BEAVER VALLEY POWER STATION, UNIT 1

REPORT ON THE REANALYSIS OF SAFETY-RELATED PIPING SYSTEMS

FOR

BEAVER VALLEY UNIT 1
DUQUESNE LIGHT COMPANY

ORIGINAL - JUNE 15, 1979

REVISION 1 - JULY 11, 1979

541 042

Stone & Webster Engineering Corporation
Boston, Massachusetts

If a seismic event which results in accelerations greater than acceleration level of 0.01g occurs during the period of interim operation, the plant will be shut down for inspection of those piping systems and supports which have not been shown to be fully acceptable for the OBE case. As discussed in the FSAR, Section 5.2.8.1, the accelerometers are initiated and recording started at a setpoint of 0.01 g acceleration. All seismic monitoring instrumentation is demonstrated operable in accordance with the test methods and testing frequencies specified in Table 4.3-4 of the Technical Specifications. The seismic instrumentation will be checked prior to startup.

This report addresses details of the analysis work, results of pipe and support analyses to date, presents a discourse on conservatisms, and discusses other topics within the scope of the reanalysis task. The report represents all work to date and is in addition to other submittals previously forwarded since the Order to Show Cause.

The seismic reanalysis is based on piping analysis programs, SHOCK3 and NUPIPE, that use 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 Safety Analysis Report. The reanalysis effort has [illegible]strated the conservative nature of the original seismic analysis. The piping systems have been found to be impacted only slightly after thorough, rigorous reanalysis.

541 043

Results to date also show that no piping of any size will have to be replaced or repaired.

Abbreviations used in this report are defined in Table 1-1.

541 044

TABLE 1-1

ABBREVIATIONS

S_{LP} = Pressure Stress

S_{DL} = Deadload Stress

S_h = Allowable Stress at Maximum (Hot) Temperature

S_{OBET} = Total Stress under OBE Condition

S_{DBET} = Total Stress under DBE Condition

S_{DBEI} = Inertial Effect of DBE

S_a = Allowable Stress

S_y = Yield Strength

S_u = Ultimate Strength

S_{th} = Thermal Stress

541 045

SECTION 3

RESPONSES TO NRC LETTERS AND ADDITIONAL QUESTIONS

The following four questions were raised by NRC personnel during a visit to the Beaver Valley Unit 1 project at S&W, June 5-7, 1979. Each NRC question is followed by the response.

NRC Questions

1. Indicate the frequency range over which the new SSI-ARS is not enveloped by the previous spectra. Discuss the effect this has on components, equipment, and piping analyzed to the old spectra.

   Response

The problems listed below with the system piping frequency and period use the old ARS curve as the run of record. A review of the curves included in this section which indicated a comparison between the peak spread SSI curve vs the old ARS curve shows that none of these problems except as noted fall into the period range where the SSI curve is not enveloped by the old ARS curve.

541 046

Problem No.	Frequency (cycles/seconds)	Period (seconds)
100	9.16	.11
179	10.87	.09
215	5.42	.18
101	4.98	.20
3063	9.51	.11
204	13.17	.08
785	3.78	.26
157	13.42	.07
158	23.95	04
212	10.47	.10
228	8.71	.11
229	9.41	.11
2112	3.31	.30
610	16.22	.06
612	16.66	.06
3011	3.87	.25
1	5.73	.18

Problems 785, 3011, and 1 presently fall into the area where the SSI curve is not enveloped by the old ARS.

541 047

A review of Problem No. 785 (Feedwater System) indicates that 72.6 percent of the allowable seismic OBE stress was attained using the original amplified response spectrum. Therefore, a substantial increase, 1.38 times for the OBE, would still be acceptable. The portion of the SSE curve for the horizontal earthquakes that exceeds the acceleration values of the original ARS is not seen by the piping system. For the vertical earthquake, the increase in acceleration is 20 percent which would still result in acceptable stress levels. For the DBE case, the horizontal accelerations increase 1.4 times and the vertical accelerations increase 1.2 times. These values are seen by the piping system and would result in stress levels below the allowable stress.

Problem No. 785: $S_{LP} + S_{DL} = 4832$; $S_h = 15000$

$$S_{LP} + S_{DL} + S_{OBET} = 14397 \;;\; 1.2\,S_h = 18000$$

$$S_{LP} + S_{DL} + S_{DBET} = 20232 \;;\; 1.8\,S_h = 27000$$

A review of Problem No. 3011 (Residual Heat Removal System) indicates that 98 percent of the allowable seismic OBE stress was attained using the original amplified response spectrum. For the DBE case, only 62 percent of the allowable stress was exhausted. Therefore, an increase of 1.02 times for the OBE and 1.61 times for the DBE case would be acceptable. A comparison of the original ARS with the SSI-ARS indicates that the acceleration values of the SSI-ARS not bounded by the original ARS were in a frequency range not experienced by the piping system. The

541 048

only exception to this is for the DBE case where the acceleration values of the Y-direction earthquake increased. The acceleration values for the first and second modes increased 1.17 and 1.5 times, respectively. This is within the 1.61 allowable increase given above. The contribution of the Y-direction earthquake is minor due to the rigidity of the system to the vertical response.

Problem No. 3011: $S_{LP} + S_{DL} = 4144$; $S_h = 14950$

$S_{LP} + S_{DL} + S_{OBET} = 17656$; $1.2\ S_h = 17940$

$S_{LP} + S_{DL} + S_{DBET} = 18148$; $1.8\ S_h = 26910$

A review of Problem No. 1 (River Water System) indicates that only 54.3 percent of the allowable seismic OBE stress was attained using the original amplified response spectrum. Similarly for the DBE case, 46.5 percent of the allowable was used. Therefore, a substantial increase, 1.84 times for OBE condition and 2.15 for the DBE condition, would be acceptable. For the OBE case, a comparison of the original ARS and the SSI-ARS indicates a slight increase in acceleration values for the SSI curve. This increase is only for the Y-direction earthquake, which does not contribute heavily to the overall response of the system.

For the DBE case, the comparison of the curves indicated an increase in the X-, Y-, and Z-direction earthquake acceleration values. However, the

541 049

stress margin [7.15] readily accommodates this increase. Problem No. 1 does not have any supports.

Problem No. 1:

$$S_{LP} + S_{DL} = 2244 \quad ; \quad S_h = 15000$$
$$S_{LP} + S_{DL} + S_{OBET} = 10801 \quad ; \quad 1.2\ S_h = 18000$$
$$S_{LP} + S_{DL} + S_{DBET} = 13754 \quad ; \quad 1.8\ S_h = 27000$$

The following ARS for the intake structure have not been peak spread; however, the problems (157, 158) using these curves have been reviewed and the system frequency is well beyond the spread peak.

A review of procedures used for the qualification of Seismic Category I equipment and the potential effect of SSI-ARS indicates that the original plant qualifications basis is conservative and that increased margins of safety would generally result from the use of SSI-ARS. This conclusion is confirmed by comparison of the original plant ARS with SSI-ARS and by review of procedures and seismic data used for the original equipment qualification basis.

Procedures used for the qualification of Seismic Category I equipment are described in BVPS FSAR Section B.2.2. These procedures resulted in qualification programs being implemented for balance-of-plant equipment. Mechanical equipment was principally qualified by static analysis

techniques and instrumentation and electrical equipment by shake table tests.

The original plant ARS was conservatively used for both analytical and test qualification programs. A review of the original plant ARS on a building-by-building and elevation-by-elevation basis indicated that peak resonant responses occurred below 10 Hz and that amplification of ground motion principally occurred below 20 Hz for structures housing Seismic Category I equipment. For each building a "cutoff frequency" was selected (i.e.. 10 or 20 Hz) in order to identify seismic acceleration levels above and below the cutoff frequency for calculational purposes. The "g" level identified below the cutoff frequency was a minimum of 1.3 times (Ref. FSAR Question 3.15) the peak ARS response. At the cutoff frequency the rigid range g value was conservatively selected. Equipment having a natural frequency below the cutoff frequency was qualified to an equivalent static acceleration of 1.3 times the peak ARS response. When equipment frequency characteristics were rigid (above the cutoff frequency) the maximum rigid range g values were used. For tested equipment, the maximum rigid range g levels were conservatively used for qualification.

A comparison of the ARS used for the original plant design with the SSI-ARS indicates that the original plant ARS are conservative based upon the

541 051

above seismic specification of static g values for qualification by static analysis and testing. Seismic Category I equipment was qualified on this conservative basis.

Seismic qualification of Seismic Category I equipment may also be established by the response spectrum modal analysis or seismic testing (Test Response Spectra) techniques. For these options, the ARS used for the original plant design provide the appropriate seismic definition for qualification. In this regard it is noted that peaks of the SSI-ARS are significantly lower than the peaks of the original plant ARS. The SSI-ARS peaks occur in the 2 to 5 Hz region for all structures evaluated and there is little amplification of maximum floor acceleration above 10 Hz. In some isolated cases the SSI-ARS curves exceed the original plant ARS in the low frequency region (below 5 Hz) distant from peak original ARS responses. This breaching of the original ARS would only potentially affect equipment whose natural frequency is below 5 Hz. One item, the outside recirculating spray pumps, was found which exhibited natural frequencies below 5 Hz. This component was qualified by dynamic analysis using the original plant ARS. It was concluded to be seismically qualified on the basis of a significant reduction of the primary modes response. Seismic Category I equipment which exhibits natural frequencies in excess of 5 Hz cannot be affected.

541 052

Based on this discussion it is concluded that the ARS used for original plant design provide an acceptable basis for qualification of Seismic Category I equipment.

Components loaded by piping systems are reviewed by analytical techniques described above. Each components nozzle is first reviewed to assure local component integrity. Loads for all nozzles were combined with the component seismic response to assure adequacy of component supports (near term). All components required for near term have been qualified to their revised loadings. Each component's seismic response was not revised to reflect changes due to SSI consideration. This is extremely conservative and facilitated an expeditious review of nozzle load data.

2. Indicate which code or what criteria is used for the evaluation of local stresses and whether anything different from the original analysis is being done in this respect.

<u>Response</u>

Local stresses are those induced at welded attachments to pipe, such as lugs or trunnions. Criteria for local stress evaluation are established through application of Welding Research Council Bulletin 107 (WRC-107).

541 053

This method of analysis is consistent with the original analysis performed.

3. Indicate whether eccentricities, e.g., valve center of gravities, are accounted for in the piping analyses.

Response

The eccentricity of the operators on all motor-operated and air-operated valves is included in the pipe stress analysis/review.

4. If interim operation is proposed, indicate how I&E Bulletin 79-02 will be addressed prior to startup for any support which contains base plates and concrete expansion anchor bolts, which are not found to be completely acceptable.

Response

Duquesne Light Company has a program underway for inspecting base plate and anchor bolts in the plant. Those supports which at this time are not completely acceptable have been included as priority items for this inspection.

541 054

NRC Letter

The following are responses to questions raised in an NRC letter (Appendix G) from Mr. D.G. Eisenhut to Mr. C.N. Dunn of Duquesne Light Company.

1. Indicate whether both OBE and DBE seismic stresses always include stresses due to seismic anchor movements (if any) and show how they are combined; e.g., sum of the absolute values. Is anything being done differently now than was done in the original SHOCK2 analysis? Your answer should include an explanation of the second paragraph of page B. 2-2 of the FSAR.

   Response

   For the reanalysis effort, the effects of the seismic anchor displacements have been evaluated statically and separately from the inertia effect. Static analysis is performed for each direction of relative displacement and for each earthquake, leading to a total of six evaluations. Internal moments resulting from the three evaluations for each earthquake are combined by SRSS on a component level and are then combined with the inertia effects by absolute summation, also on a component level. This procedure differs from the SHOCK2 procedure in that the SHOCK2 program utilized a single static analysis for each earthquake that incorporated the anchor movements in each of three directions simultaneously with the

541 055

equivalent inertia forces resulting from the intramodal and then the intermodal summation procedures of SHOCK2.

Calculated stresses in Table 4-1 include the effect of anchor displacement combined with inertia effects with the resulting response then combined and deadload and pressure stresses to form the total stress which is compared to the allowable stress, as follows:

$$S_{LP} + S_{DL} + S_{OBET} \leq 1.2\ S_h$$

$$S_{LP} + S_{DL} + S_{DBET} \leq 1.8\ S_h$$

Problem No. 120 (River Water System) has been evaluated for the DBE case as follows:

$$S_{LP} + S_{DL} + S_{DBET} \leq 1.8\ S_h$$

At the time the Beaver Valley 1 procedures were formulated, the B31.1 code did not address seismic design in the sense of providing detailed rules for stress determination and load combinations. Further, the code did not deal with Normal, Upset, Emergency, and Faulted stress limits. Since that time, development of B31.7 and ASME III have addressed these rules and limits.

541 056

Current rules allow two significant departures from the original techniques utilized on Beaver Valley Unit 1.

A. An option is provided for Upset Conditions whereby the anchor displacement effect can be considered in equation 9 along with deadweight, pressure, and seismic inertia effects or they may be combined with thermal expansion effects and evaluated under equation 10.

B. For Emergency and Faulted Conditions, the codes require evaluation of only the primary portion (inertia effect) of the seismic loadings and do not require that the anchor displacement effect be considered, since it is secondary in nature. Also allowed is a Faulted Stress allowable of 2.4 S_h, which was not stated in the Beaver Valley Unit 1 licensing documents; the equivalent value utilized was 1.8 S_h.

2. State how support stiffness is being accounted for in the current reanalysis effort and whether anything different from the original analysis is being done in this respect.

541 057

<u>Response</u>

Reanalysis efforts are utilizing two programs, SHOCK3 and NUPIPE. If SHOCK3 is utilized, supports and restraints are modeled in the manner of SHOCK2 as rigid members, essentially allowing zero deflection in each restrained direction. When NUPIPE is utilized, representative spring stiffnesses are input in each restrained direction.

Consistent support stiffnesses are used for each problem.

3. Provide the acceptance criteria used in the design of the pipe supports, including weld and bolt sizing criteria, and indicate any deviations from criteria originally used (except criteria established in addressing I&E Bulletin 79-02). Also, state your intention to comply, prior to facility startup, with I&E Bulletin 79-02 for all cases where loading on a pipe support increases as a result of the piping reanalysis and the support reevaluation indicates that any part of the support is not within the applicable acceptance criteria.

541 058

Response

Acceptance criteria used in the design of pipe supports are shown in Table 3-1. Allowable loads for drilled-in-concrete anchor bolts are shown in Table 3-2. These criteria are being utilized for the reevaluation effort except under the conditions of Section 2 which addresses interim startup conditions.

Duquesne Light Company has a program underway that addresses the following items as a plan of action to comply with IE Bulletin 79-02 for those pipe supports requiring modifications based upon pipe stress analysis described in this report.

a. Where pipe support reanalysis results in new supports, the base plates and anchor bolts shall be designed incorporating IE Bulletin 79-02 criteria.

b. Where pipe support reanalysis results in modifications to existing supports, the base plates and anchor bolts shall be evaluated incorporating IE Bulletin 79-02 criteria.

c. Field inspections shall be performed on those existing base plates being modified in order to ensure bolt integrity.

541 059

4. Discuss the impact the current piping stress reanalysis effort has on the FSAR pipe break criteria. Indicate whether postulated pipe break locations could or have change(d) as a result of the reanalyses and, if so, what you propose to do in the event a break location previously not designed for must be postulated.

Response

The reanalysis performed to date to the licensed acceptance criteria indicates that stress patterns have not changed significantly since maximum stresses occur at points of stress intensification, such as elbows and branch connections.

A detailed review of these problems indicates that the first five highest stress points occur at points of stress intensification. They also occur in those areas where the lines are fully restrained by pipe whip restraints and therefore no additional restraints are required.

FSAR Section 5.2.6.3 states that break locations have been postulated for only the main steam and feedwater inside containment and Appendix D of the FSAR states that breaks need only be postulated in the main steam and feedwater systems outside containment.

