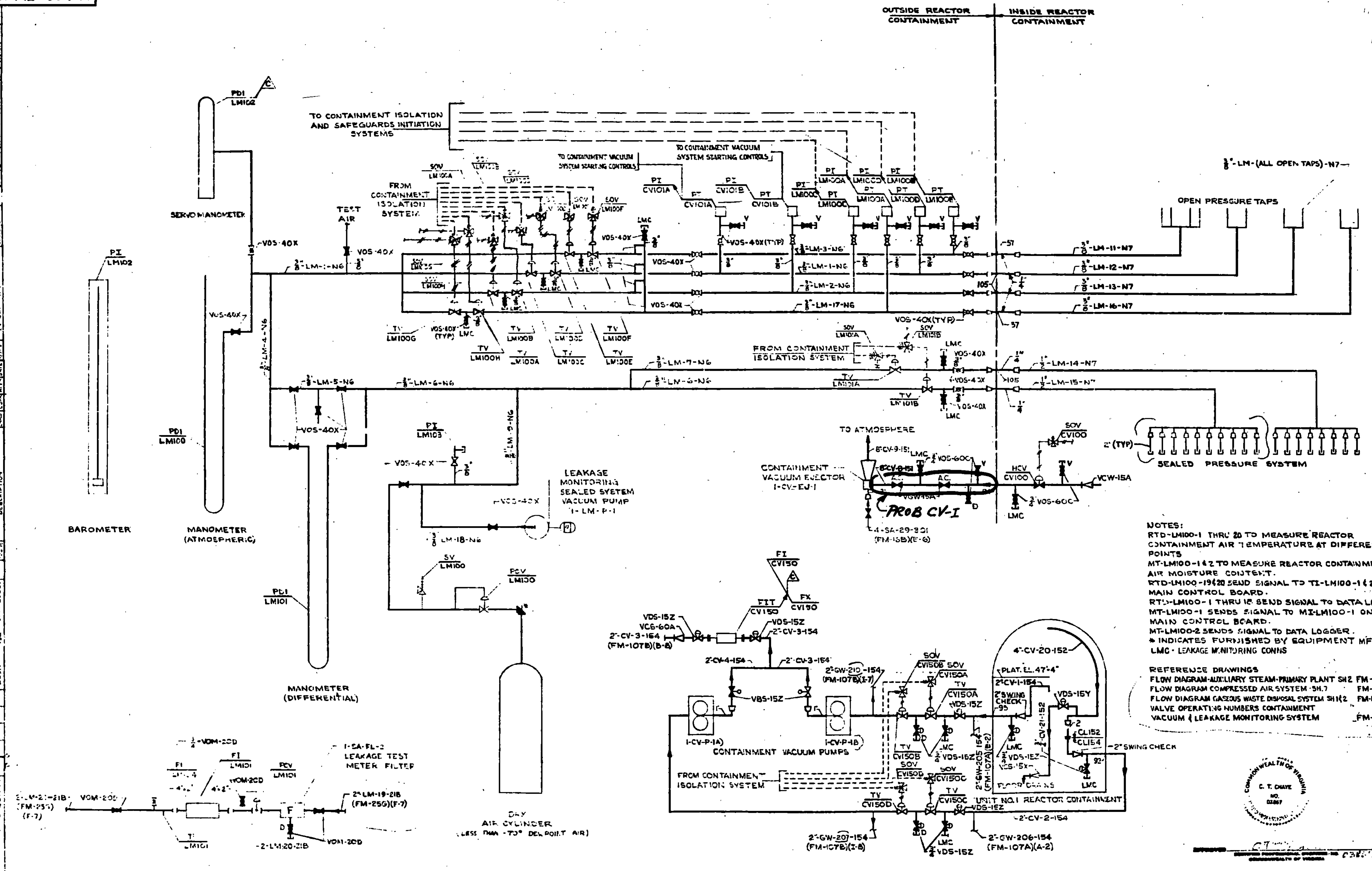


SERVICE WATER SYSTEM
SURRY POWER STATION



MAIN STEAM SYSTEM
UNIT I.
SURRY POWER STATION


11448-FM-18A
STEAM GENERATORS
FEED WATER HEATERS
CONDENSER
1-CN-SC-1A
WALL OF REACTOR CONTAINMENT
MISSILE PROTECTION
100,000 GAL CONDENSATE STORAGE TANK
1-CN-TK-1
AUX ST GEN FEED PUMPS
700 GPM @ 2730 FT TDH
FULL SIZE
AUX TURBINE DRIVE
350 GPM @ 2730 FT TDH
HALF SIZE
MOTOR DRIVES
STEAM GENERATOR
FEED PUMPS
13,800 GPM @ 1700 FT TDH
HALF SIZE
FROM CONDENSATE SYSTEM
NOTES:
12 ETC SHOWN AT EQUIP CONN INDICATES MFR
CONN NO - CGND DWG N4-225D-RREBT-501X1.
-REACTOR CONTAINMENT PENETRATIONS 11448-FV-1A.
ALL PRESSURE CONN TO BE 3/4".
ALL TEMPERATURE CONN TO BE 1 1/2 REAMED HOLE.
* INDICATES FURNISHED BY EQUIP MFR.
INDICATES COMPUTER POINT
PROB 323
PROB 322
PROB 334
PROB 417
PROB 607
PROB 611*
* - SHOCK1
FEEDWATER SYSTEM
SURRY POWER STATION



NOTES:
RTD-LM100-1 THRU 20 TO MEASURE REACTOR CONTAINMENT AIR TEMPERATURE AT DIFFERENT POINTS
MT-LM100-1 & 2 TO MEASURE REACTOR CONTAINMENT AIR MOISTURE CONTENT.
RTD-LM100-19 & 20 SEND SIGNAL TO TI-LM100-1 & 2 ON MAIN CONTROL BOARD.
RTD-LM100-1 THRU 18 SEND SIGNAL TO DATA LOGGER
MT-LM100-1 SENDS SIGNAL TO MI-LM100-1 ON MAIN CONTROL BOARD.
MT-LM100-2 SENDS SIGNAL TO DATA LOGGER.
* INDICATES FURNISHED BY EQUIPMENT MFR
LMC - LEAKAGE MONITORING CONNS

REFERENCE DRAWINGS

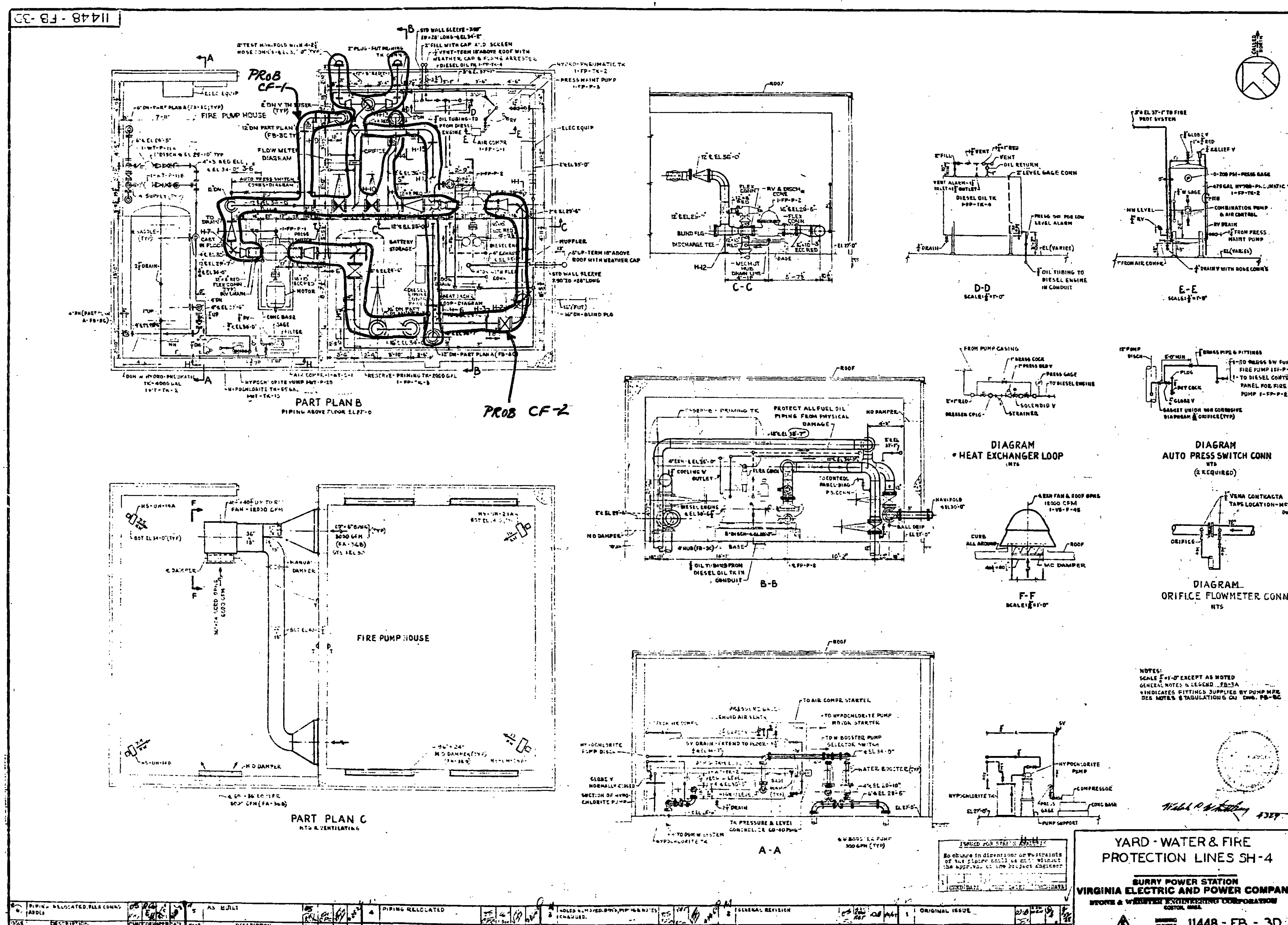
FLOW DIAGRAM-AUXILIARY STEAM-PRIMARY PLANT SH 2	FM-16B
FLOW DIAGRAM COMPRESSED AIR SYSTEM SH.7	FM-25G
FLOW DIAGRAM GASEOUS WASTE DISPOSAL SYSTEM SH 1 & 2	FM-107A & B
VALVE OPERATING NUMBERS CONTAINMENT VACUUM & LEAKAGE MONITORING SYSTEM	FM-85A

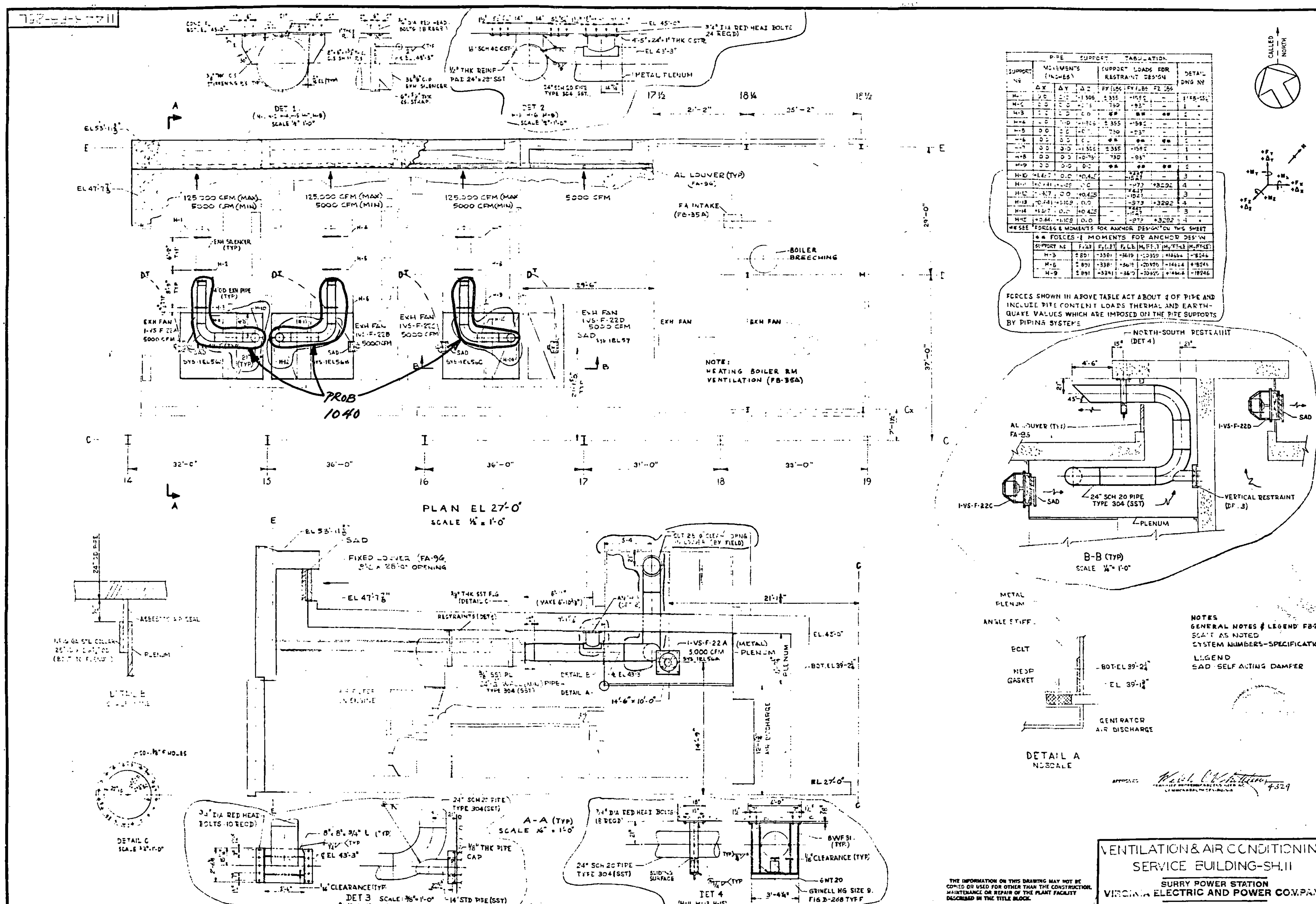
ISSUE	DESCRIPTION
13	REVISED EQUIPMENT NO'S, LINE NO'S & DWG. REFS.
12	ADDED VENT VALVE AT EV. [illegible]
3	LINE NUMBERS [illegible]
2	MINOR CHG.
1	ORIGINAL ISSUE
4	MINOR CHG.
5	GENERAL REVISION