541 060

NRC Letter

The following are responses to questions raised in a second NRC letter, dated May 25, 1979 (Appendix G) also from Mr. D.G. Eisenhut to Mr. C.N. Dunn of Duquesne Light Company.

1. All pipe runs analyzed with SHOCK2 must be identified.

   Response

   Appendixes A and B identify problems originally analyzed with SHOCK2. Appendix A lists those problems addressed for interim startup and Appendix B lists those problems to be analyzed in the long term.

2. Request the following full size drawings:

   RM-21B

   RM-27A, B

   RM-29A, B, C, D

   RM-37A

   RM-39A, B

541 061

Response

Full size drawings were provided to the NRC by S&W during the meeting at S&W on June 5, 1979.

3. Reanalysis of the primary component cooling water heat exchanger discharge piping.

Lines: 18"-WR-14-151 Q3
18"-WR-15-151 Q3
18"-WR-16-151 Q3
30"-WR-17-151 Q3

Failure of any of these lines would result in flooding of redundant safety related equipment.

Response

These lines have been added to the problems for interim startup. Problem No. 121 includes:

18"-WR-14-151-Q3
18"-WR-15-151-Q3

541 062

30"-WR-16-151-Q3

30"-WR-17-151-Q3

Problem No. 122 includes:

30"-WR-17-151-Q3

4. Reanalysis of the following lines located in the intake structure.

30"-WR-171-151-Q3

30"-WR-172-151-Q3

30"-WR-175-151-Q3

18"-WR-154-151-Q3

12"-WR-177-151-Q3

10"-SWW-14-151-Q3

10"-SWW-1-121*

Failure of any of these lines could result in possible flooding of safety related pumps. The asterisked line, unlike the other lines, was not considered safety-related during the plant design and was never seismically analyzed. This line runs above and adjacent to River Water Pump 1B and can only be isolated from the seismically designed piping by a

541 063

manually operated butterfly valve which is normally open during plant operation.

Response

These lines have been added to the problems for interim startup. Problem No. 152 includes:

30"-WR-171-151-Q3

30"-WR-172-151-Q3

30"-WR-175-151-Q3

Problem No. 160, which overlaps problem No. 159, includes:

18"-WR-154-151-Q3

Problem No. 161 includes:

12"-WR-177-151-Q3

541 064

Problem No. 165 includes:

10"-SWW-14-151-Q3

10"-SWW-1-121

5. The cooling water discharge lines from the emergency diesel generator cooling system heat exchangers downstream of the normal open isolation valves are not seismically qualified. These lines are located in the diesel generator compartments and their failure could impact on the operation of the emergency diesels. A seismic analysis should be performed on these lines.

Response

The cooling water discharge lines, which are less than 6 inches, were not analyzed on SHOCK2 but were hand calculated and seismically supported based on standard spacing between supports.

6. The discharge lines of the quench spray pumps have not been proposed for reanalysis.

10"-QS-3-153-Q3

10"-QS-4-153-Q3

541 065

8"-QS-22-153-Q3

8"-QS-23-153-Q3

Justify that reanalysis of the above lines is not necessary.

Response

The discharge lines of the quench spray pumps were seismically analyzed on NUPIPE for the DBE plus water hammer loads previous to the present reanalysis effort; consequently, these lines were not included in this reanalysis effort. The OBE case for which the SHOCK2 run is the calculation of record will be rerun in the long term reanalysis.

7. The recirculation spray piping both inside and outside containment with the exception of the lines listed below is not being reanalyzed. Justify that reanalysis of the recirculation spray system is not necessary.

12"-RS-5-153-Q3

12"-RS-7-153-Q3

12"-RS-8-153-Q3

541 066

Response

The recirculation spray lines were seismically analyzed on NUPIPE previous to the present reanalysis; consequently, these lines were not included in this reanalysis effort. The OBE case for which the SHOCK2 run is the calculation of record will be rerun on the long term reanalysis.

8. Verify that the discharge lines from the control room air condition condensers, the charging pump coolers, and line 6"-WR-53-151-Q3 have been seismically analyzed by an acceptable method. These lines are part of the river water system.

Response

The discharge lines were not analyzed on SHOCK2, but were hand calculated and seismically supported based on standard spacing between supports.

Additional NRC Questions

The following questions were raised during a telephone conversation among Duquesne Light Company, Stone & Webster, and NRC personnel on June 28, 1979.

541 067

1. It appears that Table B-2 contains some problems which should be addressed in the short term.

Response

The problems previously included in Table B-2 have been rereviewed in depth and, as a result, the short-term effort has been revised to include the following:

Problem 213
Problem 2113
Problem 616
Problem 651
Problem 652
Problem 653
Problem 301 (Comprised of Problems 308, 3007, 3008, 3013 and 3014)

The following problems have been found to be checks of the hand calculations of record and have been transferred to Table B-3:

541 068

310	3021	3131
312	3031	
341B	3035	
655C	3043	
840	3100	
965	3127	

Problem 139 was voided because the line was not required to be seismically supported.

2. For those problems not included in the interim scope, what is the consequence of a failure?

Response

The systems which are not included in the interim scope are (1) component cooling water system outside containment, (2) fuel pool purification and cooling system, and (3) quench and recirculation spray system.

The component cooling water (CC) system outside containment has been evaluated using the short-term criteria with the following results:

541 069

Problem 171, the supply to the CC heat exchangers has one support (H-65) out of 15 which has a local overstress of 4 percent. Problem 270, the discharge from the CC heat exchangers has one support (H-56) out of 13 which has a local overstress in a lug of 78 percent. It is considered that, if a DBE were to occur, failure of these two supports would not cause a system rupture or a resultant loss of function.

The fuel pool cooling and purification system is presently isolated since there is no spent fuel being stored.

The quench and recirculation spray systems have been completely analyzed for DBE and water hammer loads using NUPIPE. The OBE case will be run in the long term.

3. How have stress intensification factors been applied at branch connections during the reanalysis?

Response

Appropriate stress intensifications from B31.1 have been applied to the run pipe at reduced outlet branch connections. Branches which are uncoupled have been evaluated for the effects of the movements of the run

541 070

pipe using appropriate stress intensification. The thermal and seismic displacements of the run pipe are applied at the branch with the stresses being determined by the use of a flexibility nomograph. The stresses are then compared to code allowables.

4. In the SSI Report, where do the building displacements come from? Which data sets were used and what are the bases for their selection?

Response

The building acceleration and displacement profiles, illustrated in Figures 4-11 and 4-12 of the Report on "Soil-Structure Interaction in the Development of Amplified Response Spectra for Beaver Valley Power Station Unit 1," are maxima from the time history responses at each mass point in the structural dynamic model and are determined automatically by the FRIDAY computer program. They are based on the FSAR earthquake, the strain-compatible free-field soil properties from the final iteration of the SHAKE computer program, and a structural damping ratio of 0.02. This is consistent with the basis used for generation of Amplified Response Spectra (ARS) and conservative with respect to soil properties associated with broadened and 'bumped' ARS, referred to under Item 7, Section 9.5 of the report. Displacements calculated on this basis are, therefore, reasonable for use in the reevaluation of piping systems.

5. Provide a general statement relative to the selection of an amplified response spectrum at the highest support location versus the center of gravity of the piping system.

Response

Appendix B2.1 (page B2.2) of the FSAR states that Beaver Valley Unit 1 dynamic piping stress analysis is based on a response spectra curve closest to, but higher than, the center of gravity of the piping system. However, the procedure that is being implemented on the reanalysis effort, that is, to use the amplified response spectra at the highest pipe support elevation is always conservative, because the ARS at the highest support location will always result in higher acceleration levels than at the center of gravity.

For the reanalysis to date, only two problems have used ARS curves which have been applied just above the center of gravity of the piping system. These two problems are the pressurizer relief valve discharge lines (833) and the pressurizer spray line (1200); both systems encompass a large elevational change from termination to termination. In these two cases, it has been deemed to be more reasonable to use an ARS curve close to the center of gravity of the system, rather than at the highest support location.

541 072

TABLE 3-1

PIPE SUPPORT ACCEPTANCE CRITERIA

Load Combination	Tension	Shear	Column Buckling	Welds
Maximum of:				
DL + TH + OBET or DL + TH + DBET	0.8 Sy	0.513 Sy (web) 0.53 Sy	Note (1)	0.3 Su

Note (1): Column buckling criteria are established by Euler equations and are a function of $(\frac{kl}{r})$ in accordance with Table 1-36, p 5-84 of AISC.

541 073

TABLE 3-2

DRILLED-IN-CONCRETE ANCHOR BOLT ALLOWABLE LOADS

1. Red head self-drill type S, and type JS installed in 3,000 psi concrete; see Attachment A, Tables I and II, respectively

   For reductions in allowable loads due to closer spacing, see Attachment A, Tables III and IV, respectively.

2. Star slugin compounded cinch anchor bolts and ring wedge cinch anchors; see Attachment A, Tables V and VI, respectively.

3. Hilti or Phillips wedge type anchor bolts are as follows:

Bolt Diameter	Allowable Tension (lbs)	Loads Shear (lbs)
3/8"	950	1150
1/2"	2185	2180
5/8"	2145	2845
3/4"	3525	3800
7/8"	4100	4585
1"	5710	6780

   The one-third increase does not apply to drilled-in-concrete anchor bolts.

4. Anchor bolt tension and shear interaction equation:

$$\left(\frac{T}{T_A}\right)^{5/3} + \left(\frac{S}{S_A}\right)^{5/3} \leq 1.0$$

   Where T/T_A and S/S_A are the ratios of the actual over the allowable for tension and shear, respectively.

541 074

TABLE 3-2 (Cont)

ATTACHMENT A
TABLES I & II

TABLE I

Cat. No. Self-Drill	Cat. No. Non-Drill	Bolt Dia.	Anchor Data O.D.	Anchor Data Depth	Installed in Concrete: Allowable Pullout Values I.C.B.O. (1) 2000 (1)	2500	3000	3500	Installed in Concrete: Allowable Shear Values I.C.B.O. 2000	2500	3000	3500	Installed in Grouted Concrete Blocks (3) I.C.B.O. Pullout C.B.	Shear Reg. C.B.	Shear Lt.Wt. C.B.
S-14		1/4"	7/16"	1-3/32"	410	455	500	540	490	490	490	490	375	550	340
	J-14			1-1/8"											
S-16		5/16"	15/32"	1-5/16"	575	660	750	840	770	770	770	770	550	800	500
	—			—											
S-38		3/8"	9/16"	1-17/32"	800	910	1020	1140	1000	1100	1100	1100	750	900	740
	J-38			1-9/16"											
S-12		1/2"	11/16"	2-1/32"	1210	1380	1550	1720	1180	1350	1550	1750	1150	1100	880
	J-12			2-1/16"											
S-58		5/8"	27/32"	2-15/32"	1550	1780	2000	2230	1400	1600	1820	2020	1450	1300	1060
	J-58			2-9/16"											
S-34		3/4"	1"	3-1/4"	1880	2120	2350	2570	1670	1900	2120	2370	1700	1600	1300
	J-34			3-3/16"											
S-78		7/8"	1-1/8"	3-11/16"	2050	2330	2630	2900	2050	2230	2500	2800	1800	2000	1600
	—			—											

TABLE II

Cat. No.	Bolt Dia.	Drilling Data Drill Dia.	Drilling Data Hole Depth	Installed in Concrete: Allowable Pullout Values I.C.B.O. 2000	2500	3000	3500	Installed in Concrete: Allowable Shear Values I.C.B.O. 2000	2500	3000	3500
JS-14	1/4"	1/4"	1-5/8"	390	450	530	540	490	490	490	490
JS-38	3/8"	3/8"	1-7/8"	570	665	775	885	570	620	720	825
JS-12	1/2"	1/2"	2-1/4"	840	975	1130	1300	750	830	910	1000
JS-58	5/8"	5/8"	2-3/4"	1150	1320	1540	1775	1000	1100	1250	1370
JS-34	3/4"	3/4"	3-1/4"	1800	2080	2100	2300	1500	1650	1800	1950

RED HEAD
PHILLIPS ANCHORS

APPROVED BY: UL FM

NOTES:
1. All column headings refer to 28 day strength of stone aggregate concrete.
2. Quoted from International Conference of Building Officials report #1372.6.
3. Allowable loads apply to anchors installed in fully grouted cells. Values are for concrete masonry units conforming to U.B.C. standard No. 24-4-64 with special inspection. Without special inspection use 50 per cent of listed values.

541 075

PULLOUT CAPACITIES OF PHILLIPS RED HEAD CONCRETE ANCHORS AS AFFECTED BY SPACING

In compliance with the request of the client, Doberne & Eigenson conducted a series of tests to develop the information used in this report. The test facilities of the Smith-Emery Company, an independent testing laboratory, were used.

The purpose of these tests was to determine the load holding characteristics of Phillips anchors under various spacing arrangements.

<u>Results</u>

1. When the spacing between adjacent anchors reaches a distance equal to several times the anchor diameter, there is no loss in capacity. The following table shows the minimum center-to-center spacing that could be used with each anchor without causing a loss in individual capacity.

TABLE III

Anchor Bolt Size	1/4"	5/16"	3/8"	1/2"	5/8"	3/4"	7/8"
Minimum Spacing for 100% capacity	3"	3-1/4"	4"	5"	6"	7"	8"

2. When the center-to-center spacing, as shown in the above table is reduced, the capacity of the individual anchor decreases.

   The following table shows center-to-center spacing corresponding to a 20% reduction in individual anchor capacity.

TABLE IV

Anchor Bolt Size	1/4"	5/16"	3/8"	1/2"	5/8"	3/4"	7/8"
Minimum Spacing for 80% Capacity	1-1/2"	1-5/8"	2"	2-1/2"	3"	3-1/2"	4"

Dimensions of blocks used for tests were 8" x 8" x 16" with an average compressive strength of 2650 psi.

541 076

TABLE V

STAR SLUGIN COMPOUNDED CINCH ANCHOR BOLTS
(THREADED OR PLAIN TYPE)

Bolt size		1/4"	3/8"	1/2"	5/8"	3/4"	7/8"	1"	1 1/8"	1 1/4"	1 1/2"
Drill size and hole diameter		1/2"	3/4"	1"	1 1/8"	1 3/8"	1 1/2"	1 5/8"	2"	2 1/8"	2 3/8"
Minimum depth for hole (2 unit set)		1 1/4"	1 1/2"	1 7/8"	2"	2 3/8"	2 3/4"	3 3/4"	4 1/2"	4 3/4"	5 1/4"
Shear strength of bolt in lbs.-Ult. 30,000 lbs		600	1550	2850	4550	6380	9150	12400	20800	26700	53300
Breaking strain of bolt in lbs: Ult 60,000 lbs		1750	4800	8200	13150	19650	27200	35800	45000	57800	84000
Safe load for each one unit set	*	125	300	400	725	800	850	900	On Application		
Safe load for each two unit set	*	250	600	800	1450	1600	1700	1800			
Safe load for each three unit set	*	375	900	1200	2175	2400	2550	2700			
Safe load for each four unit set	*	500	1200	1600	2900	3200	3400	3600			
Safe load for each five unit set	*	625	1500	2000	3625	4000	4250	4500			
Safe load for each six unit set	*	750	1800	2400	4350	4800	5100	5400			

* Based on a safety factor of 10 to 1. Safe loads for anchors are for tension or shear

TABLE VI

RING WEDGE CINCH ANCHORS

Bolt size	1/4"	3/8"	1/2"	5/8"	3/4"	7/8"	1"	1 1/8"	1 1/4"	1 1/2"
Area of bolt at shank	.047	.110	.196	.307	.448	.601	.785	.994	1.227	1.767
Area of bolt at thread	.027	.068	.126	.202	.302	.419	.551	.693	.890	1.294
Diameter of hole & drill equal	5/8"	3/4"	1"	1 1/8"	1 3/8"	1 1/2"	1 5/8"	2"	2 1/8"	2 3/8"
No. of units req'd to equal strength of bolt *	2	2	2	3	3	3	4	4	4	4
Minimum depth of hole for number of units specified	1 1/16"	1 1/2"	1 3/4"	2 7/8"	3 3/4"	4"	6 1/2"	7 1/2"	8"	9 1/4"
Strength of bolts, Breaking strain, Pounds	2202	4950	8803	13754	19779	26880	35168	42336	52192	75264
Safe load for U.S. Standard Steel Bolts - Pounds **	172	451	845	1370	2070	2900	3800	4790	6210	9060

* When masonry is of doubtful grade additional units should be used

** Based upon approximately 1/10th actual holding power

POOR ORIGINAL

541 077

SECTION 4

PIPE STRESS RESULTS

A total of 120 pipe stress problems have been identified for reanalysis and are being analyzed by Stone & Webster Engineering Corporation in Boston, Massachusetts.

The pipe stress reanalysis consists of substituting the SHOCK3 or NUPIPE code for the SHOCK2 code. SHOCK3 is a current seismic code that calculates both intramodal and intermodal seismic forces using a modified square root of the sum of the squares (SRSS) technique and an SRSS technique, respectively, rather than an algebraic summation. The NUPIPE Program utilizes modal response combinations as follows:

Intermodal - SRSS for combination, grouping for modal combination (where closely spaced modes are combined by absolute sum).

Intramodal - SRSS for direction combination.

541 078

Field verified piping fabricator isometric drawings provide the basis for program inputs for the pipe stress reanalysis.

Additionally, in some cases, piping is analyzed utilizing amplified response spectra (ARS) that are developed using soil structure interaction techniques (SSI-ARS). The resultant stresses and loads are used to evaluate piping, supports, nozzles, and penetrations. These techniques are discussed in Section 8.7.

Of the 120 SHOCK2 problems, 93 have been reanalyzed and are within allowable stress values. Table 4-1 lists the problems including the peak stress values for the SHOCK3 and NUPIPE pipe stress runs.

Stresses were computed by the SHOCK3 or NUPIPE program using different mass models and in some cases different ARS than the original calculations. More importantly, the reanalyses were based on field-verified, as-built conditions which in some cases differ significantly from the original design conditions. For these reasons, the originally calculated stresses are not comparable to the new stresses.

Table 4-2 summarizes the nozzles and penetrations evaluated under the reanalysis program. Of a total of 87 nozzles on problems within the scope of the interim effort, 82 have been evaluated and found to be acceptable, and 5

541 079

are contained in problems for which the final pipe stress analysis is not complete but are expected to be acceptable based on reanalysis.

The SHOCK2 stress problems contained in the interim effort include 50 penetrations, all of which have been evaluated and found to be acceptable.

Summary

During the period between the initial issue of this report and this revision, 30 additional problems have been rerun on NUPIPE using the SSI-ARS curve. All of the above 30 problems have been reanalyzed and were found to be within allowable stress limits.

541 080

TABLE 4-1

PIPE STRESS REEVALUATION SUMMARY

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
Reactor Coolant			
653A	12,547 18,820	7,671 8,189	NUPIPE/SSI-ARS
653B	19,200 28,800	10,084 10,575	NUPIPE/SSI-ARS
653C	19,200 28,800	15,060 17,244	NUPIPE/SSI-ARS
833 & 8	17,220/19,200[1] 25,830/28,200	12,420[2] 17,300	NUPIPE/SSI-ARS
1200	19,200 28,800	12,690 16,424	NUPIPE/SSI-ARS
1201	19,200 28,800	9,711 10,442	NUPIPE/SSI-ARS
Safety Injection			
391A	19,080 28,620	15,425 18,228	SHOCK3/SSI-ARS
2112	22,500 33,750	20,754 25,002	SHOCK3
610	18,586 27,878	2,081 2,328	NUPIPE/SSI-ARS
613	21,180 31,770	9,802 14,336	NUPIPE/SSI-ARS
611	19,500/20,280[1] 29,250/30,400	17,585 16,069	NUPIPE/SSI-ARS
15	17,340/19,200[1] 26,010/28,800	7,214 8,123	SHOCK3/SSI-ARS

541 081

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
1011	20,850 31,275	8,245 14,087	SHOCK3/SSI-ARS
301	19,200 28,800	7,722 8,712	NUPIPE/SSI-ARS
213[4]	20,388 30,582	5,837 5,437	NUPIPE/SSI-ARS
2113[4]	20,388 30,582	4,078 4,443	NUPIPE/SSI-ARS
Quench Spray			
211	22,500 33,750	1,807 2,653	SHOCK3/SSI-ARS
212	22,500 33,750	10,445 11,639	SHOCK3
228	22,500 33,750	12,149 16,589	SHOCK3
229	22,500 33,750	11,810 15,987	SHOCK3
Recirculation Spray			
612	18,796 28,193	1,366 1,434	NUPIPE/SSI-ARS
Charging and Volume Control			
100	18,660 27,990	15,220 15,468	SHOCK3
102	18,660 27,990	6,289 6,621	SHOCK3/SSI-ARS

541 092

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
Residual Heat Removal			
255A	17,940 26,910	5,075 6,041	NUPIPE/SSI-ARS
256	17,160 25,740	11,843 15,063	NUPIPE/SSI-ARS
14	17,940/19,200(1) 26,910/28,800	8,740 10,376	SHOCK3/SSI-ARS
3011	17,940 26,910	17,656 18,148	NUPIPE
616	18,300 27,450	8,498 11,500	NUPIPE/SSI-ARS
Component Cooling Water			
302	18,000 27,000	6,295 10,271	NUPIPE/SSI-ARS
303	18,000 27,000	6,906 10,377	NUPIPE/SSI-ARS
304	18,000 27,000	7,836 11,108	NUPIPE/SSI-ARS
305	18,000 27,000	5,835 8,330	NUPIPE/SSI-ARS
306	18,000 27,000	4,246 5,077	NUPIPE/SSI-ARS
307	18,000 27,000	6,133 7,780	SHOCK3/SSI-ARS
180E(3)	18,000 27,000	7,55[illegible] 7,370	NUPIPE/SSI-ARS

541 083

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
181E[3]	18,000 27,000	7,516 15,168	NUPIPE/SSI-ARS
170C[3]	18,000 27,000	4,339 5,031	NUPIPE/SSI-ARS
171[3]	18,000 27,000	5,472 5,667	NUPIPE/SSI-ARS
172 [3]	18,000 27,000	8,319 13,734	NUPIPE/SSI-ARS
173D[3]	18,000 27,000	5,107 5,994	NUPIPE/SSI-ARS
174D[3]	18,000 27,000	3,036 4,115	NUPIPE/SSI-ARS
175B[3]	18,000 27,000	5,197 5,414	NUPIPE/SSI-ARS
176A[3]	18,000 27,000	3,505 3,777	SHOCK3/SSI-ARS
177[3]	18,000 27,000	9,223 13,707	SHOCK3/SSI-ARS
178C[3]	18,000 27,000	15,703 15,797	NUPIPE/SSI-ARS
179[3]	18,000 27,000	2,081 2,995	NUPIPE/SSI-ARS
183[3]	18,000 27,000	9,320 11,336	NUPIPE/SSI-ARS
184[3]	18,000 27,000	10,133 11,734	NUPIPE/SSI-ARS
186A[3]	18,000 27,000	15,703 15,797	NUPIPE/SSI-ARS
270A[3]	18,000 27,000	15,703 15,797	NUPIPE/SSI-ARS

541 084

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
215	18,000 27,000	16,311 26,731	SHOCK3
217	18,000 27,000	15,751[2] 23,924	NUPIPE/SSI-ARS
930	18,000 27,000	15,751 23,924	NUPIPE/SSI-ARS
931	18,000 27,000	15,751 23,924	NUPIPE/SSI-ARS
214	18,000 27,000	14,740 25,774	NUPIPE/SSI-ARS
River Water			
1	18,000 27,000	10,801 13,759	SHOCK3
30	18,000 27,000	4,830 7,576	NUPIPE/SSI-ARS
31	18,000 27,000	4,830 7,576	NUPIPE/SSI-ARS
32	18,000 27,000	5,363 8,390	NUPIPE/SSI-ARS
33	18,000 27,000	5,169 8,241	NUPIPE/SSI-ARS
140	18,000 27,000	13,758 16,349	SHOCK3/SSI-ARS
384	18,000 27,000	6,156 8,512	SHOCK3[5]
157	18,000 27,000	1,884 2,011	NUPIPE/SSI-ARS
158	18,000 27,000	1,976 2,090	NUPIPE/SSI-ARS

541 085

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
159(b)	18,000 27,000	10,443 17,277	NUPIPE/SSI-ARS
128	18,000 27,000	10,760 12,562	NUPIPE/SSI-ARS
127	18,000 27,000	13,384 15,970	NUPIPE/SSI-ARS
125	18,000 27,000	10,760 12,562	NUPIPE/SSI-ARS
124	18,000 27,000	13,384 15,970	NUPIPE/SSI-ARS
123	18,000 27,000	10,861 17,797	NUPIPE/SSI-ARS
120	18,000 27,000	8,820 (?)	NUPIPE/SSI-ARS
126	18,000 27,000	13,384 15,970	NUPIPE/SSI-ARS
216	18,000 27,000	6,047 9,989	NUPIPE/SSI-ARS
203	18,000 27,000	2,444 4,189	NUPIPE/SSI-ARS
2031	18,000 27,000	8,260 9,699	NUPIPE/SSI-ARS
152	18,000 27,000	4,950 6,032	NUPIPE/SSI-ARS
121	18,000 27,000	6,068 8,354	NUPIPE/SSI-ARS
122	18,000 27,000	8,038 14,096	NUPIPE/SSI-ARS
165	18,000 27,000	4,950 6,032	NUPIPE/SSI-ARS

541 086

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
652	18,000 27,000	1,231 1,394	NUPIPE/SSI-ARS
653	18,000 27,000	1,495 1,624	NUPIPE/SSI-ARS
Main Steam			
658	22,500 33,750	10,248 12,025	SHOCK3/SSI-ARS
6590	18,000 27,000	9,977 11,108	SHOCK3/SSI-ARS
101	18,000 27,000	16,917 18,277	SHOCK3
659	22,500 33,750	10,544 12,570	SHOCK3/SSI-ARS
660	22,500 33,750	11,121 13,304	SHOCK3/SSI-ARS
3063	22,500 33,750	12,289 16,481	SHOCK3
Feed Water			
204	18,000 27,000	2,952 3,761	SHOCK3
783	18,000 27,000	9,361 11,624	SHOCK3/SSI-ARS
784	18,000 27,000	10,853 13,726	SHOCK3/SSI-ARS
785	18,000 27,000	14,397 20,232	SHOCK3
261	18,000 27,000	10,479 13,585	SHOCK3/SSI-ARS

541 087

TABLE 4-1 (Cont)

System and Problem No.	Allowable Stress (psi)	Reanalysis Maximum Stress	Reanalysis Method
Diesel Generator Exhaust			
651	12,960 19,440	1,201 1,717	NUPIPE/SSI-ARS

Notes: SSI-ARS = Amplified response spectra developed using soils structure interaction techniques

Stresses shown are Operational Basis Earthquake (OBE) Stresses / Design Basis Earthquake (DBE) Stresses

(1) TP304/TP316 allowables

(2) After modification

(3) Problems are no longer within scope of short-term reanalysis effort. See Appendix B.

(4) Problems 213 and 2113 include $S_{DL} + S_{LP} + S_{OBEI}$ and $S_{DL} + S_{LP} + S_{DBEI}$ only.

(5) Being rerun with SSI-ARS.

(6) Problem 159 includes Problems 160 and 161.

(7) Evaluated for the DBE case only.

541 088

TABLE -2

NOZZLE AND PENETRATION SUMMARY

System/ Problem No.	Total No. of Nozzles/ Penetrations	No. Acceptable After Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
Reactor Coolant			
653A	6/0	6/0	0/0
653B	8/0	8/0	0/0
653C	8/0	8/0	0/0
833 & 8	4/0	4/0	0/0
1200	1/0	1/0	0/0
1201	0/0	0/0	0/0
Safety Injection			
391A	1/0	1/0	0/0
2112	0/0	0/0	0/0
610	2/2	2/2	0/0
613	0/0	0/0	0/0
615	2/3	2/3	0/0
15	1/0	1/0	0/0
1011	0/0	0/0	0/0
301	0/2	0/2	0/0
213	0/0	0/0	0/0
2113	0/0	0/0	0/0

541 089

TABLE 4-2 (Cont)

System/ Problem No.	Total No. of Nozzles/ Penetrations	No. Acceptable After Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
Quench Spray			
211	1/0	1/0	0/0
212	1/0	1/0	0/0
228	1/0	1/0	0/0
229	1/0	1/0	0/0
Recirculation Spray			
612	2/2	2/2	0/0
Charging & Volume Control			
100	2/0	2/0	0/0
102	1/0	1/0	0/0
Residual Heat Removal			
255A	6/0	6/0	0/0
256	0/0	0/0	0/0
14	1/0	1/0	0/0
3011	0/0	0/0	0/0
616	0/1	0/1	0/0
Component Cooling Water			
302	1/1	1/1	0/0
303	1/1	1/1	0/0

541 090

TABLE 4-2 (Cont)

System/ Problem No.	Total No. of Nozzles/ Penetrations	No. Acceptable After Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
304	1/1	1/1	0/0
305	1/1	1/1	0/0
306	0/1	0/1	0/0
307	0/1	0/1	0/0
180E(1)	2/0	2/0	0/0
181E(1)	2/0	2/0	0/0
170C(1)	3/0	1/0	2/0
171(1)	6/0	6/0	0/0
172(1)	0/0	0/0	0/0
173D(1)	0/0	0/0	0/0
174D(1)	0/0	0/0	0/0
175B(1)	0/0	0/0	0/0
176A(1)	0/0	0/0	0/0
177(1)	1/0	1/0	0/0
178C(1)	1/0	1/0	0/0
179(1)	1/0	1/0	0/0
183(1)	3/0	3/0	0/0
184(1)	2/0	2/0	0/0
186A(1)	0/0	0/0	0/0
270A(1)	3/0	3/0	0/0
215	0/4	0/4	0/0
217	0/4	0/4	0/0

541 091

TABLE 4-2 (Cont)

System/ Problem No.	Total No. of Nozzles/ Penetrations	No. Acceptable After Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
930	0/1	0/1	0/0
931	0/1	0/1	0/0
214	0/1	0/1	0/0
River Water			
1	4/4	4/4	0/0
30	1/1	1/1	0/0
31	1/1	1/1	0/0
32	1/1	1/1	0/0
33	1/1	1/1	0/0
140	1/0	1/0	0/0
384	1/0	1/0	0/0
157	0/0	0/0	0/0
158	0/0	0/0	0/0
159	3/0	3/0	0/0
128	0/0	0/0	0/0
127	0/0	0/0	0/0
125	0/0	0/0	0/0
124	0/0	0/0	0/0
123	0/4	0/4	0/0
120	0/4	0/4	0/0
126	3/0	3/0	0/0
216	1/1	1/1	[illegible]/0

541 092

TABLE 4-2 (Cont)

System/ Problem No.	Total No. of Nozzles/ Penetrations	No. Acceptable After Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
203	3/0	3/0	0/0
2031	0/0	0/0	0/0
152	0/0	0/0	0/0
121(2)	3/0	0/0	0/0
122	0/0	0/0	0/0
165	0/0	0/0	0/0
652(2)	1/0	0/0	0/0
653(2)	1/0	0/0	0/0
Main Steam			
658	1/1	1/1	0/0
6590	0/0	0/0	0/0
101	0/0	0/0	0/0
659	1/1	1/1	0/0
660	1/1	1/1	0/0
3063	0/0	0/0	0/0
Feed-water			
204	3/0	3/0	0/0
783	1/1	1/1	0/0
784	1/1	1/1	0/0
785	1/1	1/1	0/0
261	0/0	0/0	0/0

541 093

TABLE 4-2 (Cont)

System/ Probiem No.	Total No. of Nozzles/ Penetrations	No. Acceptable 'fter Pipe Stress Re-analysis	No. Requiring Further Re-Analysis
Diesel Generator Exhaust			
651	0/0	0/0	0/0

NOTES:

(1) Not within the short term reanalysis effort.