FLOW DIAGRAM
CONTAINMENT VAC. & LEAKAGE MONITOR SYSTEM

SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 11448-FM-102A


11448 - FB - 3D
PROB CF-1
PROB CF-2
FIRE PUMP HOUSE
PART PLAN B
PART PLAN C
C-C
D-D
E-E
B-B
F-F
A-A
DIAGRAM
HEAT EXCHANGER LOOP
DIAGRAM
AUTO PRESS SWITCH CONN
DIAGRAM
ORIFICE FLOWMETER CONN
YARD - WATER & FIRE
PROTECTION LINES SH-4
SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY
STONE & WEBSTER ENGINEERING CORPORATION
11448 - FB - 3D
PIPING RELOCATED
ORIGINAL ISSUE
GENERAL REVISION
AS BUILT
DESCRIPTION



PIPE SUPPORT TABULATION

SUPPORT Nº	MOVEMENTS (INCHES) ΔX	ΔY	ΔZ	SUPPORT LOADS FOR RESTRAINT DESIGN FX (LBS)	FY (LBS)	FZ (LBS)	DETAIL DWG Nº
H-1	0.0	0.0	-1.306	±355	-1592	–	1 (FB-25L)
H-2	0.0	0.0	[illegible]	730	-937	–	1 "
H-3	0.0	0.0	0.0	**	**	**	2 "
H-4	0.0	0.0	-1.306	±355	-1592	–	1 "
H-5	0.0	0.0	[illegible]	730	-937	–	1 "
H-6	0.0	0.0	0.0	**	**	**	2 "
H-7	0.0	0.0	-1.306	±355	-1592	–	1 "
H-8	0.0	0.0	-0.751	730	-937	–	1 "
H-9	0.0	0.0	0.0	**	**	**	2 "
H-10	+1.427	0.0	+0.425	–	+447 -1527	–	3 "
H-11	+0.641	+1.109	0.0	–	-973	+3292	4 "
H-12	+1.427	0.0	+0.425	–	+447 -1527	–	3 "
H-13	+0.641	+1.109	0.0	–	-973	+3292	4 "
H-14	+1.427	0.0	+0.425	–	+447 -1527	–	3 "
H-15	+0.641	+1.109	0.0	–	-973	+3292	4 "

** SEE "FORCES & MOMENTS FOR ANCHOR DESIGN" ON THIS SHEET

** FORCES & MOMENTS FOR ANCHOR DESIGN

SUPPORT Nº	F_x (LB)	F_y (LB)	F_z (LB)	M_x (FT-LB)	M_y (FT-LB)	M_z (FT-LB)
H-3	±891	-3381	-3619	-20329	+14664	-18246
H-6	±891	-3381	+3619	+20329	-14664	+18246
H-9	±891	-3381	-3619	-20329	+14664	-18246

FORCES SHOWN IN ABOVE TABLE ACT ABOUT ℄ OF PIPE AND INCLUDE PIPE CONTENT LOADS THERMAL AND EARTHQUAKE VALUES WHICH ARE IMPOSED ON THE PIPE SUPPORTS BY PIPING SYSTEMS

APPROVED: 4329

ISSUE	DESCRIPTION
1	ORIGINAL ISSUE

VENTILATION & AIR CONDITIONING SERVICE BUILDING-SH.11

SURRY POWER STATION
VIRGINIA ELECTRIC AND POWER COMPANY

STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.

DRAWING NUMBER 11448-FB-25L

APPENDIX C 1.9

SNUBBER CAPACITIES 1.11

ITT Grinnell Corporation
Pipe Hanger Division
260 West Exchange Street
Providence, Rhode Island 02901
Telephone (401) 831-7000

April 11, 1979

Stone & Webster Engineering Corp.
245 Summer Street
Boston, MA 02210

Attention: Mr. Mel Pedell

Subject: ITT Grinnell Snubber Load Ratings

Dear Mel:

As we discussed on April 10, 1979, ITT Grinnell Load Ratings shown in our 1969 and 1972 catalogs for Hydraulic Snubbers were the normal condition Load Ratings. Not published in those catalogs, was a one-time Load Rating, which was available to our Engineering Staff for use on contracts whose specifications required this Loading Condition. The one-time Load Rating is described as an event whose Loading on the Snubber is expected only once, after which event the Snubber will be inspected and replaced. Examples of such events are Pipe Whip Restraint and SSE or DBE Seismic. The design criteria used for this vintage of Snubber was a maximum design stress in the Snubber of approximately 0.45Sy for normal condition and 0.9Sy for the one-time rating.

As I understand the situation, you have an OBE Seismic condition that would correspond to an occassional Loading Condition as defined by ANSI B31.1. Based upon this, we can increase our allowable stresses for the Snubber by 20% according to paragraph 121.1.2 (A.1) of ANSI B31.1 1967 and 1973 editions. This, in turn, will permit us to allow an increase of 20% for our Snubber catalog Load Ratings, prior to ASME III Subsection NF, if your OBE Seismic is categorized as an Occassional Load. These increases will apply to our Hydraulic Snubbers manufactured during the period of time in question for any of the five plants for which you are providing piping re-analysis. Our cylinder vendors during this time were Lindco, Lynair, Tompkins-Johnson and possibly Miller Fluid Power.

F 1958 Printed in U.S.A.

PAGE

TO Stone & Webster Engineering Corp.
245 Summer Street
Boston, MA 02210

April 11, 1979

If you have any further requirements that would be of service to this endeavor, please feel free to contact me.

Very truly yours,

ITT GRINNELL CORPORATION

R. J. Masterson, Manager
Research and Development
Engineering
Pipe Hanger Division

RJM:jp

cc: D. Brown
R. Lundgren

APPENDIX D 1.10

RESPONSE TO IE BULLETIN 79-04 1.12

SURRY POWER STATION, UNIT 1

APPENDIX D 1.12

RESPONSE TO IE BULLETIN 79-04 1.14

Velan swing check valves, sized 3 and 6 inches, are installed in the following 1.17
Seismic Category I piping systems:

a. Chemical and volume control system 1.19

b. Safety injection system 1.20

A detailed listing by line number follows. 1.22

Lines with 6 inch check valves were seismically analyzed by computer program. 1.24
The re-evaluation of these systems using the correct valve weight is currently 1.25
being done under the NUPIPE program. The results have shown that the pipe 1.26
stress is within the allowable for all lines. 1.27

Lines with 3 inch check valves were analyzed by hand calculations. An 1.29
estimated weight, overly conservative, was used instead of actual valve
weights. The incorrect valve weight has no effect on these calculations and 1.30
re-evaluation is not required.

LISTING OF VELAN SWING CHECK VALVES

COVERED BY IE BULLETIN NO. 79-04

SAFETY INJECTION SYSTEM - UNIT 1

6 Inch	1-SI-79	6-RC-17-1502
	1-SI-241	6-SI-145-1502
	1-SI-82	6-RC-19-1502
	1-SI-85	6-RC-20-1502
	1-SI-88	6-RC-18-1502
	1-SI-91	6-RC-16-1502
	1-SI-94	6-RC-21-1502
	1-SI-242	6-SI-144-1502
	1-SI-243	6-SI-153-1502
	1-SI-239	6-SI-49-1502
	1-SI-238	6-SI-48-1502
	1-SI-240	6-SI-50-1502
	1-SI-228	6-SI-48-1502
	1-SI-229	6-SI-49-1502
3 Inch	1-SI-224	3-SI-146-1503
	1-SI-225	3-SI-70-1503
	1-SI-226	3-SI-147-1503
	1-SI-227	3-SI-72-1503

CHEMICAL AND VOLUME CONTROL SYSTEM - UNIT 1

3 Inch	1-CH-258	3-CH-81-1503
	1-CH-267	3-CH-2-1503
	1-CH-276	3-CH-3-1503
	1-CH-196	3-CH-200-152
	1-CH-430	3-CH-1-1502
	1-CH-312	3-CH-1-1502
	1-CH-309	3-CH-79-1503

APPENDIX E 1.10

COMPARISON OF SHOCKO RESULTS WITH NUPIPE 1.12

APPENDIX E

COMPARISON OF SHOCKO RESULTS WITH NUPIPE

Problem No. 747 - Fuel Pool Cooling

For this example the natural frequencies computed by NUPIPE differ from those originally computed by SHOCKO, but the computed pipe stresses do not differ appreciably. The SHOCKO and NUPIPE stresses show similar patterns, with peak stresses at the same locations, and the maximum stress computed by SHOCKO is substantially more conservative than the maximum NUPIPE stress. The SHOCKO support reactions are also more conservative than the NUPIPE results. The original licensed amplified response spectra were used for both analyses. Supporting data follow.

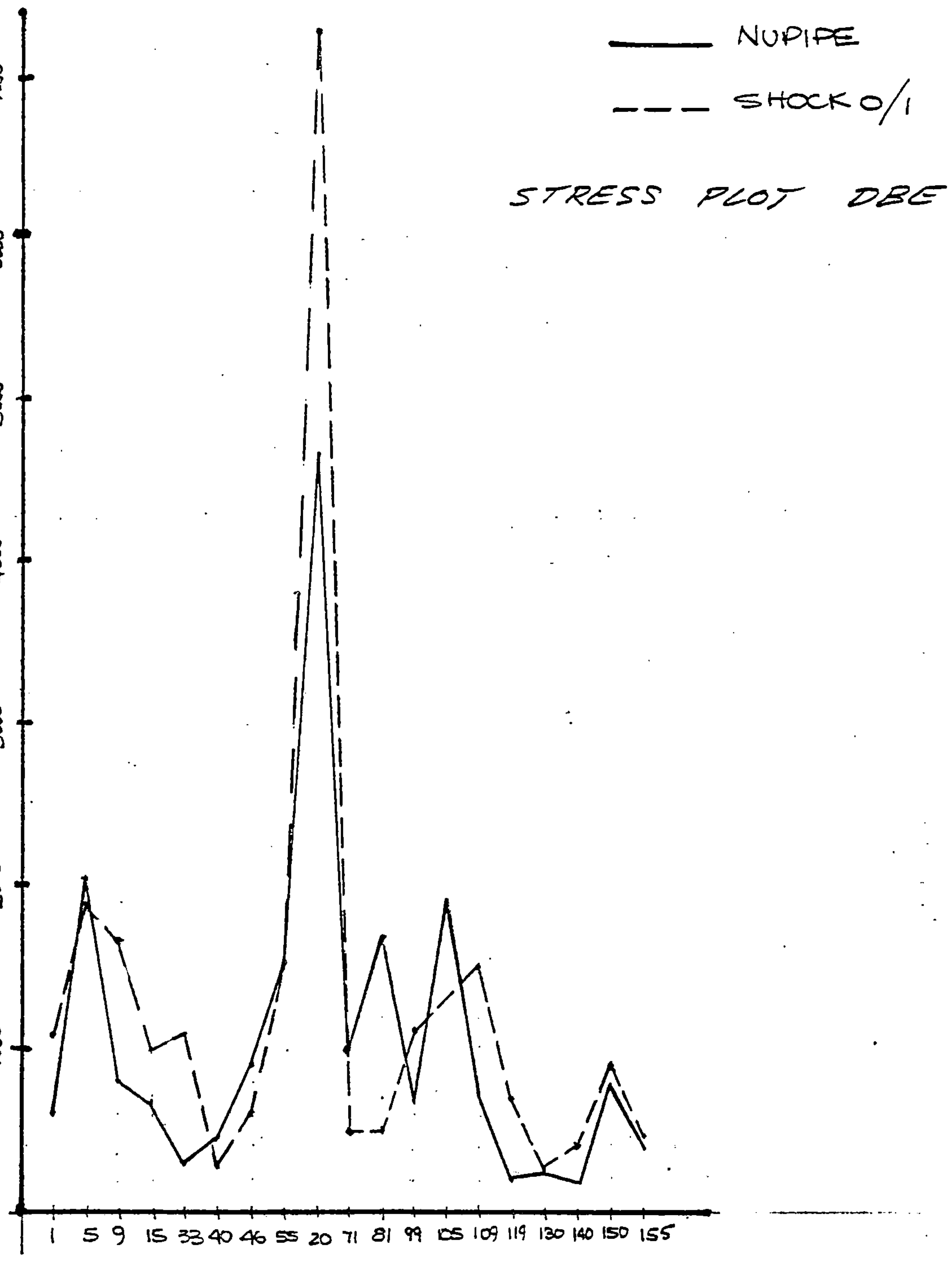
Problem No. 606 - Component Cooling Water