(2) These problems recently added to the interim scope. Results of the reanalysis are not available at this time.

541 094

SECTION 5

PIPE SUPPORT RESULTS

Table 5-1 summarizes the pipe supports evaluated in the reanalysis program. There are 696 pipe supports on lines within the interim reanalysis effort; of these, 508 have been evaluated and found acceptable and 7 have been modified to be acceptable. A support is considered acceptable if all the load components are lower in magnitude than those for which the support was originally designed. If some load components are greater than the original design load components, the support is reanalyzed using the new loads. Of the total 188 supports requiring reanalysis, 68 have been found to be acceptable based on DBEI+DL, 111 have not been accepted at this time. Of the 111 unacceptable supports, 76 have not been evaluated at this time due to their recent addition to the reanalysis effort. There is sufficient analytical information available for the remaining 35 supports to exercise engineering judgment in determining whether the unacceptable condition will become acceptable.

1. The use of ASME III Section NF faulted allowable stress values for structural members

541 095

2. The use of one time load for snubbers

3. Use of DBEI plus dead load

If a support is unacceptable using any of the above approaches, a modification is required. Table 5-2 identifies those supports where acceptance is based on he future use of the options listed above. Hardware modifications and additions are discussed in Section 6.

With respect to item 3 above, acceptance criteria for pipe support design and analysis are presented in Table 3-1 of this report. As a basis for interim startup of the Beaver Valley Unit 1 facility, supports which do not meet these criteria will be reevaluated using the allowables of ASME III, Subsection NF, Appendix XVII and Appendix F for the design basis earthquake (DBE). The load combinations and a summary of significant allowable stresses to which evaluation will be made under these ASME criteria appear in Table 5-3.

Support designs which are not in accordance with either of these criteria will be suitably modified against the acceptance design criteria of Table 3-1 prior to interim plant operation.

Base plate design criteria and anchor bolt pullout and shear allowable loads are addressed in Section 3. The seismic support loadings which will be

541 096

utilized for the NF evaluation will be the result of either SHOCK3 or NUPIPE evaluations using SSI-ARS.

Summary

The pipe support reanalysis effort which took place between the original issue and Revision 1 of this report includes accepting 97 supports; 68 based on DBEI+DL and 29 based on long-term criteria. Also, one additional modification was necessary for the 14" RHR line off the reactor coolant loop.

541 097

BEAVER VALLEY POWER STATION, UNIT 1

TABLE 5-1

PIPE SUPPORTS SUMMARY

System/ Problem No.	Total No. of Supports	No. Presently Acceptable Based on Reanalysis	No. Acceptable for Interim Operation	Modifications or Additiors Required
Reactor Coolant				
653A	2	2	0	0
653B	16	12	0	4
653C	8	8	0	0
833&8	15	15	0	1
1200	18	15	3	0
1201	19	19	0	0
Safety Injection				
391A	11	11	0	0
2112	8	8	0	0
610	2	2	0	0
613	5	5	0	0
615[2]	11	6	5	0
15	11	7	4	0
1011	19	16	3	0
301[3]	56	0	56	0
213[4]	16	0	0	0
2113[4]	16	0	0	0
Quench Spray				
211	5	5	0	0
212	3	3	0	0

541 098

TABLE 5-1 (Cont)

System/ Problem No.	Total No. of Supports	No. Presently Acceptable Based on Reanalysis	No. Acceptable for Interim Operation	Modifications or Additions Required
228	0	0	0	0
229	0	0	0	0
Recirculation Spray				
612	0	0	0	0
Charging Volume Control				
100	9	9	0	0
102	8	8	0	0
Residual Heat Removal				
255A	8	4	4	0
256	6	6	0	0
14	15	15	0	0
3011	11	10	1	0
616(4)	7	0	0	0
Component Cooling Water				
302	23	23	0	0
303	23	23	0	0
304	33	31	2	0
305	30	29	1	0
306	11	10	1	0
307	10	10	0	0

541 099

TABLE 5-1 (Cont)

System/ Problem No.	Total No. of Supports	No. Presently Acceptable Based on Reanalysis	No. Acceptable for Interim Operation	Modifications or Additions Required
180E[1]	5	5	0	0
181E[1]	5	4	1	0
170C[1]	17	16	1	0
171[1]	15	11	4	0
172[1]	13	12	1	0
173D[1]	15	14	1	0
174D[1]	20	16	4	0
175B[1]	6	5	1	0
176A[1]	5	5	0	0
177[1]	9	9	0	0
178C[1]	14	10	4	0
179[1]	8	8	0	0
183[1]	9	8	1	0
184[1]	14	11	3	0
186A[1]	6	3	3	0
270A[1]	10	6	4	0
215	8	6	2	0
217	10	10	0	1
930	3	3	0	0
931	2	2	0	0
214	5	5	0	0

541 100

TABLE 5-1 (Cont)

System/ Problem No.	Total No. of Supports	No. Presently Acceptable Based on Reanalysis	No. Acceptable for Interim Operation	Modifications or Additions Required
River Water				
1	0	0	0	0
30	2	2	0	0
31	2	2	0	0
32	2	1	1	0
33	2	2	0	0
140	2	2	0	0
384	5	5	0	0
157	3	3	0	0
158	2	2	0	0
159(3)	8	8	0	0
128	1	0	1	0
127	10	4	6	0
125	12	10	2	0
124	13	12	1	0
123	15	12	0	3
120	11	5	6	0
126	7	7	0	0
216	2	2	0	0
203	16	15	1	0
2031	9	9	0	0
121(4)	15	0	0	0

541 101

BEAVER VALLEY POWER STATION, UNIT 1

TABLE 5-1 (Cont)

System/ Problem No.	Total No. of Supports	No. Presently Acceptable Based on Reanalysis	No. Acceptable for Interim Operation	Modifications or Additions Required
122[4]	19	0	0	0
165[4]	1	0	0	0
152	8	7	1	0
652[5]	1	0	1	0
653[5]	2	0	2	0
Main Steam				
658	6	6	0	0
6590	3	3	0	0
101	4	4	0	0
659	2	1	1	0
660	7	7	0	0
3063	0	0	0	0
Feedwater				
204	15	15	0	0
783	9	9	0	0
784	6	6	0	0
785	3	3	0	0
261	6	6	0	0
Diesel Generator Exhaust				
651[4]	2	0	0	0

NOTES:

[1] Supports are no longer in scope for interim startup. See Appendix B.

541 102

TABLE 5-1 (Cont)

[2] Problem 615 contains modifications (NPSH) scheduled for installation during the first refueling outage. Therefore, it will only be analyzed for interim operation.

[3] Problem 159 includes Problems 160 and 161.

[4] Problem recently added to scope and result not available.

[5] Analyzed based on DBEI+DL only.

541 103

BEAVER VALLEY POWER STATION, UNIT 1

TABLE 5-2

ENGINEERING EVALUATION OF REMAINING SUPPORTS

Problem No.	Support No.	Overstress Condition	Resolution
SAFETY INJECTION SYSTEM			
615	A37	Pad to Run Pipe Weld Overstressed	Will be Acceptable Based on DBEI+DL
	R61	Frame Overstressed	Will be Acceptable Based on DBEI+DL
	HSS-211	Member Overstressed	Will be Acceptable Based on DBEI+DL
	HSS-212A	Snubber Overloaded	Will be Acceptable Based on DBEI+DL
	HSS-212B	Snubber Overloaded	Will be Acceptable Based on DBEI+DL
15	H2	Member/Base Plates/Bolt Pullout	Will be Acceptable Based on DBEI+DL
	H8	Lug to Pipe Weld Overstress/ Bolt Pullout	Will be Acceptable Based on DBEI+DL
	H102A	Snubber Overloaded	Will be Acceptable Based on DBEI+DL
	H102B	Snubber Overloaded	Will be Acceptable Based on DBEI+DL
1011	R13	Member Overstressed	Will be Acceptable Based on DBEI+DL
	R14	Member Overstressed	Will be Acceptable Based on DBEI+DL
	R16	Member Overstressed	Will be Acceptable Based on DBEI+DL
RESIDUAL HEAT REMOVAL SYSTEM			
255	H11	Member Overstress	Will be Acceptable Based on DBEI+DL
	H16	Member Overstress	Will be Acceptable Based on DBEI+DL
	H21	Member Overstress	Will be Acceptable Based on DBEI+DL
	H22	Member Overstress	Will be Acceptable Based on DBEI+DL
3011	H10A.4	Weld Overstressed/Bolt Pullout	Will be Acceptable Using SSI-ARS Curve

541 104

TABLE 5-2 (Cont)

Problem No.	Support	Overstress Condition	Resolution
COMPONENT COOLING WATER SYSTEM			
304	R182	Bolt Pullout/Member Overstressed	Will be Acceptable Based on DBEI+DL
	R10D.6	Local Stress	Will be Acceptable Based on DBEI+DL
305	R176	Bolt Pullout	Will be Acceptable Based on DBEI+DL
306	R264	Local Stress/Trunnion Overstress	Will be Acceptable Based on DBEI+DL
215	R201	Bolt Pullout	Will be Acceptable Based on DBEI+DL
	R203	Member Overstressed	Will be Acceptable Based on DBEI+DL

The following problems are not required for interim startup:

170,171,172,173,174,175,176,
177,178,179,180,181,183,184,
186 & 270. Refer to Appendix B.

Problem No.	Support	Overstress Condition	Resolution
RIVER WATER SYSTEM			
127	H56	Member Overstressed	Will be Acceptable Based on DBEI+DL
	H63	Member Overstressed	Will be Acceptable Based on DBEI+DL
125	H57	Bolt Pullout	Will be Acceptable Based on DBEI+DL
	H49	Bolt Pullout	Will be Acceptable Based on DBEI+DL
124	H28A	Weld Overstressed	Will be Acceptable Based on DBEI+DL

541 105

TABLE 5-2 (Cont)

Problem No.	Support No.	Overstress Condition	Resolution
COMPONENT COOLING WATER SYSTEM			
304	R182	Bolt Pullout/Member Overstressed	Will be Acceptable Based on DBEI+DL
	R10D.6	Local Stress	Will be Acceptable Based on DBEI+DL
305	R176	Bolt Pullout	Will be Acceptable Based on DBEI+DL
306	R264	Local Stress/Trunnion Overstress	Will be Acceptable Based on DBEI+DL
215	R201	Bolt Pullout	Will be Acceptable Based on DBEI+DL
	R203	Member Overstressed	Will be Acceptable Based on DBEI+DL

The following problems are not required for interim startup:

170,171,172,173,174,175,176, 177,178,179,180,181,183,184, 186 & 270. Refer to Appendix B.

Problem No.	Support No.	Overstress Condition	Resolution
RIVER WATER SYSTEM			
127	H56	Member Overstressed	Will be Acceptable Based on DBEI+DL
	H63	Member Overstressed	Will be Acceptable Based on DBEI+DL
125	H57	Bolt Pullout	Will be Acceptable Based on DBEI+DL
	H49	Bolt Pullout	Will be Acceptable Based on DBEI+DL
124	H28A	Weld Overstressed	Will be Acceptable Based on DBEI+DL

541 106

Following reanalysis of Problem No. 833, an additional snubber was designed and will be installed to alleviate a pipe overstress occurring under upset (OBE) and faulted (DBE) conditions.

Similarly, an additional snubber was designed and will be installed in Problem No. 217 to alleviate a pipe overstress occurring under the same conditions.

Three supports in Problem No. 123 will be modified, one to make the as-built condition agree with the original design, one to strengthen a marginal original design, and one to alleviate an overstressed weld in the support resulting from seismic uplift forces.

Similarly, four supports in Problem No. 653B will be modified, three to make the as-built condition agree with the original design, and one to alleviate an overstressed member in the support resulting from seismic forces.

541 107

SECTION 7

LONG TERM REANALYSIS PROGRAM

The long term reanalysis program will consist of preparing completely documented calculational packages, utilizing the NUPIPE computer program with amplified response spectra (ARS) based on soil-structure interaction (SSI), for the problems identified in Appendixes A and B. In addition, the problems associated with the quench and recirculation spray systems will be analyzed for the operating basis earthquake.

ANCHOR MOVEMENT CRITERIA

Pipe stress analysis for Beaver Valley Unit 1 was performed in accordance with the ANSI B31.1 Power Piping Code - 1967. In formulating load combinations to meet paragraphs 102.3.3(a) and (d), seismic anchor displacement effects were included with seismic inertia effects to form total seismic response for the DBE case.

Inclusion of the DBE anchor displacement effects in combination with the DBE inertia effects is not a requirement of current codes, neither ANSI B31.1 or ASME III NC or ND3600, since the displacement effect is secondary in nature.

541 108

Under the long term piping analysis criteria established for Beaver Valley Unit 1, anchor displacement effects need not be combined with the inertia effects of the DBE event when evaluating primary stresses in the system.

541 109

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

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

8.1 STRESS LIMITS

The analyses and reanalyses of Seismic Category I piping systems are based upon the conservative stress limit of $1.8S_h$ under the limiting faulted or DBE loading conditions. The present ASME Section III Code specifies the piping stress limit to be $2.4S_h$ under the Faulted DBE Condition. Only the quench and

541 110

recirculation spray systems were redesigned and reanalyzed in 1975 for the DBE condition including water hammer loads using $2.4S_h$ as an allowable. In July 1978, the 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 stress exceeds 1.5 to 2.0 times the yield strength (S_y) of piping material, but for stresses below these values, functional capability was maintained.

For Beaver Valley, Unit 1, the majority of carbon steel piping material is of SA-106 Grade B steel. Using the lower limit of $1.5S_y$ from NUREG/CR-0261 and representative properties of SA-106 Grade B steel, the added margin 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 Beaver Valley Unit 1 pipe stress reanalysis calculations have included the seismic stress due to anchor displacements in the DBE condition. Inclusion of the anchor movement stresses was not explicitly required by ANSI Code B31.1, used for the original design, and is not required by current 1979 codes, for the faulted DBE condition. Addition of this stress component is a significant conservatism for the long term reanalysis.

* 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

541 111

8.4 FIELD VERIFICATION OF AS-BUILT CONDITIONS

The documentation of as-built conditions for Beaver Valley Unit 1 began in September 1974 and was completed prior to startup. The effort was manned by Pipe Stress Analysis and Pipe Support Designers who walked all Category I piping systems to ensure compliance with the stress analysis summaries (MSKs). All Category I piping was checked for piping configuration, pipe support location, and pipe support type. The results of this effort were documented and reported on Southwest Fabricating isometric drawings which then became part of the permanent plant record. These isometric drawings supersede the RP series drawings. Duquesne Light Co. personnel verified the accuracy of a portion of these isometric drawings during March and April of 1979 subsequent to the shutdown order.

8.5 ENGINEERING ASSURANCE

A comprehensive and extensive Engineering Assurance program has been developed and applied to the reanalysis activities. A detailed project procedure was developed that includes provisions for design control, document control, and interface controls. Each new project procedure developed received a full review and approval by the S&W Engineering Assurance (EA) staff.

541 112

APPENDIX A

SYSTEMS AFFECTED FOR INTERIM STARTUP

The reanalysis for interim startup includes those lines originally computer analyzed with the SHOCK2 code and which are necessary for safe shutdown. In order to evaluate safety system lines which have interconnecting ties with nonsafety system lines, the reanalysis was extended to include lines attached to the safety systems, past the first automatic trip valve or the first normally closed manual valve, to the first piping anchor.

System	Line Number	Problem No.	FSAR Fig. No.
Reactor Coolant (RC)	29"-RC-4-2501R-Q1	653A	4-1
	27.5"-RC-6-2501R-Q1		
	31"-RC-5-2501R-Q1		
	8"-RC-29-2501R-Q1		
	8"-RC-27-2501R-Q1		
	31"-RC-8-2501R-Q1	653C	4-1
	29"-RC-7-2501R-Q1		
	27.5"-RC-9-2501R-Q1		
	8"-RC-37-2501R-Q1		
	8"-RC-39-2501R-Q1		
	14"-RC-86-2501R-Q1		4-1,4-2
	29"-RC-1-2501R-Q1	653B	4-1
	27.5"-RC-3-2501R-Q1		
	31"-RC-2-2501R-Q1		
	8"-RC-17-2501R-Q1		
	8"-RC-19-2501R-Q1		
	12"-RC-111-602	833 & 8	4-2
	6"-RC-100-602		
	6"-RC-101-602		
	6"-RC-102-602		
	6"-RC-108-602		
	6"-RC-104-1502-Q1		
	6"-RC-97-1502-Q1		
	6"-RC-98-1502-Q1		
	6"-RC-99-1502-Q1		
	3"-RC-105-1502-Q1		
	3"-RC-106-1502-Q1		
	3"-RC-107-1502-Q1		
	4"-RC-71-1502-Q1	1200	4-1,4-2
	4"-RC-72-1502-Q1		
	4"-RC-71-1502-Q1	1201	4-1,4-2
	4"-RC-72-1502-Q1		

541 113

System	Line Number	Problem No.	FSAR Fig. No.
Safety Injection (SI)	6"-SI-41-153W-Q2	100	9.1-1
	6"-SI-42-153W-Q2		
	6"-SI-40-153W-Q2		
	12"-SI-110-602-Q1	653C	6.3-2
	12"-SI-111-1502-Q2		4-1, 6.3-2
	6"-SI-40-153W-Q2	102	9.1-1
	6"-SI-44-153W-Q2		
	8"-SI-2-153W-Q2		
	8"-SI-2-153W-Q3	2112	6.3-1
	8"-SI-2-153W-Q3	1011	6.3-1, 9.1-1
	12"-SI-5-153A-Q2	610	6.3-1, 6.3-2
	12"-SI-6-153A-Q2		
	12"-SI-7-153A-Q2		
	12"-SI-8-153A-Q2		
	12"-SI-13-153A-Q2		
	12"-SI-6-153A-Q2	613	6.3-1, 6.3-2
	12"-SI-1-153W-Q2		
	10"-SI-15-1502-Q1	615	
	10"-SI-16-153W-Q2		
	10"-SI-17-153W-Q2		
	10"-SI-18-1502-Q1		
	10"-SI-26-153W-Q2		
	10"-SI-27-153W-Q2		
	10"-SI-28-1502-Q1		
	6"-SI-32-1502-Q1		
	6"-SI-33-1502-Q1		
	6"-SI-34-1502-Q1	615	6.3-1, 6.3-2
	6"-SI-40-153W-Q2		
	6"-SI-44-153W-Q2		
	12"-SI-121-1502-Q1	15	6.3-2,4-1
	12"-SI-108-602-Q2		
	12"-SI-1-153W-Q3	391A	6.3-1, 6.4-1
	8"-SI-2-153W-Q3		6.3-1
	12"-SI-101-1502-Q1	14	6.3-2,4-1
	10"-RH-23-1502-Q1		9.3-1
	12"-SI-120-602-Q2		
	10"-RH-16-602-Q2		
	6"-SI-30-1502-Q1	301	6.3-2

System	Line Number	Problem No.	FSAR Fig. No.
	6"-SI-29-1502-Q1 6"-SI-20-1502-Q1 6"-SI-19-1502-Q1 6"-SI-32-1502-Q1 6"-SI-33-1502-Q1		
	6"-SI-40-153W-Q2	2113	6.3-1
	6"-SI-44-153W-Q2	213	6.3-1
Quench Spray (QS)	12"-QS-2-153B-Q3	211	6.4-1
	12"-QS-1-153B-Q3	212	6.4-1
	12"-QS-1-153B-Q3	228	6.4-1
	12"-QS-2-153B-Q3	229	6.4-1
Recirculation Spray (RS)	12"-RS-7-153A-Q2 12"-RS-8-153A-Q2 12"-RS-5-153A-Q2	612	6.4-1
Charging and Volume Control (CH)	6"-CH-63-153W-Q2 6"-CH-67-153W-Q2 8"-CH-15-153W-Q2	100	9.1-1
	8"-CH-15-153W-Q2 6"-CH-68-153W-Q2	102	
Residual Heat Removal (RH)	12"-RH-6-602-Q2 12"-RH-9-602-Q2 12"-RH-12-602-Q2 10"-RH-4-602-Q2 10"-RH-5-602-Q2 10"-RH-7-602-Q2 10"-RH-8-602-Q2 10"-RH-10-602-Q2 10"-RH-19-602-Q2	255A	9.3-1
	12"-RH-9-602-Q2 12"-RH-12-602-Q2 10"-RH-16-602-Q2 10"-RH-17-602-Q2 6"-RH-20-602-Q2 3"-RH-13-602-Q2	256	9.3-1
	10"-RH-16-602-Q2	3011	9.3-1

541 115

System	Line Number	Problem No.	FSAR Fig. No.
	14"-RH-1-1502-Q1 14"-RH-2-602-Q2 14"-RH-18-602-Q2	653B	4-1,9.3-1
	10"-RH-24-1502-Q1	653C	6.3-2
	10"-RH-23-1502-Q2	14	9.3-1, 6.3-2
	6"-RH-14-152-Q2	616	9.3-1
Component Cooling	18"-CC-118-151-Q3	302	9.4-4
	18"-CC-116-151-Q3	303	9.4-4
	18"-CC-114-151-Q3	304	9.4-4
	18"-CC-130-151-Q3	305	9.4-4
	8"-CC-255-151-Q3 8"-CC-256-151-Q3 8"-CC-257-151-Q3 6"-CC-261-151-Q3 8"-CC-476-151-Q3	306	9.4-3
	6"-CC-258-151-Q3 6"-CC-265-151-Q3 8"-CC-259-151-Q3 8"-CC-260-151-Q3 8"-CC-517-151-Q3	307	9.4-3
	6"-CC-519-151-Q3 24"-CC-125-151-Q3 18"-CC-489-151-Q2 18"-CC-490-151-Q2 18"-CC-529-151-Q3 18"-CC-530-151-Q3 6"-CC-488-151-Q2 6"-CC-526-151-Q3 4"-CC-487-151-Q3 4"-CC-525-151-Q2 3"-CC-486-151-Q3 3"-CC-523-151-Q2 2"-CC-485-151-Q2 2"-CC-524-151-Q3 24"-CC-266-151-Q3 6"-CC-518-151-Q3	215	9.4-4
	24"-CC-112-151-Q3 24"-CC-113-151-Q3	217	9.4-3, 9.4-4

541 116

System	Line Number	Problem No.	FSAR Fig. No.
	6"-CC-510-151-Q3 6"-CC-511-151-Q3 6"-CC-512-151-Q3 6"-CC-482-151-Q3 18"-CC-483-151-Q3 18"-CC-484-151-Q3 18"-CC-527-151-Q3 18"-CC-528-151-Q3		9.4-3, 9.4-4
	8"-CC-517-151-Q2	214	9.4-3
	6"-CC-481-151-Q2 6"-CC-511-151-Q3	930	9.4-4
	6"-CC-480-151-Q2 6"-CC-510-151-Q3	931	9.4-4
Chilled Water (CW)	8"-CW-8-151	216	9.4-3
	8"-CW-9-151	214	9.4-3
River Water (WR)	6"-WR-117-151-Q3	203	10.3-5
	14"-WR-64-151-Q2	30	9.9-1A
	14"-WR-82-151-Q2	31	9.9-1A
	14"-WR-89-151-Q2	32	9.9-1A
	14"-WR-87-151-Q2	33	9.9-1A
	8"-WR-228-151-Q3 8"-WR-229-151-Q3 8"-WR-230-151 8"-WR-231-151	140	9.9-1.
	8"-WR-234-151-Q3	214	9.4-3
	14"-WR-63-151-Q2 14"-WR-65-151-Q2 14"-WR-86-151-Q2 14"-WR-88-151-Q2	1	9.9-1A

System	Line Number	Problem No.	FSAR Fig. No.
	14"-WR-25-151-Q3 14"-WR-26-151-Q3 14"-WR-27-151-Q3 14"-WR-28-151-Q3 24"-WR-29-151-Q3	120	9.9-1A
	14"-WR-21-151-Q3 14"-WR-22-151-Q3 14"-WR-23-151-Q3 14"-WR-24-151-Q3 24"-WR-19-151-Q3 24"-WR-20-151-Q3	123	9.9-1A
	24"-WR-19-151-Q3 24"-WR-187-151-Q3	124	9.9-1A
	24"-WR-20-151-Q3 24"-WR-186-151-Q3	125	9.9-1A
	24"-WR-7-151-Q3 24"-WR-8-151-Q3 24"-WR-9-151-Q3 18"-WR-11-151-Q3 18"-WR-12-151-Q3 18"-WR-13-151-Q3	126	9.9-1A
	24"-WR-19-151-Q3	127	9.9-1A
	24"-WR-20-151-Q3	128	9.9-1A
	24"-WR-99-151-Q3	157	9.9-1A
	24"-WR-100-151-Q3	158	9.9-1A,B
	20"-WR-1-151-Q3 20"-WR-2-151-Q3 20"-WR-3-151-Q3 20"-WR-4-151-Q3 20"-WR-5-151-Q3 20"-WR-6-151-Q3 24"-WR-99-151-Q3 24"-WR-100-151-Q3 18"-WR-154-151-Q3 12"-WR-177-151-Q3	159(1)	9.9-1A,B
	8"-WR-227-151	216	9.4-3, 9.9-1A
	30"-WR-171-151-Q3	152 547 118	9.9-1B

System	Line Number	Problem No.	FSAR Fig. No.
	30"-WR-172-151-Q3		
	30"-WR-175-151-Q3		
	10"-SWW-14-151-Q3	165(2)	9.9-1B
	10"-SWW-1-121		
	18"-WR-14-151-Q3	121	9.9-1A
	18"-WR-15-151-Q3		
	18"-WR-16-151-Q3		
	30"-WR-17-151-Q3	122	9.9-1A
	6"-WR-155-151-Q3	384	9.9-1B
	6"-WR-214-151-Q3	652	RM-53A
	6"-WR-215-151-Q3	653	RM-53A
Main Steam (MS)	3"-SDHV-1-601-Q2	101	10.3-1
	3"-SDHV-2-601-Q2		
	3"-SDHV-3-601-Q2		
	4"-SDHV-4-601-Q2		
	32"-SHP-56-601-Q2	658	10.3-1
	32"-SHP-57-601-Q2	659	10.3-1
	32"-SHP-58-601-Q2	660	10.3-1
	4"-SHP-19-601-Q2	6590	10.3-1
	4"-SHP-20-601-Q2		
	4"-SHP-21-601-Q2		
	6"-SAE-1-601		
	6"-SAE-2-601		
	6"-SAE-3-601		
	32"-SHP-56-601-Q2	3063	10.3-1
	32"-SHP-57-601-Q2		
	32"-SHP-58-601-Q2		
	32"-SHP-22-601-Q2		
	32"-SHP-23-601-Q2		
	32"-SHP-24-601-Q2		
	10"-SSVD-[illegible]-601		
	10"-SSVD-2-601		
	10"-SSVD-3-601		
	10"-SSVD-4-601		
	10"-SSVD-5-601		
	10"-SSVD-6-601		
	10"-SSVD-7-601		
	10"-SSVD-8-601		

541 119

System	Line Number	Problem No.	FSAR Fig. No.
	10"-SSVD-9-601 10"-SSVD-10-601 10"-SSVD-11-601 10"-SSVD-12-601 10"-SSVD-13-601 10"-SSVD-14-601 10"-SSVD-15-601		
Main and Auxiliary Feedwater (FW)	4"-WAPD-3-601-Q3 4"-WAPD-4-601-Q3 4"-WAPD-5-601-Q3 4"-WAPD-6-601-Q3 6"-WAPD-1-601-Q3 6"-WAPD-2-601-Q3	204	10.3-5
	16"-WFPD-22-601-Q2	783	10.3-5
	16"-WFPD-24-601-Q2	784	10.3-5
	16"-WFPD-23-601-Q2	785	10.3-5
	16"-WFPD-9-601-Q2 16"-WFPD-13-601-Q2 16"-WFPD-17-601-Q2	0261	10.3-5
	6"-WD-23-151-Q3 6"-WD-24-151-Q3 6"-WD-25-151-Q3 6"-WD-26-151-Q3 4"-WD-27-151-Q3 4"-WD-41-151-Q3	203	10.3-5
	8"-WD-22-151-Q3 6"-WD-25-151-Q3 6"-WD-26-151-Q3	2031	10.3-5
Diesel Generator Exhaust (OL)	22"-OL-55-151-Q3	651	RM-53A

NOTES:

(1) Problems 160 and 161 are included within the scope of the reanalysis effort for problem 159.

(2) Problem 165 has been analyzed on NUPIPE as part of the Beaver Valley Unit 2 stress analysis effort.

These lines are identified on the flow diagrams included in Appendix C.

In addition to the problems referenced above, a number of other computer analyses were also performed for Beaver Valley - Unit 1, using the SHOCK2 code. These have been excluded from the scope of the reanalysis for interim startup and are discussed in Appendix B.

541 12

APPENDIX B

PROBLEMS TO BE REANALYZED IN THE LONG TERM

The problems described in Tables B-1, B-2, and B-3 are within the scope of the long term effort. These problems are identified on the flow diagrams included in Appendix D.

1. Primary Component Cooling Water System

The primary component cooling water (CC) system is used during normal operation and cooldown to remove heat from various primary plant components; however, safe shutdown of the reactor (i.e., hot standby) can be achieved without dependence on the CC system. By use of other systems the plant can be maintained in this condition indefinitely while restoration of the CC system is being accomplished.

The responses to the FSAR questions listed in the references below discuss in detail the effects of various CC system pipe breaks and their impacts on the operation of the plant. These discussions demonstrate the capability of the plant to maintain a safe shutdown condition without the availability of the CC system. The response to FSAR Question 9.33

541 122

describes the method of repair and the expected repair times for cracks and breaks of various severities. As mentioned in that response, the worst case repair (the addition of a piece of pipe) can be made within 5 days.

In summary, problems associated with the CC system outside the containment are included in the long-term reanalysis effort because the CC system is not required either to attain or maintain a safe shutdown condition. Table B-1 identifies the CC problems to be addressed in the long term.

References: FSAR Section 9.4.

FSAR Questions 9.2, 9.10, 9.11, 9.13, 9.33, 9.34, and 9.35.

2. Other Safety Systems

Table B-2 identifies the SHOCK2 problems that are within the scope of the long-term reanalysis effort; these lines are not required for safe shutdown.

3. Hand Calculations

Table B-3 identifies SHOCK2 problems that are not within the scope of the interim startup or long-term reanalysis effort; these SHOCK2 runs are only

541 123

check calculations of manual hand calculations. They are identified here only to show the scope of the original SHOCK2 effort.

4. Superseded Calculations

The following SHOCK2 runs have been superseded by a problem presently within the interim and long term reanalysis effort.

Superseded SHOCK2 Run	New Problem Number
122A	122
312	840
657	785
916	217
1012	391
2110	341B
6230	310

5. Seismically Supported Non-Q Lines

The following lines are not safety related but have been seismically supported as designated by an "E" in the line designation table.