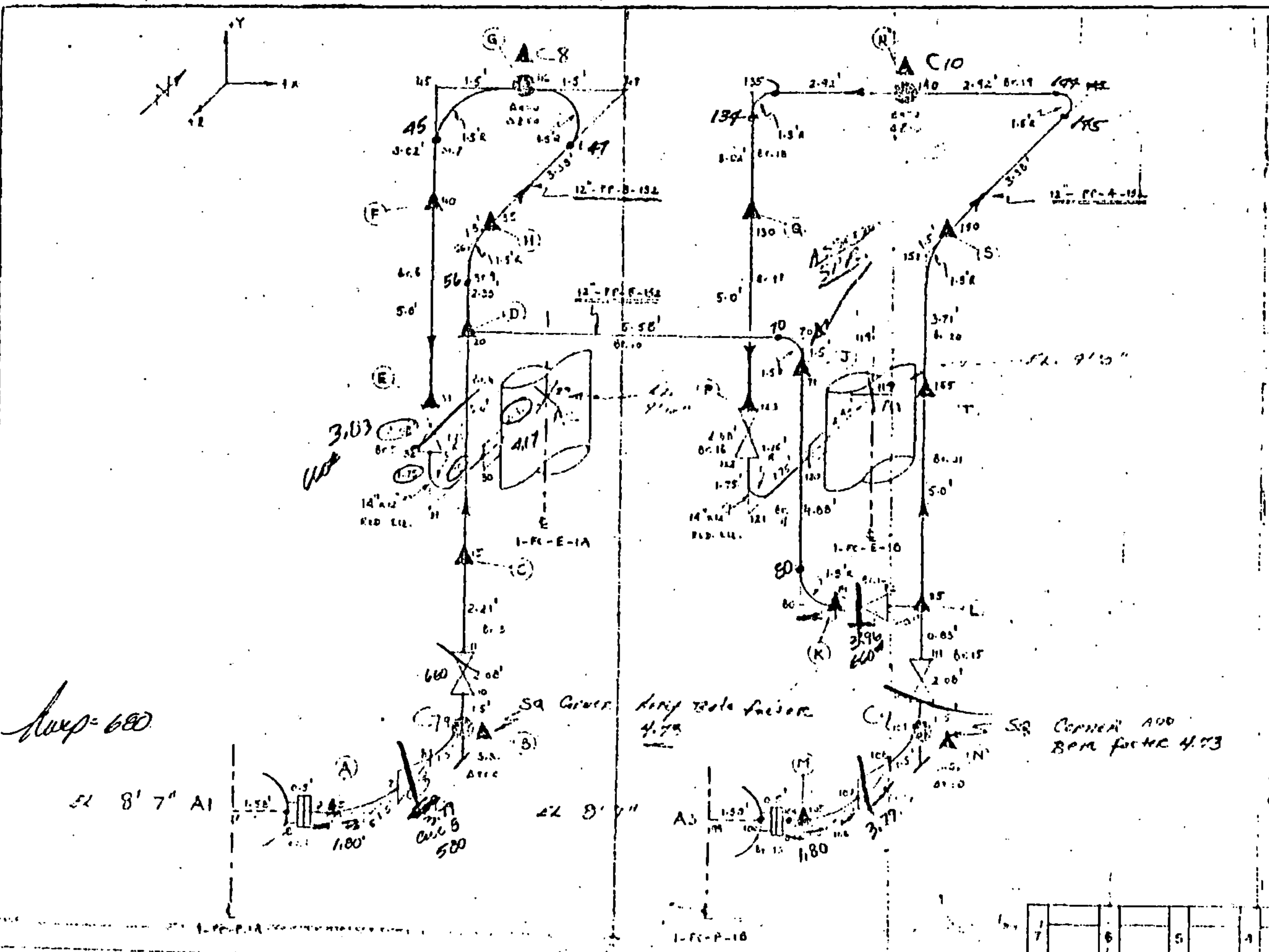
This example compares the results obtained from the original SHOCKO run with a reanalysis using the latest version of SHOCK1 (which was verified for the Maine Yankee Atomic Power Station). Although the natural frequencies

computed by SHOCK1 differ from those computed by SHOCK0, the resulting pipe 2.7
stresses are very similar. Where differences between the two runs occur, the 2.8
SHOCK0 results are the more conservative. The resultant forces and moments at 2.10
supports are also consistent between the two programs. Supporting data 2.11
follow.

SUBJECT / TITLE

SURRY - PROB 747 FUEL POOL COOLING 12" & 14" NPS





PROJECT NO. 741

PIPE DATA

SIZE	12.75" O.D ;	14.0" O.D.
WALL	0.18"	0.438"
MAT.	S.S.	
TEMP	170° F	
PRESS.	10 PSIG	
EXPN.	0.000911"/FT	
E_c	27.4 x 10^6 PSI	

LEGEND

- BRANCH NODE
- CONSTRAINT NODE
- FORCE NODE

Est Valve W/T - CK - 550 lbs, Gate - 668 lbs } H_2O included

XSE	1	12.75	.18	76.4	28.3	10.	8.
XSE	2	14.0	.438	122.	28.3	10.	8.
OPU	1	27.8	.0112	—	—	10.	

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CALCULATION SHEET

5010.61

J.O./W.O./CALCULATION NO.	REVISION	PAGE

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
R. G. Lorang 5-9-79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
SURRY - SPENT FUEL PIT COOLING	

PROB. 747

STRESS DATA (DBE)

NODE POINT	SHOCK O/I DATED 8-28-69	NUPIPE (OLD ARS) R1740688 5/9/79	
1	1286	623	
5	1868	2082	
9	1761	812	
15	1000	705	
33	1081	296	
40	293	447	
46	608	985	
55	1573	1524	
20	7316	4641	
71	483	993	
81	535	1691	
99	1212	669	
105	1539	1938	
109	1730	699	
119	737	202	
130	259	215	
140	401	196	
150	918	943	
155	435	406	
MAX	TOTAL STRESS	= D.L. + TH. +	DBE = 11,840
FROM	SHOCK RUN.	ALLOWABLE =	27,000 PSI

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R.G. Lorenz 5-9-79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
SURRY - SPENT FUEL PIT COOLING	

PROB. 747

SUPPORT LOADS

NODE POINT	SHOCK 0/1 DATED 8-28-69	NUPIPE (OLD ARS) R1740688 5/9/79	
9	Fy 564	FY 266	
46	Fx 562 Fz 808	Fx 280 Fz 250	
109	Fy 222	FY 278	
140	Fx 476 Fz 633	FX 227 FZ 129	

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R. G. Forony 5-9-79		

SUBJECT/TITLE: SURRY - SPENT FUEL PIT COOLING

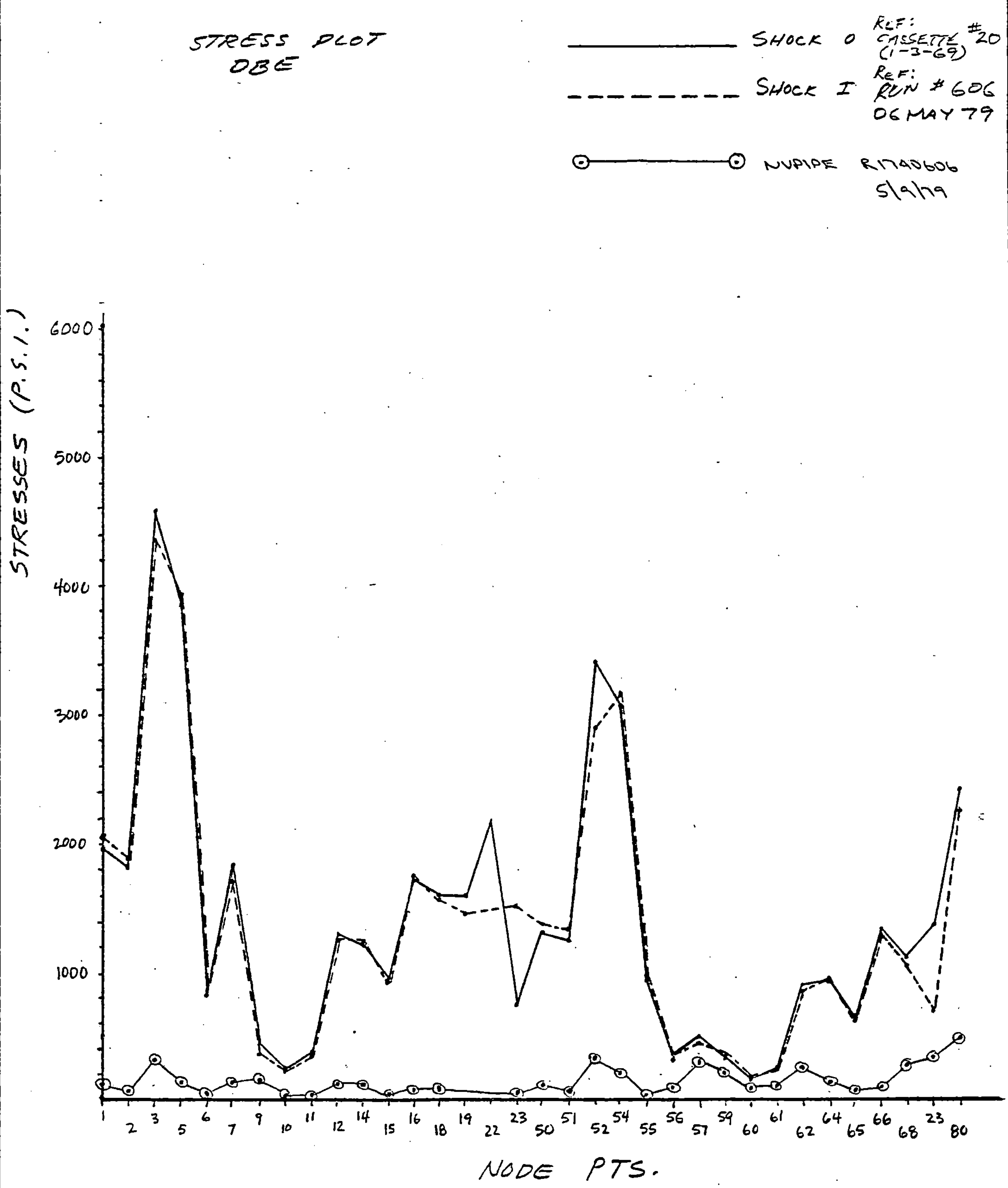
QA CATEGORY/CODE CLASS:

PROB. 747

SYSTEM FREQUENCIES

MODE	DYNAMIC FOR SHOCK Q 9/23/69	NUPIPE (OLD ARS) R1740688 5/9/79	
1	21.95	6.766	
2	11.94	7.633	
3	26.13	10.78	
4	24.05	13.51	
5	29.07	15.46	
6	27.83	16.46	
7	32.51	17.20	
8	31.55	18.62	
9	39.87	18.82	
10	38.8	20.91	
11	41.69	22.24	
12	47.57	23.45	
13	43.72	26.53	
14	57.63	30.45	
15	63.08	30.62	
16	59.13	32.81	
17	65.90	36.97	
18	65.28	41.12	
19	75.14	44.96	
20	73.47	45.38	
21	83.51	48.61	
22	90.68	52.03	
23	85.50	53.19	
24	93.33	56.37	
25	—		
26	—		
27	—		