2"-CV-1-154

2"-SHPD-5-601

2"-SHPD-6-601

2"-SHPD-7-601

2"-SHPD-8-601

1/4-SS-163-N9

1/4-SS-173-N9

1/4-SS-174-N9

TABLE B-2

SAFETY SYSTEMS TO BE ANALYZED IN THE LONG TERM

System	Line Number	Problem No.	FSAR Fig. No.
Fuel Pool Cooling & Purification System (FC)	6"-FC-4-152-Q3 6"-FC-5-152-Q3	104	
	6"-FC-8-152-Q3 6"-FC-9-152-Q3	105E	9.5-1
	10"-FC-1-152-Q3 6"-FC-2-152-Q3 6"-FC-31-152-Q3	198B	9.5-1
	4"-FC-10-152 4"-FC-11-152 6"-FC-14-152 6"-FC-17-152 6"-FC-32-152	107	9.5-1
Quench Spray (QS)	10"-QS-4-153B-Q3	614	6.4-1
	10"-QS-3-153B-Q3	617	6.4-1
	4"-QS-6-153B-Q3 10"-QS-4-153B-Q3	210	6.4-1
	4"-QS-5-153B-Q3 10"-QS-3-153B-Q3	218	6.4-1
Recirculation Spray (RS)	4"-RS-14-153B-Q2 10"-RS-10-153B-Q2 4"-RS-15-153B-Q2	611	6.4-1
River Water (WR)	30"-WR-175-151-Q3	153	9.9-1B

541 126

BEAVER VALLEY POWER STATION, UNIT 1

TABLE B-3

HAND CALCULATIONS CHECKED BY SHOCK2

System	Line Number	Problem No.
High Pressure Steam (SHP)	3"-SHP-26-601-Q2 3"-SHP-31-601-Q2	3043
Steam Generator Auxiliary Feedwater Pump Discharge (WAPD)	3"-WAPD-13-601-Q3	207
	3"-WAPD-11-601-Q3	208
Generator Water Blowdown (WGCB)	3"-WGCB-8-601-Q2 3"-WGCB-12-601-Q2	309, 3017, 6220, 3002, 3018, 6216
	3"-WGCB-4-601-Q2	310
	3"-WGCB-4-601-Q2	3100
Fuel Pool Cooling and Purification System (FC)	6"-FC-12-152-Q2	301
	6"-FC-17-152-Q2	655C
Charging and Volume Control System (CH)	3"-CH-125-1503-Q2	911, 260, 3001
	2"-CH-97-1502-Q1	200
	2"-CH-141-1503-Q1	220
	2"-CH-100-1502-Q2 2"-CH-186-152-Q2	230
	2"-CH-1-1502-Q1	240
	2"-CH-96-1502-Q1	250
	2"-CH-23-1502-Q1	300
	2"-CH-143-1502-Q1 2"-CH-149-1502-Q1 2"-CH-145-602-Q1 2"-CH-2-602-Q1 2"-CH-3-602-Q1	350

541 127

TABLE B-3 (Cont)

System	Line Number	Problem No.
	2"-CH-4-602-Q1	
	2"-CH-146-152-Q3	
	3/4"-CH-115-1502-Q2	380
	2"-CH-2-602-Q2	702
	2"-CH-148-602-Q2	703
	3"-CH-106-153W-Q2	901, 3135
	3"-CH-107-153W-Q2	3135
	3"-CH-108-153W-Q2	704, 3135
	3"-CH-110-153W-Q2	704,
	3"-CH-111-153W-Q2	3057
	3"-CH-114-152W-Q2	3129, 3044
	4"-CH-14-153W-Q2	3057
	3"-CH-6-153W-Q2	3122
	3"-CH-226-153W-Q2	
	3"-CH-13-153W-Q2	3125
	4"-CH-72-1503-Q2	3131
	4"-CH-76-1503-Q2	
	3"-CH-71-1503-Q2	
	3"-CH-75-1503-Q2	
	3"-CH-80-1503-Q2	
	3"-CH-69-1503-Q2	3031
	3"-CH-70-1503-Q2	
	3"-CH-73-1503-Q2	
	3"-CH-74-1503-Q2	
	4"-CH-72-1503-Q2	
	4"-CH-76-1503-Q2	
	3"-CH-126-1502-Q1	3035
Safety Injection (SI)	3"-SI-81-1503-Q1&Q2	900, 3004
	3"-SI-140-1503-Q1	902, 3004

541 123

TABLE B-3 (Cont)

System	Line Number	Problem No.
	6"-SI-34-1502-Q1 6"-SI-74-1502-Q1	3006
	3"-SI-60-1503-Q2	3124
	3"-SI-57-1503-Q1&Q2	900
	3"-SI-130-1503-Q1&Q2	313, 902
	3"-SI-134-1503-Q2	922
	3"-SI-81-1503-Q2	3120
	3"-SI-56-1503-Q3 3"-SI-60-1503-Q3 3"-SI-133-1503-Q3 4"-SI-75-1503-Q3 3"-SI-134-1503-Q1	3052
	3"-SI-31-153W-Q2 3"-SI-145-153W-Q2	3127
	3"-SI-35-152-Q3	965
Residual Heat Removal System (RH)	6"-RH-14-152-Q2	3012
Reactor Coolant (RC)	3"-RC-13-1502-Q1 3"-RC-23-1502-Q1 3"-RC-33-1502-Q1 3"-RC-160-153W-Q2	6530
	2"-RC-54-1502-Q1	220
	3"-RC-160-153W	917
	4"-RC-112-152-Q3	360
	3"-RC-160-153W	917
	3"-RC-160-153W-Q2	3021
Component Cooling (CC)	6"-CC-512-151-Q3	914
	4"-CC-487-151-Q2	918

541 129

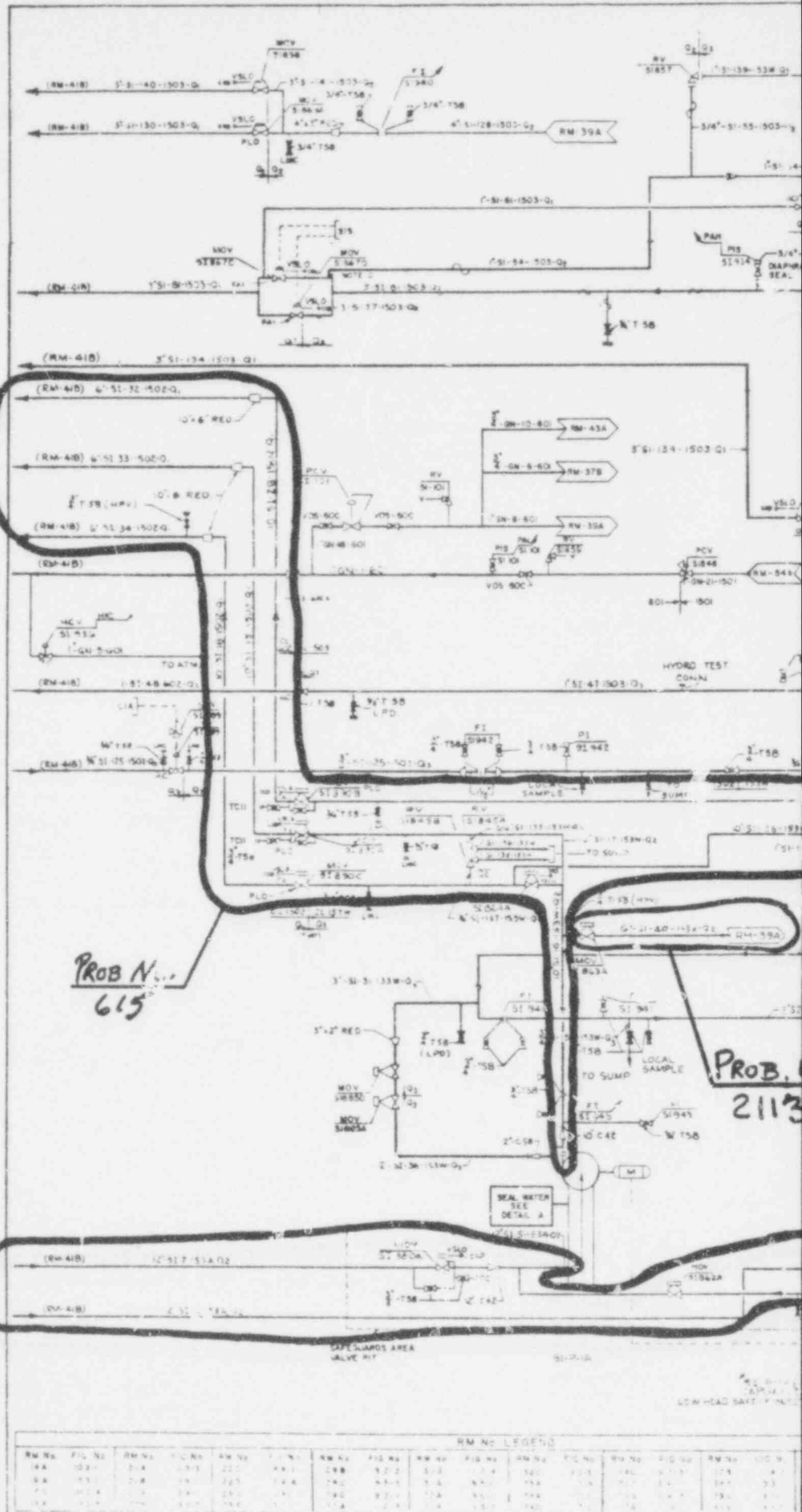
TABLE B-3 (Cont)

System	Line Number	Problem No.
	4"-CC-525-151-Q3	
	3"-CC-235-151-Q3 3"-CC-466-151-Q2 3"-CC-523-151-Q3	921
Diesel Generator Oil Line (OL)	3"-OL-46-151-Q3	650
Primary Grade Water (PG)	3"-PG-5-152	917
Quench Spray (QS)	2"-QS-29-152	315X
	6"-QS-30-153B-Q3 6"-QS-31-153B-Q3	840
	6"-QS-16-152	139
	4"-QS-8-152	341B
Neutron Shield Tank Cooling (NSL)	6"-NSL-2-152-Q3	312

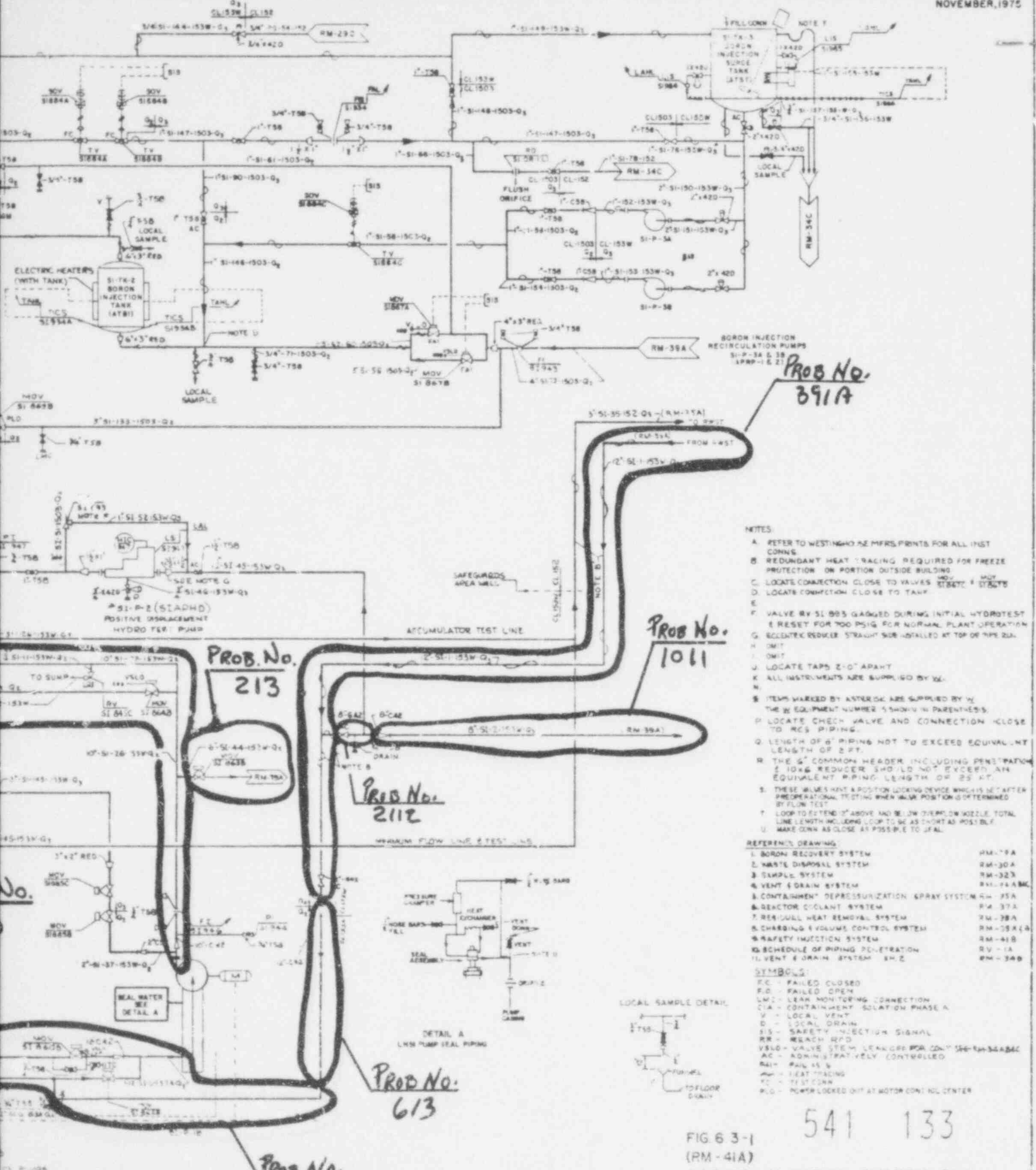
541 130

LIST OF FIGURES

541 131


PROB N...
615
PROB.
211'3
SEAL WATER SEE DETAIL A
SAFEGUARDS AREA VALVE PIT
TO SUMP
LOCAL SAMPLE
HYDRO TEST CONN
RM No LEGEND