SUBJECT / TITLE PROB 606 / REF. TAB FOR PROB 606



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5010.1D

Page No. ____ Preliminary ____ Item ____

Client VEPCO

Subject COMP. COOLING FAG EXC 1-CC-E-7A & 7B EP-16F

Based on SHOCK I COMPUTER RUN R174080 4/17/79

Location SURRY 1

Est. No. 606

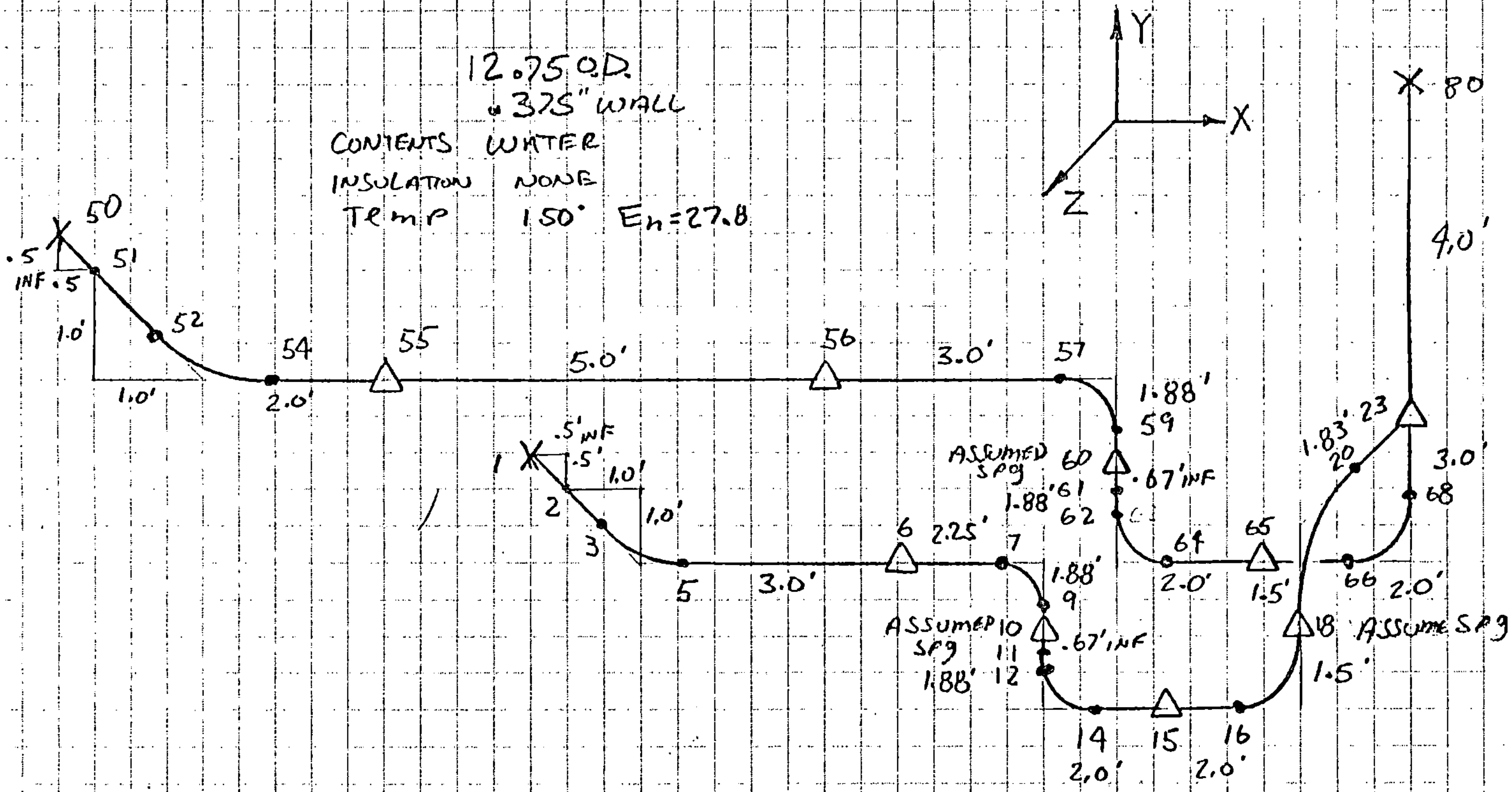
J.O. No. 12846.22

Date 5/8/79

By F. Sciarini

Checked ____ By ____

Revised ____ By ____



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CALCULATION SHEET

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J.O./W.O./CALCULATION NO.	REVISION	PAGE

PREPARER/DATE	REVIEWER/CHECKER/DATE	INDEPENDENT REVIEWER/DATE
[illegible] 5/9/79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
PROB. 606	

STRESSES

NODE POINT	Cassette #20 (1/3/69) SHOCK 0	RUN #606 5/6/79 SHOCK 1	DSE INERTIA R1740606 5/9/79 NUPIPE
1	1977	2047	151
2	1822	1881	131
3	4592	4311	311
5	3853	3921	163
6	855	838	66
7	1818	1728	157
9	439	355	161
10	255	228	72
11	392	374	72
12	1312	1289	150
14	1208	1228	150
15	940	932	61
16	1707	1729	117
18	1601	1591	138
19	1601	1446	NO POINT
23	783	(REIN) 23 1593	126
50	1326	1393	160
51	1264	1321	138
52	3404	2897	314
54	3076	3159	204
55	930	936	43
56	361	327	105
57	483	423	307
59	332	356	208
60	165	180	110
61	262	258	115
62	907	850	247

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[illegible signature] 5/9/79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
PROB 606	

STRESSES (CONT'D)

NODE POINT	Cassette #20 (1/3/79) SHOCK 0	RUN #606 5/6/79 SHOCK 1	DBE INERTIA R1790606 5/9/79 NUPIPE
64	948	955	171
65	643	611	102
66	1339	1312	111
68	1111	1045	271
23	1383	(REW.) 23 705	295
80	2406	2277	500

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[illegible] 5/9/79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
PROB. 606	

RESTRAINT LOADS

NODE POINT	Cassette #20 (1/3/69) SHOCK 0	RUN 606 5/6/79 SHOCK 1	DBE INERTIA NUPIPE R740606 5/9/79
1	FX= 572 lbs	503	93
	FY=301	219	97
	FZ= 880	943	85
	FR= 1092	FR= 1091	FR= 159
	MX= 2506 ft-lbs	2645	321
	MY= 6765	6983	303
	MZ= 196	361	332
	MR= 7217	MR= 7576	MR= 552
50	FX= 104	116	92
	FY= 26	68	84
	FZ= 393	452	98
	FR= 407	FR= 472	FR=159
	MX= 907	1007	250
	MY= 4756	4983	405
	MZ= 26	143	339
	MR= 4842	MR= 5085	MR= 584
80	FX= 207	306	98
	FY= 126	222	169
	FZ= 1701	1559	340
	FR= 1718	FR= 1604	FR= 392
	MX= 7078	6424	1734
	MY= 5006	4851	303
	MZ= 1434	2087	489
	MR= 8787	MR= 8316	MR= 1827

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[illegible] 5/9/79		

SUBJECT/TITLE	QA CATEGORY/CODE CLASS
COMPARISON OF FREQUENCIES PROB. 606	

FREQUENCIES

MODE	Cassette #20 (1/2/69) SHOCK 0	RUN # 600 (4/17/79) SHOCK 1	DRE INERTIA RMA0606 5/9/79 NUPIPE
1	32.53	22.96	14.265
2	36.98	26.00	15.787
3	40.70	34.24	20.483
4	45.81	35.79	23.288
5	50.51	50.24	30.577
6	67.75	51.57	32.956
7	70.18	57.63	42.491
8	82.70	89.02	58.646
9	85.37	119.17	66.081

APPENDIX F

SEISMIC CAPABILITY OF NUCLEAR PIPING

by

Robert L. Cloud
Robert L. Cloud Associates Inc.
Menlo Park, Calif.
May 1979

APPENDIX F

TABLE OF CONTENTS

LIST OF TABLES

1. INTRODUCTION

Before the development of the ANSI B31.7 Code for Nuclear Piping in the late 1960's and subsequent inclusion of piping under the provisions of ASME Code, Section III, all nuclear safety class piping was designed to meet the requirements of the ANSI (formerly USAS) B31.1 Code for Power Piping. As a result, many of the operating nuclear power plants in the United States today were designed and built to meet the provisions of the B31.1 code.

A general review of the methods applied to the seismic analysis of B31.1 safety class piping is given including reference to the historical evolution of these methods. The B31.1 code itself is discussed and it is demonstrated that, contrary to the belief of many, this code rests on an advanced technical base; sufficiently advanced in fact that very few changes had to be made, other than notation, to upgrade it to the B31.7 nuclear code and then to ASME Section III. The older piping code, unlike that for vessels, contained all the main features of current codes. Power plant, chemical plant, and refinery piping designed to B31.1 is not outdated and behaves very well in earthquakes.

The available data on performance of piping in seismic events are reviewed, and it is shown that engineered piping systems have performed extremely well in power plants that have experienced substantial earthquake-induced ground

motion. This outstanding performance has been exhibited even at plants in 1.27
which the piping systems were designed for seismic loadings far less severe 1.28
than current criteria would associate with the ground motions actually 1.29
experienced. It appears that piping systems engineered for the pressure and 2.1
temperature conditions typical of power plants are inherently resistant to the 2.2
effects of seismic-induced motions of their supports, whether or not such 2.3
effects are specifically addressed in the design process.

Power plant piping always has been designed to demanding standards. With the 2.5
introduction of nuclear power, these standards have been maintained and
strengthened to some degree. Thus, it is reasonable to expect that piping 2.6
systems in nuclear power plants that may experience earthquake motions will 2.7
perform as well as have the piping systems in non-nuclear power plants.

Early in the introduction of nuclear power, a major development effort began 2.8
in methods of dynamic analysis of structural response to earthquake ground 2.9
motion. This development was focused, almost exclusively, on systems (of 2.10
piping and supporting structures) conservatively assumed to respond in a 2.11
linear elastic mode. On the other hand, criteria of allowable piping stress, 2.12
which had been developed before seismic loadings were of major interest, 2.13
remained relatively unchanged by the substantial evolution of methods of 2.14
dynamic analysis. Stress criteria which had been rationally and 2.15

conservatively developed for relatively well-defined pressure-temperature-time conditions were applied with minor modifications to the less well-defined and very different earthquake conditions.

In consequence, the inherent seismic resistance of nuclear power plant piping systems designed earlier in the evolution of dynamic analysis may go unrecognized. Since systems for which earthquake effects were not even considered in design clearly have substantial resistance to such effects, it is unwarranted to only judge the seismic safety of a particular piping system by the particular ground motion specified and analytical methods used to predict response at the time of its design. To illustrate, there was a period when algebraic combination of intramodal seismic effects (of earthquake ground motions in differing directions) was common practice throughout the industry. This technique could either overestimate or underestimate earthquake loads on a piping system. In the limit, it might indicate essentially zero earthquake loading; i.e., equivalent to omitting earthquake from the design conditions. Obviously this is in no way the same as designing with zero resistance to earthquake. On the contrary, the system, designed to conservative criteria for pressure-temperature-time conditions, would certainly be resistant to substantial earthquake ground motion as may be seen by comparison with the power plants discussed herein.

2. SEISMIC ANALYSIS OF NUCLEAR PLANTS 3.5

Table F-1 shows a rough chronological development of some of the main features 3.7
of seismic design and analysis methods for nuclear plants. The first plants 3.9
were designed with static methods using lateral force coefficients as static
loads in the manner of various building codes. These plants were, in the 3.11
main, built in regions of low seismicity.

Dynamic considerations were introduced at about the time plants were built in 3.12
regions of higher seismicity. In recognition of the amplified response 3.14
possible when shaking motions have frequencies at or near the natural 3.15
frequencies of buildings and equipment, design ground response spectra were 3.16
introduced for design. Several papers that describe the derivation and 3.17
application of response spectra methods are contained in the section on 3.18
Seismic Analysis of Reference 1. This reference was compiled to provide 3.19
technical background for the advances and changes of various codes for design 3.20
and construction of pressure vessels and piping, especially for nuclear
applications. As such, the key papers that influenced the development of 3.21
nuclear seismic technology by seismic specialists such as Newmark, Hall, 3.22
Clough, Cornell, and others are reprinted conveniently in one place. 3.23

SURRY POWER STATION, UNIT 1

To obtain the seismic response of piping systems, it is necessary to study the 3.24
passage of ground motion through the soil, buildings, and equipment, all of 3.25
which cause modifications of the motion before it reaches the piping. 3.26
Originally, design response spectra were applied to piping in the simplest way 3.27
considering the first mode of each span and taking the response directly from 3.28
the ground spectrum. This approximation was an improvement over purely static 3.29
methods, but is quite simplified compared to later methods. 4.1

Subsequently in the 1960's, the effect of building motion on piping systems 4.2
was incorporated into the design process on an industry-wide basis although 4.3
the concept had been developed much earlier.[2] Conceptually, this is done by 4.5
analyzing the building for the effect of ground motion and developing new 4.6
spectra at the floors and walls of the building where piping is supported. In 4.7
practice this was done at first using records of actual earthquakes, Taft, El
Centro, etc, normalized to the design acceleration level chosen for the site. 4.8
The accelerations were applied to lumped mass building models in a time- 4.9
history fashion. At first, very few masses would be used to represent the 4.10
building, say less than 10. Also approximate methods were devised to obtain 4.11
the effect of building amplification on the design spectra[3] directly without 4.12
a time-history analysis of the building. Design floor spectra were developed 4.13
by these means and used for several plant designs.

In the 1970's, several major changes in methods of nuclear plant seismic 4.14
analysis were made. The key changes were a standardization of design ground 4.15
spectra, a requirement for three-directional analysis, and use of increased 4.16
damping values. The net effect was a more rational approach to seismic 4.17
analysis, but in any given case, computed seismic stresses tended to be 4.18
comparable to those obtained by the more approximate methods since the higher 4.19
damping compensated for the additional imposed motion. In any event, this 4.20
appendix is addressed more to B31.1 plants and subsequent developments will 4.21
not be discussed further.

3. PIPING ANALYSIS 4.23

Seismic analysis of piping systems in nuclear plants has also undergone an 4.24
evolution, outlined in Table F-2, consistent with the growth and development 4.27
of seismic methods for the plant as a whole. Early methods were based on 4.28
static analysis using a constant lateral force coefficient that was a
specified fraction of the total mass of that part of the piping under 4.29
consideration. As mentioned previously, when spectra were first used, 5.1
spectral accelerations consistent with span frequency were applied to the 5.2
piping system. These would be applied normal to the plane of the pipe, i.e., 5.3
in the worst 'direction' and combined with a vertical component.

Later, modal response spectra analysis was applied to safety class piping as 5.4
amplified floor spectra became available and were specified.[4] The 5.6
application of this approach varied between different organizations and with
the times. Although the fundamental steps and basic mathematics were 5.7
generally common to all, certain choices had to be made in combining responses 5.8
for each direction and each mode. These combinations are in a sense arbitrary 5.9
since the modal response spectra analysis method relinquishes time as a 5.10
parameter and relationships with respect to time, including phasing, are lost.

Some analyses have been done by evaluating each of three directions separately 5.11
and combining contributions from each direction. In many cases the horizontal 5.13
direction that causes the highest stress is combined with the vertical and
this 2-D planar response becomes the basis for evaluation. 5.14

The directional combinations have also been made in other ways. Since the 5.16
various response quantities are signed, algebraic summation of responses from
each direction within each mode has been done. Analyses have been completed 5.18
using other options, SRSS and absolute sum. The latter is probably overly 5.19
conservative.

After combinations have been made so that the response for each mode is 5.20
complete, the sum of all the modal responses must be obtained. Analyses have 5.22

been completed using several different methods of combining these responses.
The methods include a straightforward "square-root-of-the-sum-of-the-squares" 5.23
or SRSS, the absolute value of the single maximum modal response plus the SRSS 5.24
of the remaining modes, and other combinations including absolute values of
response of closely spaced modes plus the SRSS of the remaining modes. 5.25

The general impetus for the advance of seismic analysis and evaluation methods 5.26
came from a widely felt need both within the industry and regulatory agencies 5.27
to better understand seismic behavior of piping and equipment. As results 5.28
became available from development activities, they would be used for specific
plant analysis. The impetus for doing so would as often come from the utility 5.29
or the manufacturer as from the regulatory agency. It was a period of rapid 6.2
technical growth in which all groups concerned with the issue participated.

4. ANSI B31.1 CODE 6.4

Prior to the appearance of the ANSI B31.7 and the ASME Section III Code, all 6.5
safety class piping was evaluated according to the ANSI (formerly USAS) B31.1 6.8
Code for Power Piping. For the present discussion, the 1955 and 1967 versions 6.9
of this code are the issues of interest. There was little or no basic change 6.10
in B31.1 between the 1955 and 1967 versions. The 1955 version however was a 6.11
major departure from the previous issue of 1942 and supplements. In fact it 6.12

was in the 1955 version of B31.1 in which the basic rules and technical
philosophy were established for the design of power piping that are 6.13
essentially in existence today.

The advanced features and underlying technical sophistication of the B31.1 6.14
Code have gone relatively unnoticed in this era of rapid technical change and 6.15
innovation. The B31.1 approach, first established in 1955, contained 6.16
provisions for limiting the thermal strain range; recognized the self-limiting 6.17
nature of thermal stress; contained design rules for low cycle fatigue;
incorporated the maximum shear stress theory; and contained other 6.18
improvements. The ASME Boiler and Pressure Vessel Code contained none of 6.19
these features at that time. In fact it was not until the ASME III Nuclear 6.20
Vessel Code came out nine years later (1964) that these technical improvements
were applied to pressure vessels. 6.21

The fundamental intent of piping design lies in developing a system that has 6.22
sufficient flexibility but is sufficiently well controlled as discussed 6.23
further below. The concept of controlled flexibility is the key to successful 6.24
piping design. The Code recognizes this with an entire section devoted to 6.25
piping flexibility. The approach can be seen from the following, quoted from 6.26
paragraph 119.5 of the Code:

"Power piping systems shall be designed to have sufficient flexibility to 6.28
prevent pipe movements from causing failure from overstress of the pipe 6.29
material or anchors, leakage at joints, or detrimental distortion of 7.1
connected equipment resulting from excessive thrusts and moments.
Flexibility shall be provided by changes of direction in the piping 7.2
through the use of bends, loops, or offsets; or provisions shall be made 7.3
to absorb thermal movements by utilizing expansion, swivel or ball joints,
or corrugated pipe." 7.4

Explicit guidance is given to obtain balanced systems and to avoid problems of 7.6
strain concentration caused by non-uniform flexibility. In this connection 7.8
the concept of elastic follow-up is discussed. Design configurations 7.9
vulnerable to strain concentration are explained and cautioned against.

The phenomena of low cycle fatigue are accounted for in the design of B31.1 7.10
piping systems also. The basic allowable value of expansion stress is 7.11
multiplied by a factor which is related to the number of stress cycles. The 7.13
factor functions as an allowable stress reduction factor due to fatigue
service. The values of f are given below, where N is the number of stress 7.14
cycles.

N	f	
		7.17
7,000 and less	1.0	7.19
7,000 to 14,000	0.9	7.20
14,000 to 22,000	0.8	7.21
22,000 to 45,000	0.7	7.22
45,000 to 100,000	0.6	7.23
100,000 and over	0.5	7.24

These stress range reduction factors are based upon tests of full size pipes 7.28
made by Markl.[5] Not only is the basic fatigue process considered, but also 7.29
the deleterious effect on fatigue strength of various fittings, elbows, tees, 8.1
etc. This is accomplished by a requirement to multiply the basic components 8.2
of the expansion stress by "stress intensification factors" denoted by i. The 8.4
numerical values of i were also derived from full scale tests and are given in
the Code. The stress intensification factor bears only a nominal relation to 8.5
the stress concentration factors of elasticity; rather, i for a given fitting 8.6
is related to the ratio of the fatigue strength for the fitting to that of 8.7
straight pipe. It is in fact a fatigue strength reduction factor. 8.8

These various fatigue considerations have been condensed and codified in 8.9
apparently simple terms; but it is important to keep in mind that the approach 8.10

has a basis in full scale testing and, where simplifications have been made, 8.11
they are conservative. It is also true that even today with apparently 8.12
inexhaustible computer resources available, a single piping system is an 8.13
extraordinarily complex structure and in a single nuclear plant the safety
class piping might resolve down to as many as 90 to 100 piping problems. It 8.15
can be seen the simplifications are not only desirable, they are necessary.

Although an evidently straightforward consideration, the use of the maximum 8.16
shear stress instead of the maximum normal stress (as a limit of strength) is 8.18
worth mentioning. The advanced technical nature of B31.1 can be better 8.19
understood when it is realized that the widely accepted Boiler and Pressure 8.20
Vessel Code used the less accurate maximum principal stress theory up until
1964.

The Code has a brief paragraph that states that earthquake loads, when 8.21
applicable, must be considered; however, no explicit guidance is provided. 8.22
This matter would ordinarily be left to the designer. However, in nuclear 8.24
practice, the magnitude of design basis earthquakes is established as part of
the licensing process. Further, the methods used to seismically qualify a 8.25
plant are subject to regulatory body approval, so this combination of
requirements governed seismic design of B31.1 piping on nuclear plants. 8.26

As discussed previously, in all except the very early plants, a seismic ground 8.27
motion in the form of ground spectra and appropriate acceleration levels would 8.28
be specified. This motion would be applied to the buildings and 8.29
amplifications of the ground motion at various levels throughout the buildings 9.1
would be computed in the form of floor response spectra. It is the latter 9.2
that were used as design bases for nuclear piping.

The qualification of large piping systems of safety class categories is nearly 9.3
always done by means of a computer analysis. A dynamic analytical model of 9.5
the piping system is derived in which the mass of the system is concentrated
at a finite number of mass points and the flexibility of the system is 9.6
represented by springs connecting the masses. System damping is included as 9.7
viscous damping, normally with highly conservative numerical values of 0.5 or
1 percent of critical damping. The completed model is then analyzed for the 9.9
appropriate seismic spectral motion on the computer.

Usually, one amplified floor response spectrum is used as an input 9.10
acceleration at every point of support or connection to the building. This 9.12
simplification can be an important conservatism especially for piping systems
traversing different vertical levels or different buildings. The model of the 9.14
piping system is passed through the computer several times to account for all
directions of motion and both the operating and design basis earthquakes. 9.15

Inertia forces are developed first for all directions within each mode of 9.16
vibration, then the contributions of each mode are combined to obtain the 9.17
total force. A current controversy lies in the fact that force combinations 9.18
within each mode were in some cases combined algebraically so that some loads 9.19
would subtract from the total. The alternative would be to combine forces in 9.20
such a way that subtraction could not occur, which is the case if an SRSS
approach is used. 9.21

Effects of the inertial forces are combined with effects from relative 9.22
building displacements, gravity (weight) effects, and internal/external 9.23
pressure loadings on the pipe.

When load combinations are complete, bending moments and stresses in the 9.24
piping system are computed according to B31.1 equations. Basically, twice the 9.26
maximum shearing stress in the pipe due to bending and tension is computed and
limited to 1.2 S_h for the OBE and 1.8 S_h for the DBE in a manner very 9.27
comparable to ASME III today. S_h is the tabulated value of allowable stress 9.28
as provided by the Code, in the hot condition. In B31.1, S_h is based on the 9.29
lower of 5/8 Yield Strength or 1/4 Ultimate Strength at operating temperature,
except certain austenitic materials are permitted S_h values at operating 10.1
temperatures up to 90 percent of yield strength because of the greater 10.2
toughness and ductility of these materials. These values of allowable stress 10.3

are the lowest in use for any piping in the United States. ASME III Class 1 10.4
nuclear piping has higher allowables, as does B31.3 Refinery and Chemical
Plant Piping. B31.4 and B31.8 for Gas and Oil Transmission piping 10.5
respectively permit allowable stresses up to 72 percent of the ultimate 10.6
strength. When nuclear plant piping was moved under the aegis of ASME 10.7
Section III, Safety Class 3 and 2 continued to be designed by B31.1; however 10.8
the allowable stress for the faulted plant condition was raised to 2.4 S_h. 10.9
Mention is made of certain of these facts as an observation of the 10.10
conservative nature of the B31.1 Code even when compared to other codes that 10.11
use the same calculational basis.

The method of stress evaluation just described is a simplified overview of the 10.12
acutal process. One of the more troublesome aspects of the work is accounting 10.13
for elbows, tees, attachments, and other stress raisers. This is accomplished 10.15
by a mandatory multiplication of the stress at points of concentration by
tabulated "stress intensification factors" or i factors. 10.16

5. B31.1 AND LATER CODES 10.18

The first version of the B31.1 Code was published in 1935, and a revised 10.19
second edition was published in 1942. Then a third edition was issued in 10.22
1951. This was a period of rapid development in piping design methods and it 10.23

was found desirable to publish another revised edition of the Code in 1955. A 10.25
brief history is given in the foreword to the 1955 edition of B31.1. What is 10.26
not mentioned there, however, is that the 1955 edition of the piping code had
several far reaching engineering improvements, which have been mentioned 10.27
earlier herein.

The development of the 1955 edition and some of the changes therein are 10.28
discussed in References 6, 7. Subsequently, a new edition was published in 11.1
1967, and although there were a number of changes and minor revisions, no new 11.2
concepts were introduced.

In 1969 the ANSI B31.7 Code for nuclear piping was first published. The basic 11.4
philosophy of this code was to have nuclear primary system piping designed to
similar criteria as nuclear primary system vessels. This required B31.7 to 11.6
adopt similar approaches to the different possible types of failure and
provide comparable margins with Section III of the ASME Code. The modes of 11.8
failure for which protection is provided explicitly by the stress evaluation
procedures of Section III are bursting, excessive plastic deformation, 11.9
progressive distortion, and thermal and mechanical fatigue failure. Of course 11.11
other possible types of failure are considered in other areas of the Code,
specifically in materials selection and fabrication guidelines. 11.12

The obvious approach to develop a piping code comparable to Section III for 11.13
vessels was to attempt to adapt the existing B31.1 Code, which was the 11.14
approach taken. However, as it turned out, the B31.1 Code already contained 11.15
almost every provision of Section III, in a different format perhaps, but all 11.16
the basic concepts were in place. The dêvelopment of B31.7 then was a matter 11.17
of recasting the original provisions of B31.1 into Section III format. Only 11.19
one technical addition was required that could be considered a new concept,
and that was the addition of consideration for radial temperature gradients 11.20
through pipe walls. In certain situations or processes this could be an 11.21
important consideration, but in nuclear plants it rarely determines the 11.22
acceptability of piping systems. The net result is that B31.7, even though 11.23
different in appearance and permitting slightly thinner pipe walls due to 11.24
higher Section III S values, was not fundamentally different from the
B31.1 Code. This was especially true in the most important aspects of piping 11.25
design, the limitation on the main expansion strain range and thermal fatigue 11.26
considerations. The stress indices, C_2 and K_2 of B31.7 (and Section III), are 11.27
even generally related to the old i indices of B31.1. 11.28

$$C_2 K_2 = 2i$$ 12.1

This relationship and other background on the development of the current ASME 12.4
Section III Piping Code are in a forthcoming edition of the ASME Criteria 12.5
Background Booklet.[8]

The essential point of the preceding discussion has been to make clear that 12.6
safety class piping designed to meet the requirements of the older ASA B31.1 12.7
Code would almost without exception also meet the requirements of the latest 12.8
version of the ASME Code. A little more needs to be said about seismic design 12.9
however. The B31.1 Code of 1967 and 1955 clearly spells out that seismic 12.10
stresses are to be considered but does not say how. For nuclear plants built 12.12
to those codes, however, this is not significant for present purposes since 12.13
rigorous seismic analysis was completed for these plants to satisfy licensing
requirements. 12.14