541 132

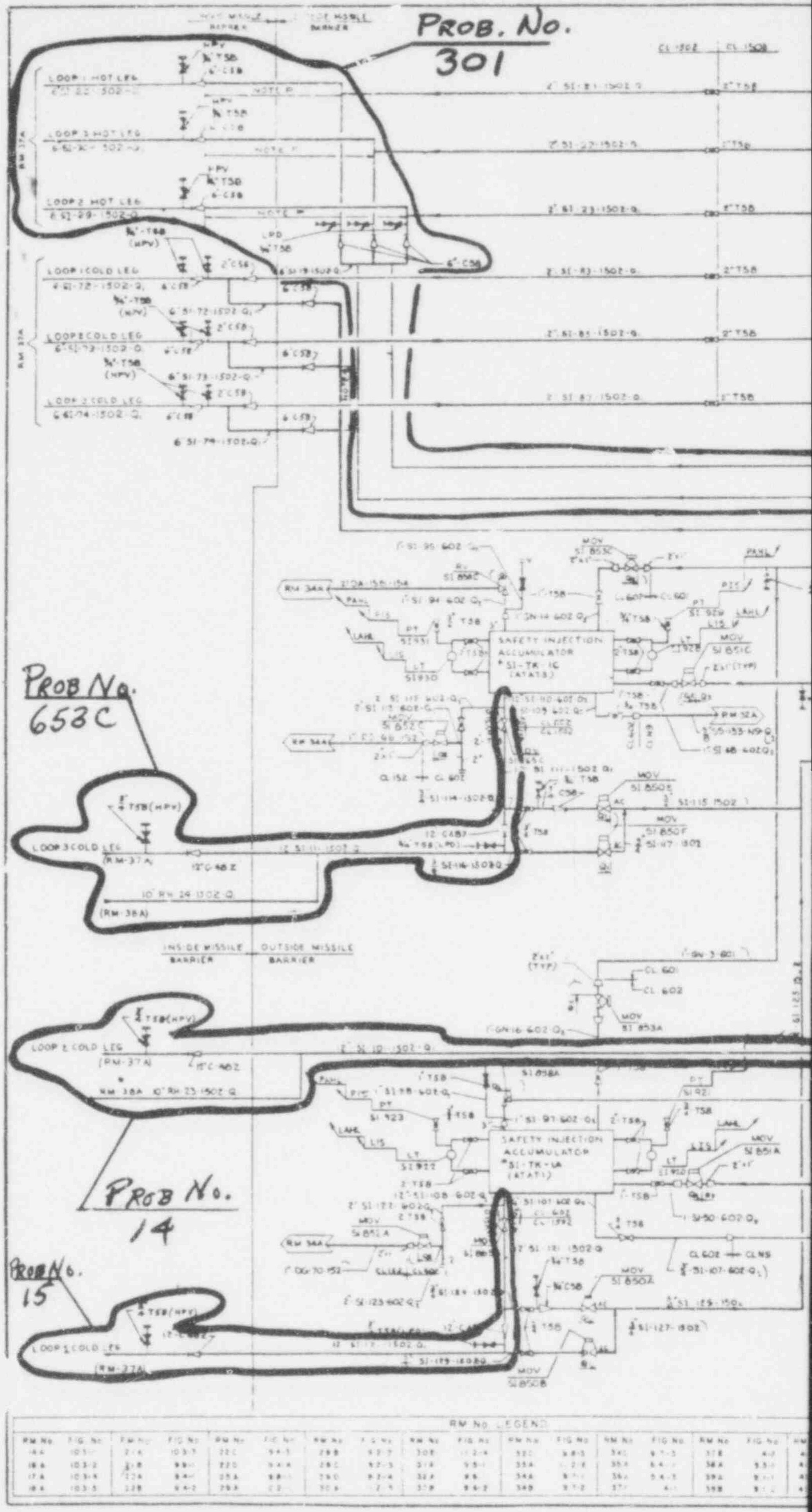


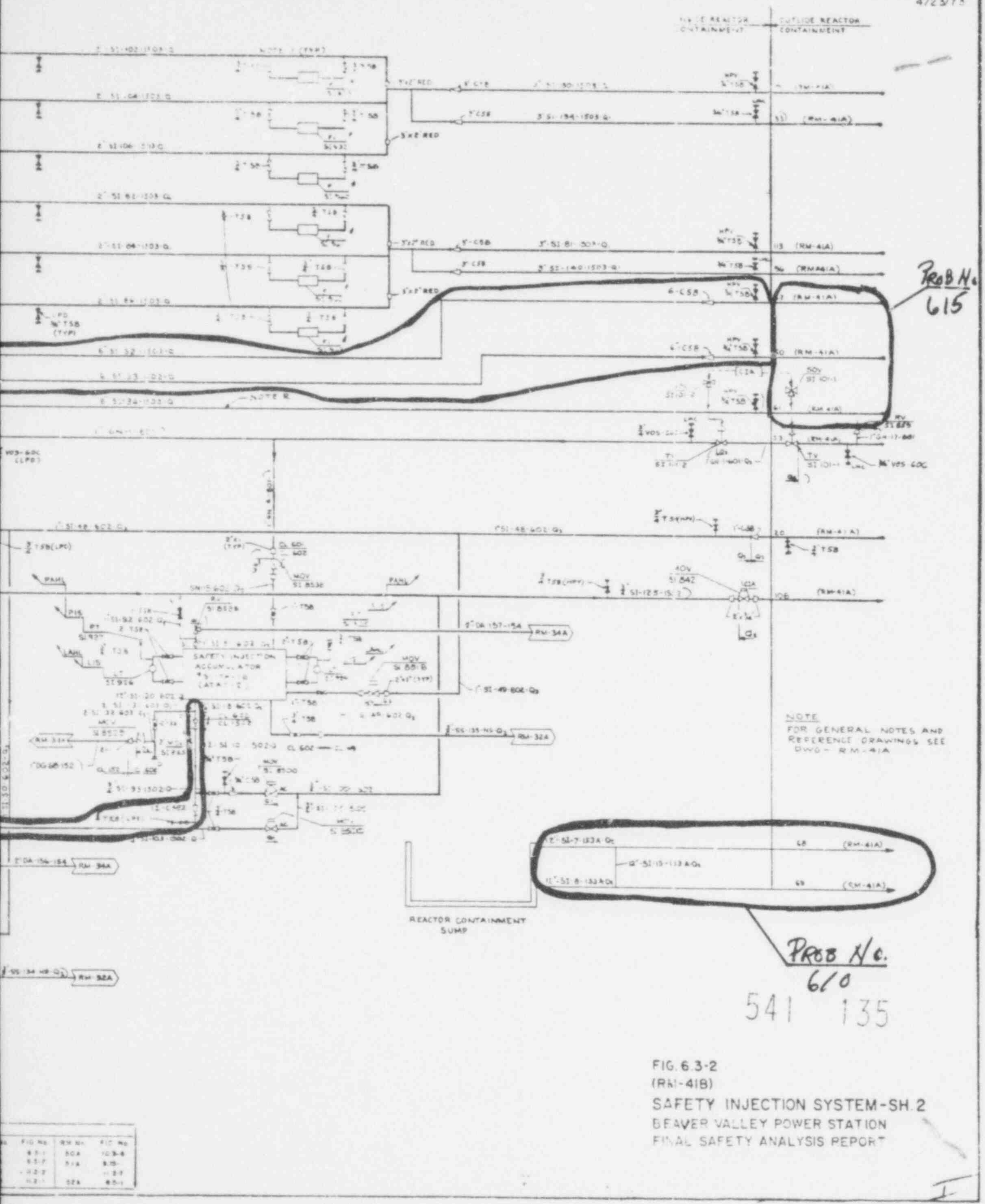
541 133

FIG. 6.3-1
(RM-41A)
SAFETY INJECTION SYSTEM-SH 1
BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

I

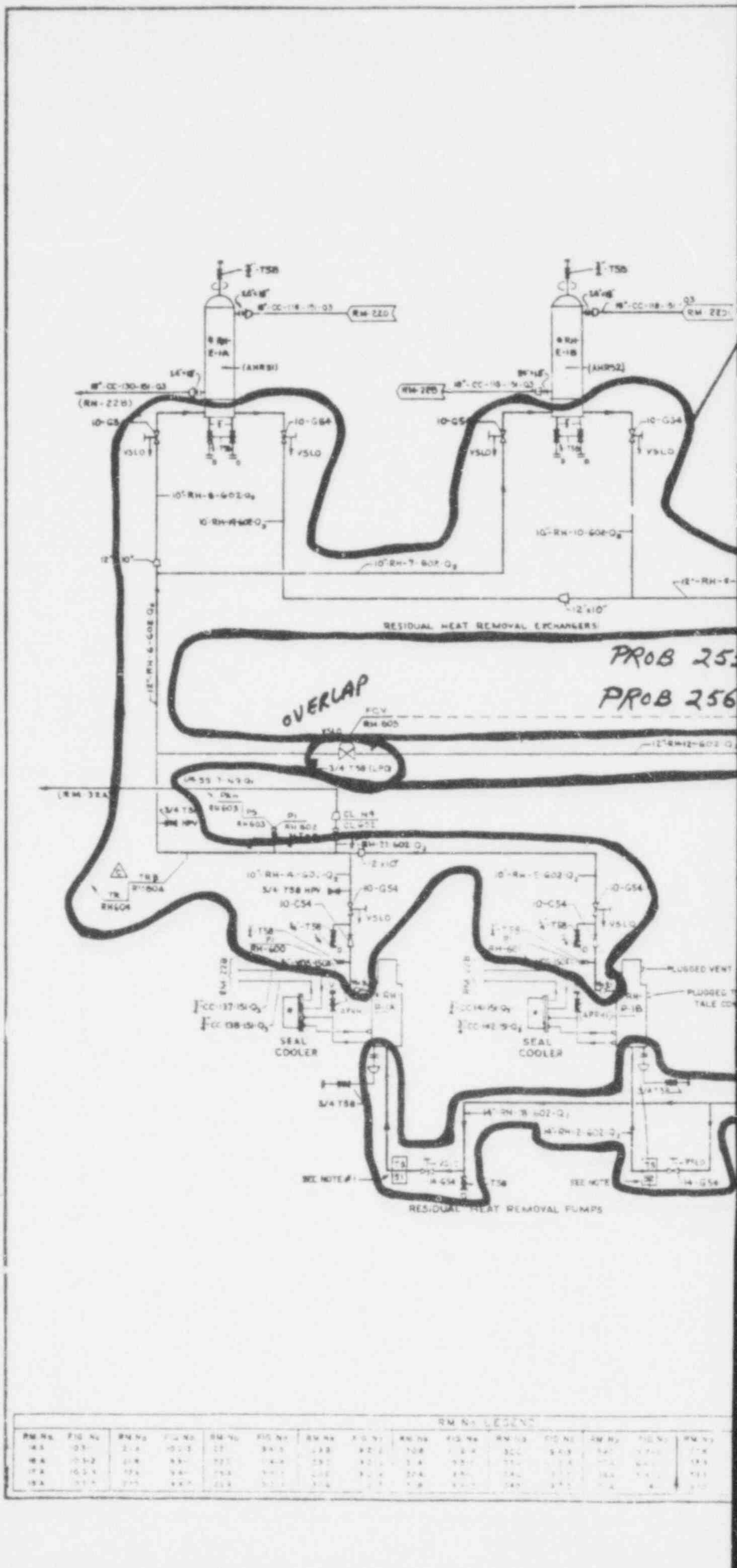
REVISION 1


PROB. No. 301
PROB No. 653C
PROB No. 14
PROB No. 15
LOOP 1 HOT LEG
LOOP 1 COLD LEG
LOOP 2 COLD LEG
LOOP 3 COLD LEG
SAFETY INJECTION ACCUMULATOR
INSIDE MISSILE BARRIER
OUTSIDE MISSILE BARRIER
RM No. LEGEND

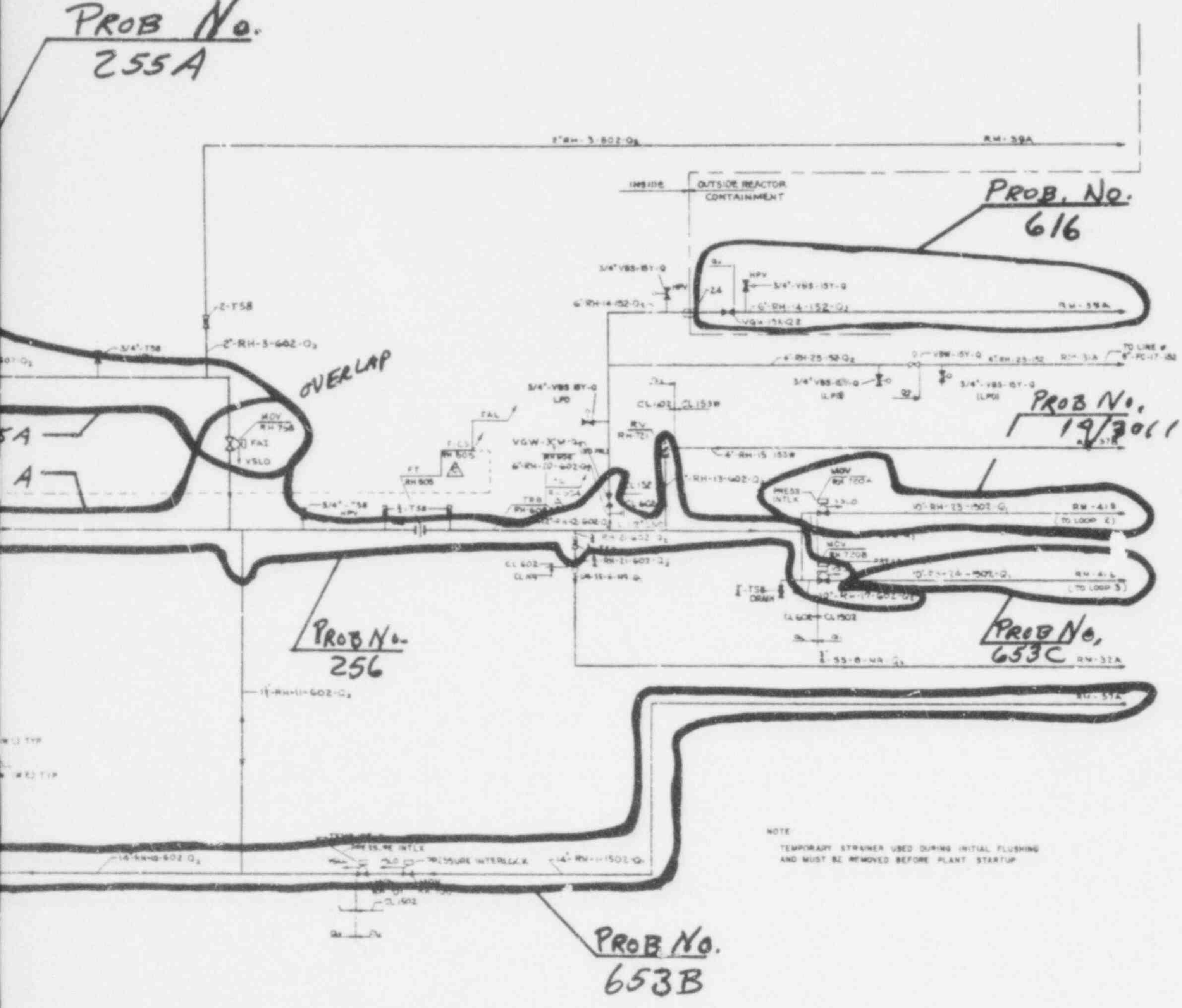


541 135

FIG. 6.3-2
(RM-41B)
SAFETY INJECTION SYSTEM-SH. 2
BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT


RESIDUAL HEAT REMOVAL EXCHANGERS
PROB 25
PROB 256
OVERLAP
SEAL COOLER
SEAL COOLER
PLUGGED VENT
RESIDUAL HEAT REMOVAL PUMPS
RM No LEGEND