6. SEISMIC PERFORMANCE OF POWER PIPING 12.16

Although there appear to be no controlled experiments of seismic performance 12.17
of actual piping systems, there is, nevertheless, a surprising amount of very 12.18
interesting data on the response of power piping to actual earthquakes. In 12.21
the following, power plant behavior in several recent earthquakes, Managua
1972, San Fernando 1971, Alaska 1964, Kern County 1952, and Long Beach 1933, 12.22

is discussed. No attempt has been made to sort or classify the observations, 12.23
rather all significant data that could be found in a short time are reported. 12.24

Possibly the most interesting of the observations are those pertaining to the 12.25
Kern Steam Station in the Kern County earthquake, and the Enaluf Steam Plant 12.26
in the Managua earthquake. Both these plants were designed by conventional 12.27
procedures, both underwent severe ground shaking, and neither suffered any 12.28
failures of the piping systems. The maximum ground accelerations were 12.29
estimated to be as high as possibly 0.6 g at Enaluf, which was adjacent to the 13.1
main fault causing the quake, and about 0.25 g for the Kern County Steam
Plant. Time and again it is seen that piping systems correctly designed for 13.2
normal service are relatively impervious to earthquake damage. The basic 13.4
concept of controlled flexibility built into power piping renders these
systems more resilient than the buildings from which they are supported. 13.5

6.1 Long Beach Steam Station 13.7

This station was located on Terminal Island in Long Beach, California, about 13.8
4 miles from the fault that caused the Long Beach earthquake on March 10, 13.9
1933. This earthquake was of magnitude 6.3 and caused accelerations at the 13.11
site of the steam plant estimated to be about 0.25 g. Damage in Long Beach 13.13

itself was very extensive, but there were no actual accelerometer records of
the earthquake. 13.14

At the steam station site there were actually three independent plants. 13.15
Plant 1 consisted of one unit and was built in 1911. It was either out of 13.17
service or in intermittent service in 1933 and the building was severely
damaged in the earthquake. Plant 2 consisted of two units and was built in 13.19
1922. Plant 3 consisted of three units and was built in 1928. This and 13.21
subsequent information was obtained from W.F. Swiger of the Stone & Webster
Engineering Corporation, designers and builders of the plant. For other 13.23
reasons it was necessary to re-examine the design of the plant at a later time
and it was determined the plant structures were designed for lateral static 13.24
forces of 0.2 g. Foundations of both plants were heavily reinforced concrete 13.25
mats supported by wooden piles 50 to 60 feet long driven to hard sands. No 13.27
information is available on seismic design of the piping and equipment, but
considering the state of the art it is probable that either the 0.2 g static 13.28
design was used, or else seismic design was not considered. 13.29

Neither plant, that is to say, none of the five units, suffered any 14.1
significant damage. Some minor damage such as to lighting fixtures was 14.2
reported; however, the steam plants either operated through the earthquake or 14.3
were shut down due to loss of load and were back in operation the same day. 14.4

The important point is that five steam units designed with at most static 14.5
methods to a g level (0.2) probably lower than actually experienced (0.25) 14.6
were undamaged and, in particular, no piping was damaged.

6.2 Kern County Steam Station 14.8

This oil-fired 60 MW steam plant was designed and built in 1947-8. It is 14.10
located on the Kern River near Bakersfield, California, about 25 miles from
the epicenter of the July 21, 1952 Kern County earthquake. 14.12

This earthquake, sometimes referred to as the Taft, the Tehachapi, or the 14.13
Arvin-Tehachapi, was of magnitude 7.7. It was the most severe earthquake 14.15
recorded in the continental United States since that of 1906 in San Francisco. 14.16
It occurred along the White Wolf fault south and east of Bakersfield. Damage 14.18
was extensive in Bakersfield and to oil production facilities in the area and
to the Southern Pacific Railroad. The railroad tunnel near Bealville crossed 14.20
the fault and was destroyed.[10]

The structures of the plant were designed for 0.2 lateral load on a static 14.22
basis with stress limits increased by 0.33 for combined dead, live, and 14.23
earthquake loadings. Foundations are soil bearing footings at shallow depth. 14.24

Anchorage systems of all major equipment including switchgear were carefully 14.25
reviewed for resistance to lateral loads. 14.26

This is one of the first electric power plants to have piping designed by 14.28
dynamic analysis. The Biot[2] smoothed response spectrum was used for the 14.29
design of the main steam and boiler feedwater piping. The response spectrum 15.2
was normalized to 0.1 g at ground level and 0.3 g at the top floor of the 15.3
buildings, with linear interpolation at other levels. In this way an 15.4
amplified response spectra was available at every floor, even though it was of 15.5
narrow band and heavily damped compared to spectra used for nuclear plants.
The spectra were applied for the steam and feed lines by calculating the first 15.6
natural frequency of each span of pipe considered as a simply supported beam, 15.7
then applying the appropriate lateral g force. Based on the dynamic analysis 15.9
of the main piping, psuedo-static g loads were developed for other piping 15.10
systems. These loads were also used to design guides and stops and to find 15.11
loads acting on the supporting structure. It is of interest to note that some 15.13
guides and stops on the main steam line had gaps or rattle space of as much as 15.14
2 inches[9].

An acceleration record obtained at Taft, California, was farther from the 15.16
epicenter than the Kern County Plant. Maximum acceleration recorded at Taft 15.17
was 0.17 g and it was estimated that ground acceleration at the plant site was 15.18

a very substantial 0.25 g. The plant operated through the earthquake with no 15.19
significant damage. It was shut down after the earthquake due to loss of load 15.20
but was returned to service in a few hours. There was some minor damage to 15.21
oil tank seals and a small house turbine thrust bearing, but no damage at all 15.22
to piping systems. This is a very clear and graphic example of the almost 15.23
complete seismic protection that is provided by even the most rudimentary 15.24
seismic design procedures (by today's standards). Of course, there was even 15.25
greater inherent reserve in the piping systems due to their natural controlled 15.26
flexibility.

6.3 Alaska Earthquake of 1964 15.28

This earthquake of 8.4 magnitude was the largest recorded earthquake of modern 16.1
times. It was centered east of the city of Anchorage, near the town of 16.3
Valdez. There was widespread destruction throughout the area, not only from 16.4
earth vibration, but from the tsunami, the failure of poor soils, and fire. 16.5

Some observations by knowledgeable engineers of power piping are available, 16.6
but there is more detailed information that is yet to be obtained. In a panel 16.8
discussion on the Nuclear Piping Code, observations were noted of power piping 16.9
behavior by an experienced piping engineer with a leading
Architect/Engineer[11]. Mr. Fred Vinson reported that he reviewed the damage 16.10

at two power stations immediately following the earthquake. The power station 16.12
at an air base in the earthquake zone had no damaged piping although there 16.13
were some "bent hanger rods," damaged lighting fixtures, and an overturned
control panel due to absence of anchor bolts. 16.14

A second power plant in the earthquake zone incurred more damage to the plant, 16.15
although there was no failure of power piping. There were failures of some 16.17
equipment supports made of malleable iron, and an ash handling line connected 16.18
with patented couplings is reported to have failed due to improper support.

The significant finding of the observations of Reference 11 is that two power 16.20
plants rode out the Alaska earthquake with no failures of the power piping,
even though the exact g levels at the sites were not reported and the design 16.21
basis was not given other than to say "very little was done in the way of 16.22
seismic design for the protection of anything." 16.23

A brief mention is made in Reference 10 of the Chugach Electric Company plant 16.24
in Anchorage. This fossil-fueled plant of about 50 MW was built between 1949 16.26
and 1957. The plant was designed to 0.1 g by the Uniform Building Code. The 16.28
buildings were of steel frame construction with corrugated panel walls. There 16.29
was no damage in the turbine room nor to piping and critical equipment. There 17.1
was minor damage in the boiler room consisting of bending of some bracing 17.2

members and appreciable damage to framing supporting the coal bunkers. Many piping hangers on the main steam lines were broken, but the piping itself was undamaged. The plant was returned to service at full power in less than 10 days.

The consulting firm of Ayres and Hayakawa of Los Angeles was asked to review all nonstructural damage to buildings due to the Alaska earthquake as part of the investigation performed by the National Academy of Sciences at the request of President Lyndon Johnson. In their report[12] power plants were not discussed separately, rather observations of piping systems of all types were discussed on a generic basis. The discussion is based on a study of large modern structures located, with few exceptions, in Anchorage.

The reference report addresses general piping systems of all types, but mainly that required in modern buildings. With the exception of certain fire protection piping, none was seismically designed. Because of the broad basis of the report, the following paragraph is quoted directly from the section entitled "Piping Systems."

> "The overall damage to piping systems was surprisingly low. Many instances were reported where piping systems remained intact, despite the significant structural and nonstructural damage suffered by the building.

For example, the plumbing pipes in the Enlisted Men's Service Club at Fort 17.24
Richardson remained standing after the earthquake although the walls
around them collapsed. Contractors also reported that most systems were 17.26
put back into service when pressure-testing revealed no leaks." 17.27

The general conclusion was that piping systems are basically earthquake 17.29
resistant. Failures occur if at all at threaded fittings. Welded steel pipe 18.2
does not fail. One instance of power piping failure was noted. Small steam 18.4
pipe drain lines anchored to building walls were torn from the steam line as 18.5
it responded to the earthquake at the Fort Richardson power plant. This is 18.6
the type of unbalanced design warned against in the piping code. Properly 18.7
detailed systems had no problems.

6.4 San Fernando, California, 1971 18.9

The San Fernando Earthquake of 1971 was centered in the northern part of the 18.11
San Fernando Valley. Ground accelerations of 0.1 to 0.19 g were recorded in 18.13
Los Angeles at distances of 35 km and 0.37 g at Lake Hughes, 25 km from the 18.14
epicenter. Figure F-1 shows recorded g levels for the 1971 earthquake at 18.15
various locations near Los Angeles. There was severe damage to a number of 18.17
structures in the valley.

The Valley Power Plant is a fossil fuel plant with three units on the site 18.19
located 2.8 miles from the line of surface rupture (Lakeview Segment) of the
primary fault break. Accelerations at the site were estimated to be in excess 18.20
of 0.25 g based upon the location of various recordings. The station was 18.22
designed to 0.2 or 0.25 g although actual details are not known.

In any event there was no damage to the plant. It was tripped off the line by 18.24
action of sudden pressure relays and loss of load, but was back on the line 18.25
inside of 2 hours[13]. There was significant motion of the piping and seismic 18.26
holddown bars came into play[14], but other than insulation, the piping itself 18.27
was undamaged. This is a graphic example of the basic point that well 18.28
designed piping to regular commercial practice is highly resistant to 18.29
earthquake damage. Piping designed to nuclear standards is that much more 19.1
resistant.

There were other power plants in the area at Playa del Rey, San Pedro, and 19.2
Seal Beach that were not as close to the epicenter as the Valley Plant and 19.3
none of these were damaged. The San Fernando Power Plant is an old hydro 19.4
plant built in 1921 and there was a structural failure of the building which 19.5
led to a penstock failure. There were numerous failures of electric 19.6
transmission facilities due to cracking of porcelain bushings and movement of 19.7

poorly anchored equipment. There were no power piping failures in the San Fernando earthquake.

6.5 Managua, Nicaragua, 1972

An earthquake of magnitude 7.5 struck Managua on December 25, 1972. There was much damage and great loss of life. The loss of life was largely unrelated to damage of industrial buildings and facilities since the earthquake occurred near midnight. A report on the damage was sponsored by the National Science Foundation and several professional societies together with the Ministry of Public Works of Nicaragua[15].

Figure F-2 taken from Reference 15 shows the fault lines along which movement occurred running through the city of Managua. The location of two industrial facilities, the ESSO refinery and the ENALUF Power Plant, are also noted. The earthquake response of these two facilities will be discussed since they contain industrial piping systems of interest for present purposes.

A complete accelerograph record was obtained at the ESSO refinery. The peak measured acceleration was 0.39 g E-W and 0.34 g N-S. The design of the refinery met provisions of the Uniform Building Code for 0.2 g, including tall fractionating towers, some of which exceeded several hundred feet. There was

almost no damage at the refinery and none to the piping systems. Some piping 20.1
jumped out of saddle supports and was pushed back into place. The facility 20.2
was shut down for an inspection but was operating at full capacity within 20.3
24 hours even though there was a loss of offsite power. The refinery provides 20.4
a clear example of the seismic capacity of welded steel pipe that has been 20.5
designed for seismic conditions, albeit statically.

Based on the earthquake magnitude, acceleration record at the refinery, and 20.7
the location of the ENALUF Plant immediately adjacent to the causative fault,
it is probable this plant experienced accelerations on the order of 0.6 g. 20.8
The power plant consists of three oil-fired units, one of 50 MW and two of 20.9
20 MW. All three units were taken off-line by protective relays. The plant 20.11
suffered some damage but none to the piping systems. It was one of the first 20.12
industrial facilities restored to service after the earthquake. One unit was 20.13
operating in two weeks, the second in three weeks. Operation of Unit 3 was 20.14
delayed due to turbine problems.

The specific damage to the three units is listed in Table F-3. Note that no 20.16
damage occurred to the piping, and that many of the problems resulted from 20.17
absent or inadequate anchors. For example, turbine bearings were lost because 20.18
emergency dc oil pumps were inoperative due to the batteries tumbling out of 20.19
their racks.

The basic facts about the power piping are that, with unknown seismic design 20.21
applied, but certainly less rigorous than used for nuclear plants, the piping 20.22
survived site accelerations on the order of 0.6 g with no failure. Modern 20.23
welded steel piping with built-in controlled flexibility is inherently highly
resistant to earthquake damage. 20.24

7. BASIS FOR SEISMIC CAPABILITY OF POWER PIPING 20.26

In the previous section the performance of piping systems in power plants and 20.27
a refinery during actual earthquakes was reviewed. It was shown that there 21.1
were no piping failures even though ground accelerations up to 0.6 g were
experienced and seismic design was usually based on static analysis to the 21.2
lower value of 0.2 g. This approach to seismic design would be considered 21.3
rudimentary by nuclear standards.

In the following paragraphs, the probable reasons for the excellent 21.4
performance of piping systems in earthquakes is explored. The fundamental 21.6
seismic capability of piping systems apparently derives from three sources:

1. The power piping design and construction code, ANSI B31.1, is quite 21.8
conservative.

2. Designs that are successful for thermal expansion also provide good 21.9
seismic capability. 21.10

3. The large damping factors that become operative in severe shaking are 21.12
neglected in normal design practice.

Taking the above factors one at a time, it is first noted that in Sections 4 21.14
and 5 of this report, the B31.1 code for power piping was discussed and it was 21.15
shown that the nuclear power piping codes derived from B31.1 have much in 21.16
common with the parent code. However the basic conservatism was not covered 21.17
in detail. There is substantial margin provided by the design rules of B31.1. 21.18
The average stress in the pipe wall due to the design pressure is limited to 21.19
1/4 of the tensile strength of the steel. Thermal expansion of the pipe may 21.21
cause stresses due to restraint of expansion, but these are displacement or 21.22
strain controlled. That is, the strains will not become larger than indicated 21.23
by the associated temperature and will always be stable, unlike a dead load or 21.24
pressure stress. The strain range due to thermal expansion is limited to a 21.25
very small fraction of the strain capability of the pipe, considering the 21.26
repetitive nature of the thermal expansion loading. The code attempts to 21.27
consider all the categories of loading that a piping system will experience
and maintain the pipe in a state of small strain, including the effects of 21.28
stress intensification at elbows and tees.

However, the significance of the rules for fabrication and construction given 21.29
by the code tend to be overlooked in discussions of design capability. The 22.2
provisions for sound weld design, weld qualification tests, heat treatment,
inspection, and tests all combine to produce piping systems as sound in the 22.3
field as they appear on the drawing board. The significance of the 22.5
requirements for construction becomes even more visible as one reads the 22.6
references that describe the results of field inspections following
earthquakes. Occasional references to failures of piping in plumbing systems 22.7
are made, e.g., Reference 12. In these cases the problems invariably occur at 22.8
threaded joints and occasionally at flanged joints. Wrought iron and cast 22.9
iron pipe also perform poorly in earthquakes. However, properly designed and 22.10
hung welded steel power piping did not fail in even very severe earthquakes.

Evidently the controlled flexibility built into well designed piping systems 22.11
imparts substantial seismic capability also. If, in the design, provision is 22.12
made for pipe displacement due to thermal growth, the pipe is then later
untroubled by forced seismic displacements. The provision for flexibility may 22.14
be the most important aspect of seismic design and is an especially important
consideration in selecting and sizing pipe hangers. It is significant that 22.16
piping hangers were reported on one occasion to have failed[12], but the piping 22.17
itself did not. Sound piping material can undergo cyclic strain of several 22.18

percent for the limited number of cycles that would be imposed by an 22.19
earthquake.

The damping associated with severe shaking is one of the most important 22.20
conservatisms in existing approaches to nuclear piping design. Normally 22.22
viscous damping is assumed with damping factors of 1/2 or 1 percent of
critical damping. In a large earthquake however, several energy dissipating 22.23
mechanisms will become operative; ordinary material damping, impact damping, 22.24
friction or coulomb damping, and plastic deformation when there are large pipe 22.25
motions. Taken together, it is clear that damping ratios much greater than 22.26
design values can be expected.

Bohm[16] has presented a reasonably comprehensive survey of damping in reactor 22.27
systems. Unfortunately, the data available were all for relatively small 22.28
deflections. However, there is a clear correlation of damping values with 22.29
amplitude of vibration. Figure F-3, taken from Reference 16, shows the 23.1
increase in damping with deflection for the data obtained from tests of full 23.2
scale nuclear plants. There are also some data from the San Onofre Nuclear 23.3
Plant in the El Cajon and San Fernando earthquakes. 23.4

It is interesting to note that the San Fernando earthquake produced ground 23.5
accelerations of 0.018 g maximum at the San Onofre site and damping of between 23.6

2 and 4 percent for deflections of about 0.03 inches were measured by plant 23.7
instrumentation on the primary equipment. Damping of 3 to 8 percent was 23.8
reported to have been measured in pluck tests at the Tsuruga Nuclear Plant.
In general, damping that is much higher than the design value was measured in 23.9
several tests at very small deflections and it increases with amplitude of 23.10
deflection. Extrapolating the curve of Figure F-3 to 0.5 inch deflection 23.11
yields 10 percent damping. 23.12

As plasticity develops in the piping even in small amounts, damping ratios of 23.13
10 percent and higher are definitely to be expected. In fact, there is a 23.15
major project underway at the present[17] to develop seismic restraints based on
cyclic plasticity of the supports. The essential quality of the relationship 23.17
between damping, acceleration level, and damage is that damage to piping does 23.18
not increase proportionately with input acceleration levels and this is due in
large part to increases in damping levels as deflections increase. 23.19

8. CONCLUSIONS AND IMPLICATIONS FOR MODERN NUCLEAR PLANTS 23.21

The evolution of seismic design methods in nuclear power plants has been 23.22
reviewed together with the development of the piping codes. It was shown that 23.25
nuclear plants that meet the older B31.1 code will more than likely also
satisfy the new nuclear codes that have better quantified conservatism. 23.26

Available data on the actual seismic performance of power piping systems were 23.27
reviewed. It was shown that operating power plants do indeed have very high 23.28
levels of seismic capability. Of the several plants that sustained severe 23.29
ground motion from 0.2 to 0.6 g, there were no failures of welded steel power 24.1
piping. Considering the magnitudes of the earthquakes and the variability of 24.2
the design practices, this is an excellent record and can only have been made 24.3
possible by the natural resiliency of power piping. 24.4

The probable reasons for this natural resiliency were discussed next. It is 24.6
believed that the main reasons are: first, the substantial conservatism of the
Code for Power Piping, B31.1, including the provisions for materials, 24.7
fabrication, and construction; second, that design of piping for thermal 24.8
expansion provides inherent seismic capability; and third, that damping 24.9
increases very rapidly with deflection levels. The large damping factors 24.10
prevent buildup of seismic disturbances in resonant systems. It is believed 24.11
these reasons explain the remarkable performance of piping systems in
earthquakes.

Based upon the foregoing observations, it is very improbable that piping- 24.12
related safety problems would occur in nuclear plants in the eastern United 24.13
States due to seismic disturbances. These plants have maximum ground motions 24.14
of 0.15 g; they have been designed by dynamic analysis; and all safety piping 24.15

systems have been specifically scrutinized. Contrast this situation with say the Kern County plant where 0.25 g was actually experienced and explicit analysis was performed only on the steam and feed lines; or the ENALUF plant which was probably designed statically and experienced perhaps 0.6 g. The contrast is simply too great; piping failures of nuclear safety systems should not result from earthquakes in the United States.

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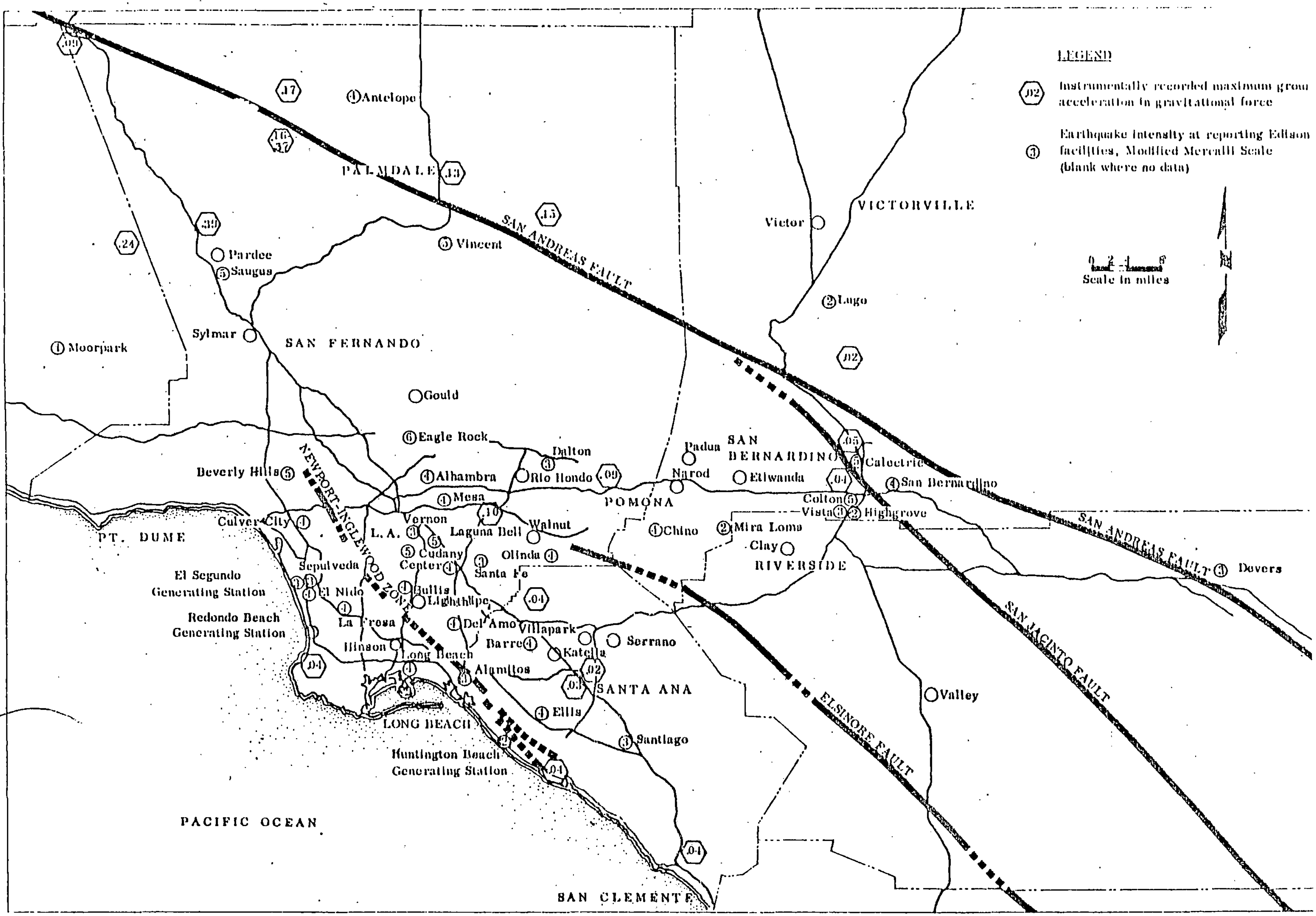

LEGEND
Instrumentally recorded maximum ground acceleration in gravitational force
Earthquake intensity at reporting Edison facilities, Modified Mercalli Scale (blank where no data)
Scale in miles
SAN ANDREAS FAULT
SAN JACINTO FAULT
ELSINORE FAULT
NEWPORT-INGLEWOOD ZONE
PALMDALE
VICTORVILLE
SAN FERNANDO
SAN BERNARDINO
POMONA
RIVERSIDE
SANTA ANA
LONG BEACH
L. A.
PT. DUME
PACIFIC OCEAN
SAN CLEMENTE
Antelope
Vincent
Pardee
Saugus
Sylmar
Moorpark
Gould
Eagle Rock
Dalton
Alhambra
Rio Hondo
Beverly Hills
Mesa
Culver City
Vernon
Laguna Bell
Walnut
Cudany
Center
Santa Fe
Olinda
Sepulveda
El Nido
El Segundo Generating Station
Redondo Beach Generating Station
La Fresa
Hinson
Long Beach
Bullis
Lighthipe
Del Amo
Villapark
Barre
Katella
Serrano
Alamitos
Ellis
Santiago
Huntington Beach Generating Station
Victor
Lugo
Padua
Narod
Etiwanda
Calectric
San Bernardino
Colton
Vista
Highgrove
Chino
Mira Loma
Clay
Devers
Valley

Fig. F-1

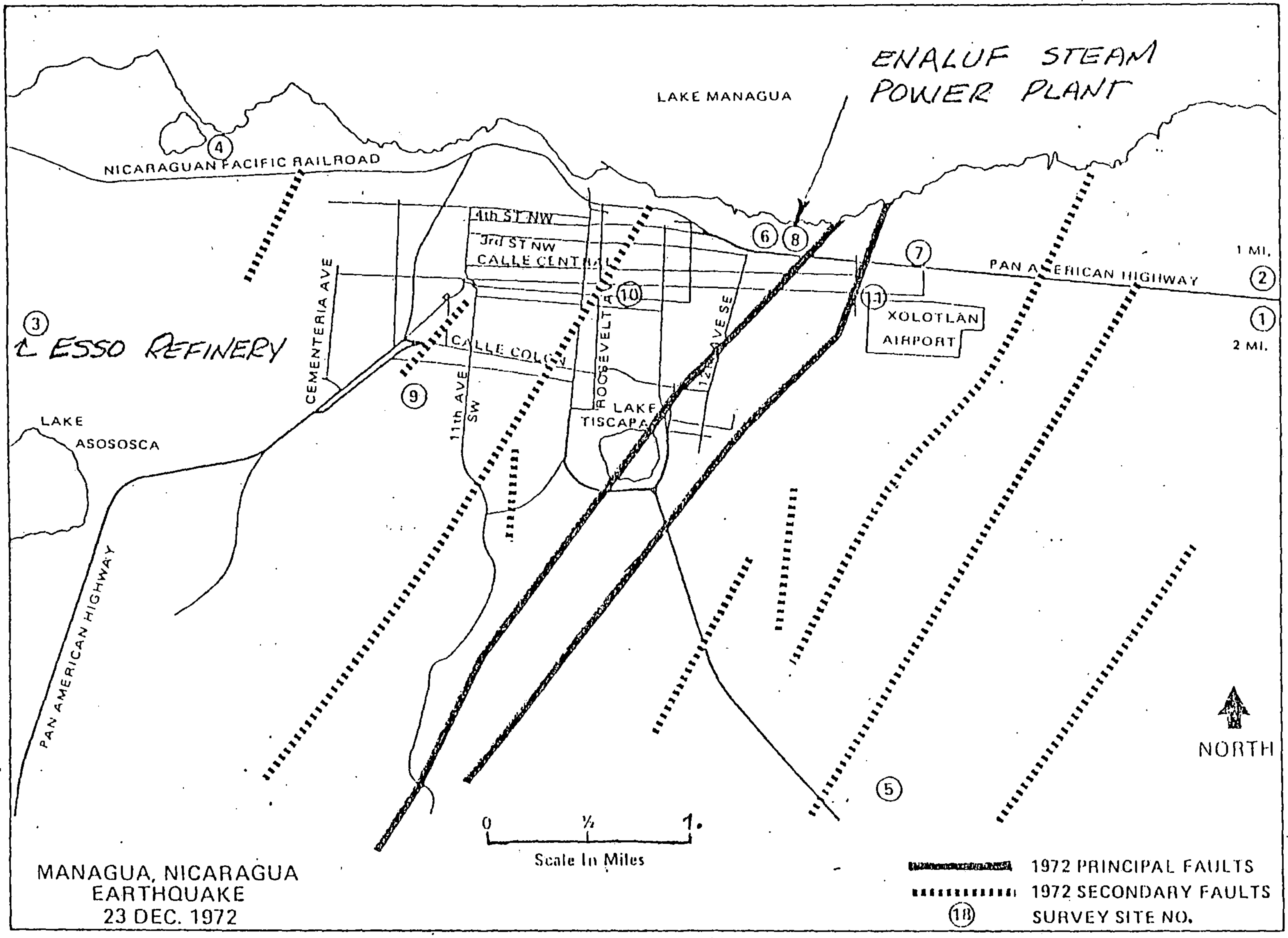

Fig. F-2 Map of Managua, Nicaragua showing the locations of facilities and buildings referred to in this report. See Table 1 for a list of the facilities.

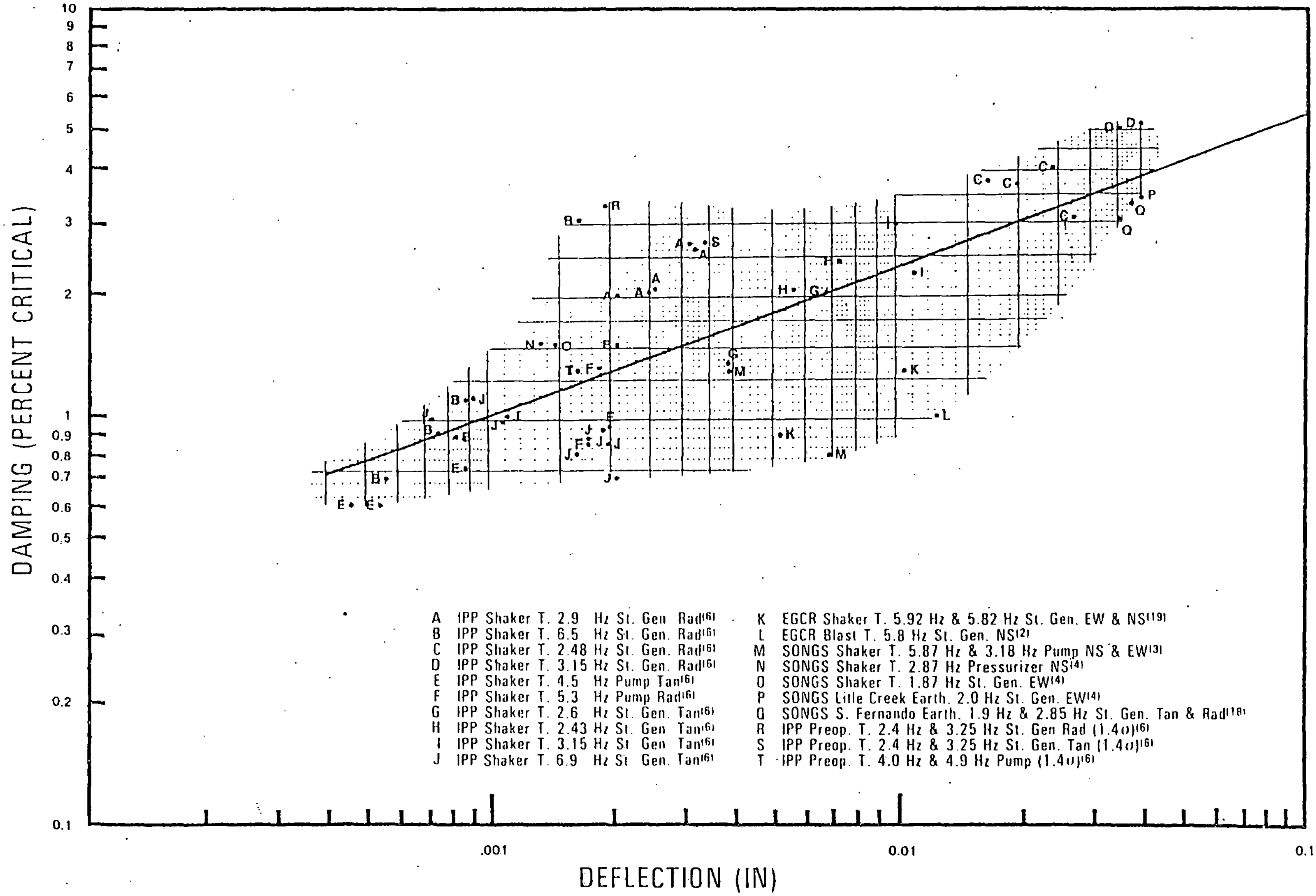

Fig. F-3 Percent of Critical Damping for Reactor Coolant Loop Components
Deflection (in)

TABLE F-1

CHRONOLOGY FOR SEISMIC ANALYSIS OF NUCLEAR PLANTS

1955	Static Methods
1960	Introduction of Ground Spectra; Buildings Considered Rigid
1965	Building Motion and Amplification of Spectra Considered
	Dynamic Analysis and Amplified Response Spectra First Applied to Piping
	Ground Spectra Change
1970	Soil Structure Interaction Considered; Ground Spectra Change
	3 Directional Earthquakes Regulatory Guides 1.92, 1.61, 1.60 Damping Changed
1975	Higher Site g Levels Considered; Systematic Reevaluation Program; Seismic Safety Research

TABLE F-2

CHRONOLOGY FOR SEISMIC ANALYSIS OF PIPING SYSTEMS

1955	Static Methods
1960	Static Application of Spectral Accelerations
1965	Response Spectra Dynamic Analysis; Consideration of Broadened Amplified Spectra; B31.7 Code - Evaluation Criteria
1970	ASME Code Section III Applied 3 Directional Earthquakes; Damping Changed; Regulatory Guides 1.92, 1.61, 1.60
1976	Occasional Time History Analysis; Occasional Plastic Analysis

TABLE F-3

DAMAGED EQUIPMENT AT THE ENALUF POWER PLANT

Unit 1

1. Forced-draft fan was out of alignment.
2. Induced-draft fan was out of alignment.
3. Bearings of the condensate pump burned out.
4. 440 V ac panel fell.
5. Condensate pump intake valve was broken.
6. Some tubing and refractory walls of the boiler were broken.
7. Deaerator number 1 fell from its base.
8. Stack suffered broken splice bolts at mid-elevation.

Unit 2

1. Forced-draft fan was out of alignment.
2. Induced-draft fan was out of alignment.
3. Refractory walls of the boiler were damaged.
4. Deaerator number 2 fell from its base.
5. The condensate pump intake valve was broken.

Unit 3

1. One 440 V ac control center fell.
2. Main transformer bushings were broken.
3. Starting transformer bushings were broken.
4. Some preheater seals were damaged.
5. Four turbine bearings burned out when the dc-powered emergency lube oil pump batteries broke.
6. A 69 kV switch bushing was broken.
7. Boiler support tubes over the preheater were broken.

TABLE F-3 (Cont)

8.	Forced-draft-fan control linkage was damaged.	2.12
9.	Miscellaneous air tubes and other tubing were broken.	2.14
10.	Evaporator drip valve was broken.	2.16
11.	Three recirculating valve bodies were broken.	2.18
12.	Batteries in the battery room fell from their supports and broke.	2.20
Miscellaneous Damage		2.23
1.	Turbine bay crane rails were bent and electrical supply conductors were broken. Crane remained in place.	2.25 2.26
2.	One 138 kV substation fell.	2.29
3.	Several transformer bushings were broken.	2.31
4.	Five lightning rods (69 to 138 kV) were broken or damaged.	2.33
5.	One capacitor transformer was broken.	2.35
6.	Miscellaneous insulators were broken.	2.37
7.	Water softener units fell from their supports and were damaged.	2.39
8.	One end of the bridge crane in the building that housed the diesel-electric generators fell from the crane girder.	2.41 2.42
9.	Other miscellaneous minor damage.	2.44

APPENDIX G 1.9

CORRESPONDENCE WITH NRC 1.11

APPENDIX G

CORRESPONDENCE WITH NRC

The following is a listing of correspondence with the NRC related to the reanalysis effort.

Date	Signature	Addressee	Letter No./Subject
		NRC to VEPCO	
3/13/79	Denton	Proffitt	Show Cause Order
4/2/79	Stello	Proffitt	Addendum to Show Cause Order
4/13/79	Stello	Proffitt	Use of Soil Structure Interaction Techniques
5/18/79	Stello	Proffitt	Request for Further SSI Information
5/25/79	Eisenhut	Proffitt	Factor Adjustment to SSI Calculated Stresses
		VEPCO to NRC	
3/30/79	Spencer	Denton/ Stello	198/Initial Response to Show Cause Order
4/23/79	Spencer	O'Reilly	289/Response to I.E. Bulletin No. 79-07
4/24/79	Spencer	O'Reilly	288/Response to I.E. Bulletin No. 79-07
4/27/79	Spencer	Denton/ Stello	311/Transmittal of Two Sample Problems to EG&G
5/2/79	Spencer	Stello	260/Submittal of SSI Information
5/22/79	Ragone	Hendrie	Comments on Moratorium/Surry Reanalysis
5/24/79	Spencer	Stello	Response to NRC Letter of 4/2/79
6/5/79	Spencer	Denton	Submittal of Report on

Date	Signature	Addressee	Letter No./Subject	
			Reanalysis	2.12
		S&W to NRC		2.16
3/22/79	Kennedy	Denton	Transmittal of S&W Computer Programs	2.19 2.20
3/30/79	Jacobs	Herring	Submittal of Computer Outputs	2.24
4/3/79	Jacobs	Bezler	Submittal of Benchmark Problem to Brookhaven National Laboratory	2.28 2.29 2.30
		S&W to NRC (Cont)		2.34
4/6/79	Kennedy	Denton	Transmittal of S&W Computer Programs	2.37 2.38
4/6/79	Jacobs	Stello	Plan for Verification of Dynamic Analysis Codes	2.40 2.41
4/11/79	Jacobs	Bezler	Submittal of Computer Outputs	2.43
4/13/79	Jacobs	Stello	Update and Status of Verification Plan for Dynamic Analysis Codes	2.45 2.46 2.47
4/18/79	Jacobs	Hartman	Submittal of Computer Outputs	2.49
4/27/79	Jacobs	Bezler	Submittal of Benchmark Problems	2.51 2.52
4/27/79	Jacobs	Stello	Status of Verification Plan for Dynamic Analysis Codes	2.54 2.55
5/14/79	Kennedy	Denton	Submittal of SHOCK1 Program Listing	2.57 2.58