541 136

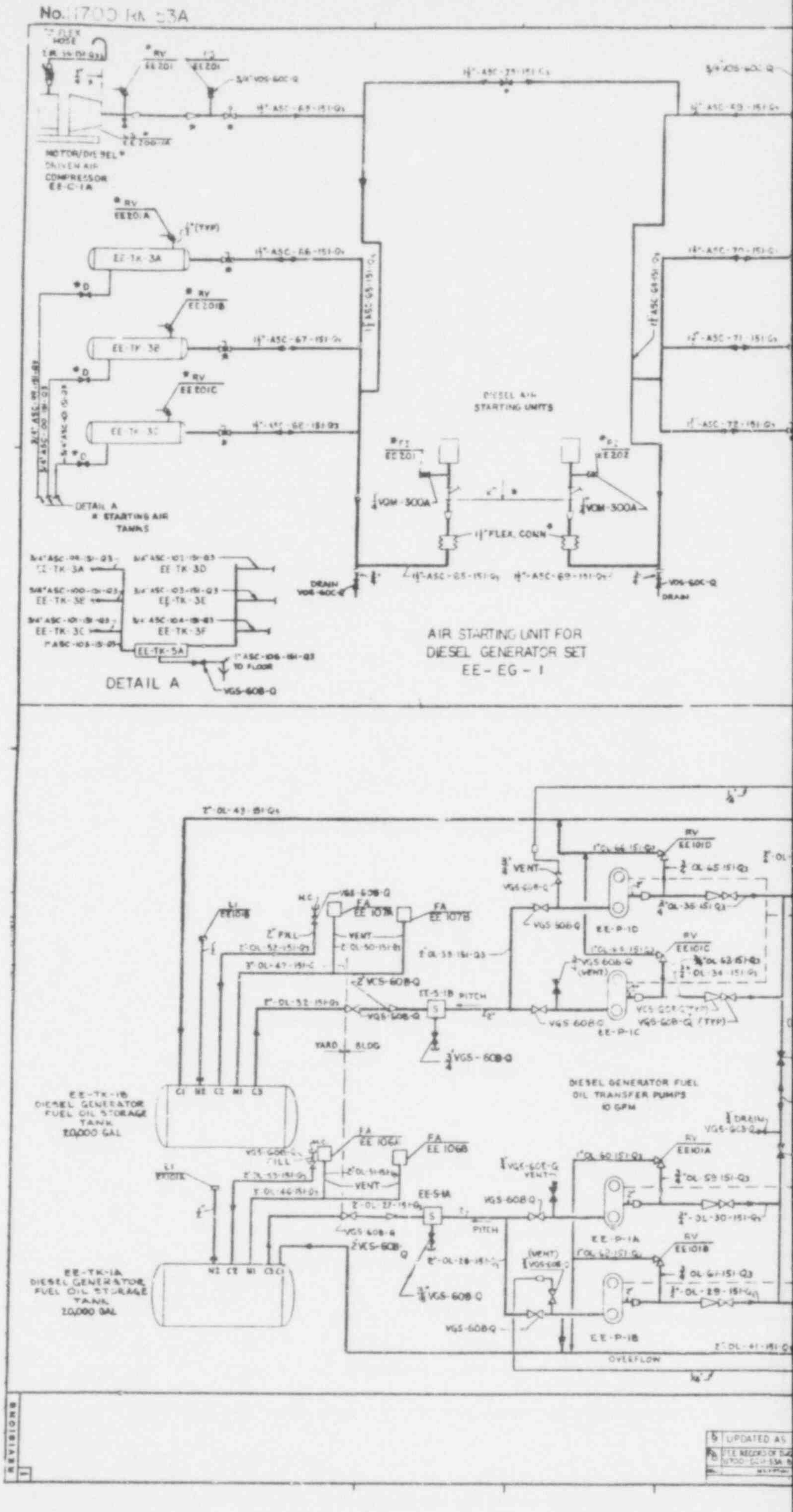


541 137

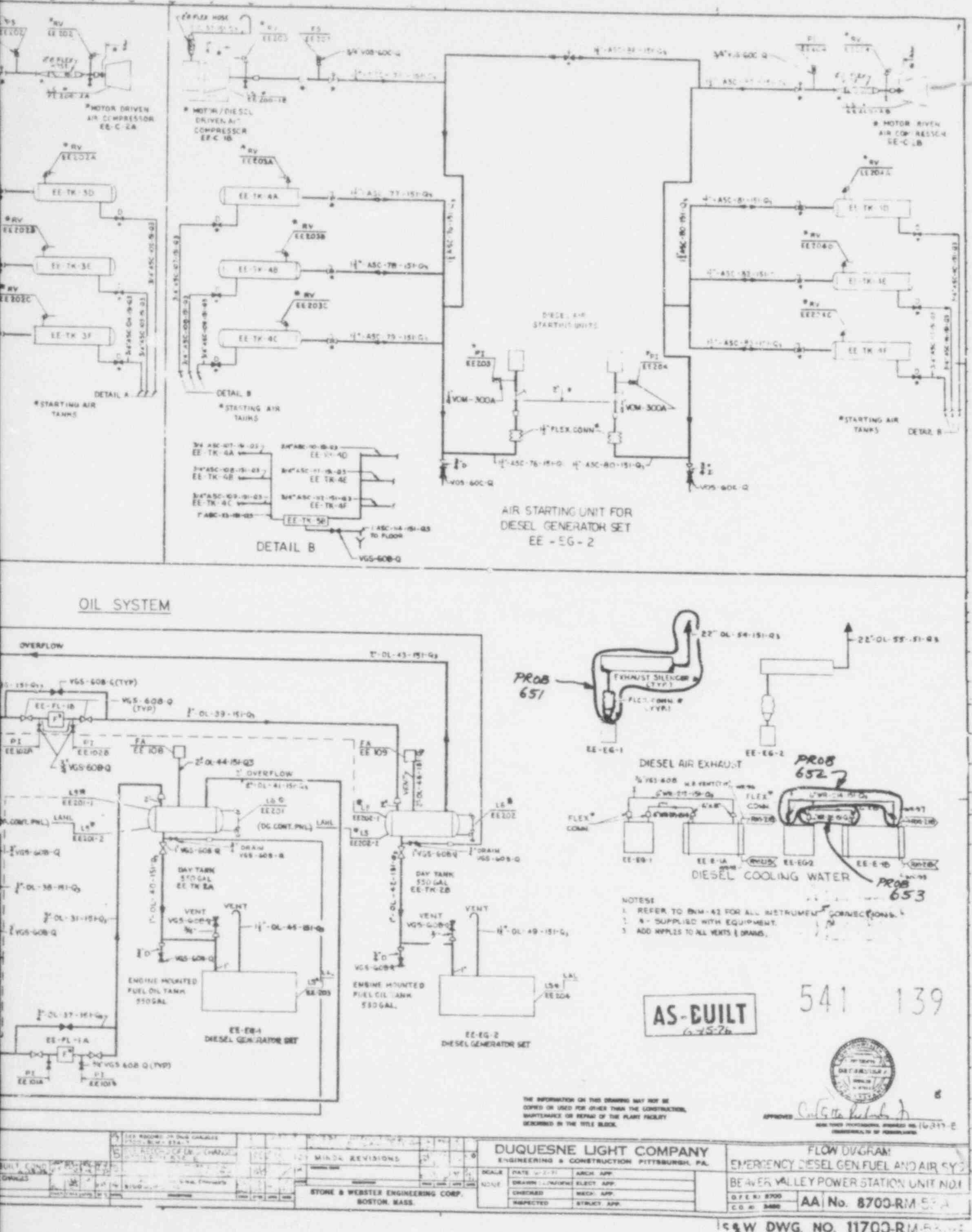
FIG. 9.3-1
(RM-38A)
RESIDUAL HEAT REMOVAL SYSTEM
BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

I

POOR ORIGINAL



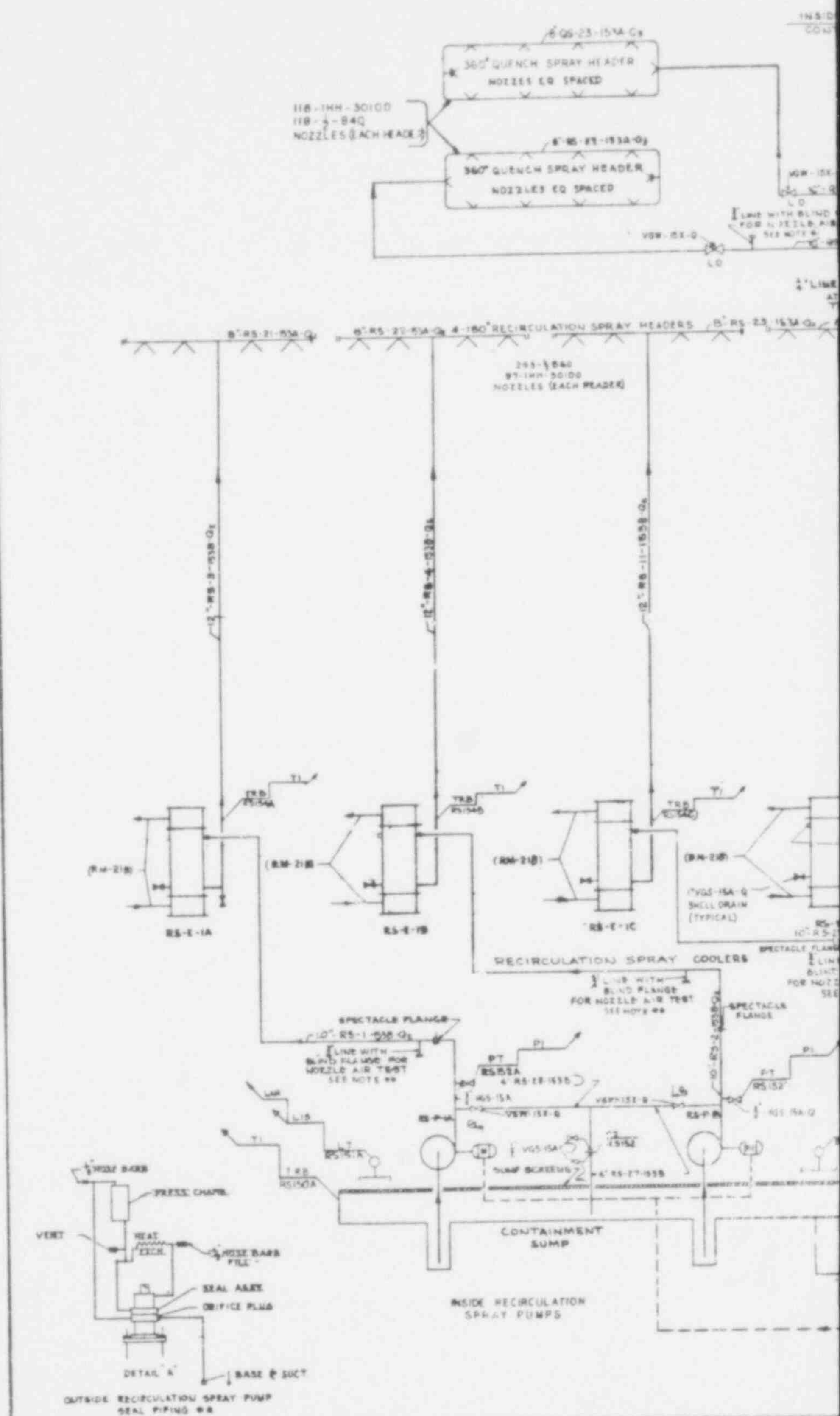
541


*MOTOR DRIVEN AIR COMPRESSOR EE-C-2A
*MOTOR/DIESEL DRIVEN AIR COMPRESSOR EE-C-1B
*MOTOR DRIVEN AIR COMPRESSOR EE-C-2B
EE-TK-3D
EE-TK-3E
EE-TK-3F
EE-TK-4A
EE-TK-4B
EE-TK-4C
DETAIL A
*STARTING AIR TANKS
DETAIL B
*STARTING AIR TANKS
DIESEL AIR STARTING UNITS
VOM-300A
1½" FLEX CONN*
*STARTING AIR TANKS
DETAIL B
AIR STARTING UNIT FOR DIESEL GENERATOR SET EE-EG-2
DETAIL B
OIL SYSTEM
OVERFLOW
EE-FL-1B
DAY TANK 550 GAL EE-TK-2A
DAY TANK 550 GAL EE-TK-2B
ENGINE MOUNTED FUEL OIL TANK 550 GAL
EE-EG-1 DIESEL GENERATOR SET
ENGINE MOUNTED FUEL OIL TANK 550 GAL
EE-EG-2 DIESEL GENERATOR SET
EE-FL-1A
PROB 651
EXHAUST SILENCER (TYP)
EE-EG-1
EE-EG-2
DIESEL AIR EXHAUST
PROB 652
DIESEL COOLING WATER
PROB 653
NOTES:
1. REFER TO BVM-43 FOR ALL INSTRUMENT CONNECTIONS.
2. *- SUPPLIED WITH EQUIPMENT.
3. ADD NIPPLES TO ALL VENTS & DRAINS.
AS-BUILT
6-25-76
541 139
THE INFORMATION ON THIS DRAWING MAY NOT BE COPIED OR USED FOR OTHER THAN THE CONSTRUCTION, MAINTENANCE OR REPAIR OF THE PLANT FACILITY DESCRIBED IN THE TITLE BLOCK.
APPROVED
DUQUESNE LIGHT COMPANY
ENGINEERING & CONSTRUCTION PITTSBURGH, PA.
STONE & WEBSTER ENGINEERING CORP. BOSTON, MASS.
FLOW DIAGRAM
EMERGENCY DIESEL GEN. FUEL AND AIR SYS
BEAVER VALLEY POWER STATION UNIT NO.1
AA No. 8700-RM-53A
S&W DWG. NO. 11700-RM-53

LIST OF FSAR FIGURES

541 140

POOR ORIGINAL



RM No LEGEND

RM No	FIG No	RM No	FIG No	RM No	FIG No	RM No	FIG No	RM No	FIG No	RM No	FIG No	RM No	FIG No	RM No	FIG No
14A	10.3-1	21A	10.3-3	22C	9.4-3	29B	9.2-2	30B	11.2-4	32C	9.6-3	34C	9.7-3	37B	4-2
16A	10.3-2	21B	9.9-1	22D	9.4-4	29C	9.2-3	31A	9.5-1	33A	11.2-6	35A	6.4-1	38A	9.3-1
17A	10.3-4	22A	9.4-1	25A	9.8-1	29D	9.2-4	32A	9.6-1	34A	9.7-1	36A	5.4-3	39A	9.1-1
18A	10.3-5	22B	9.4-2	29A	9.2-1	30A	11.2-5	32B	9.6-2	34B	9.7-2	37A	6-1	39B	9.1-2

541 141

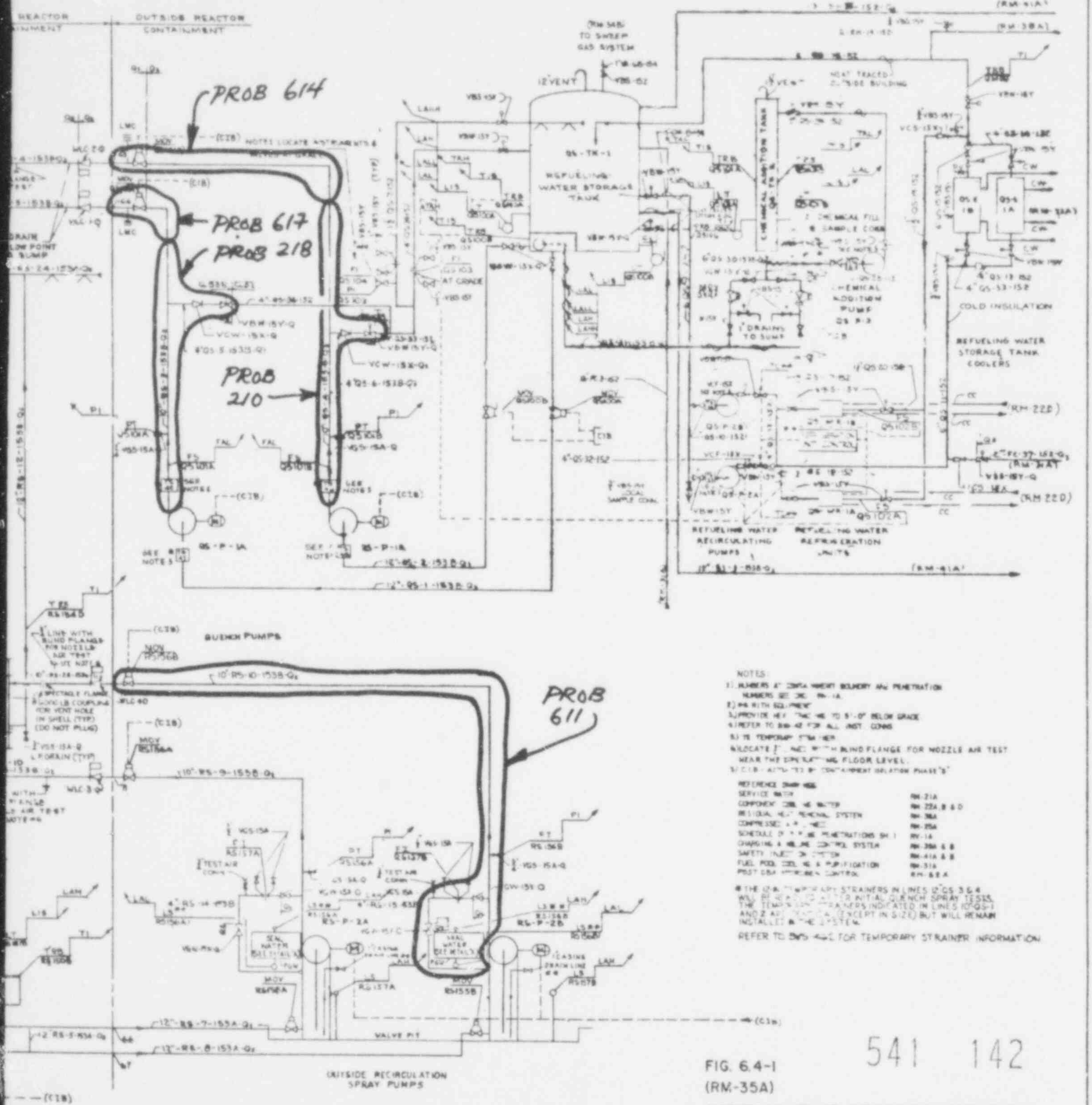


FIG. 6.4-1
(RM-35A)

CONTAINMENT DEPRESSURIZATION SYSTEM

BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

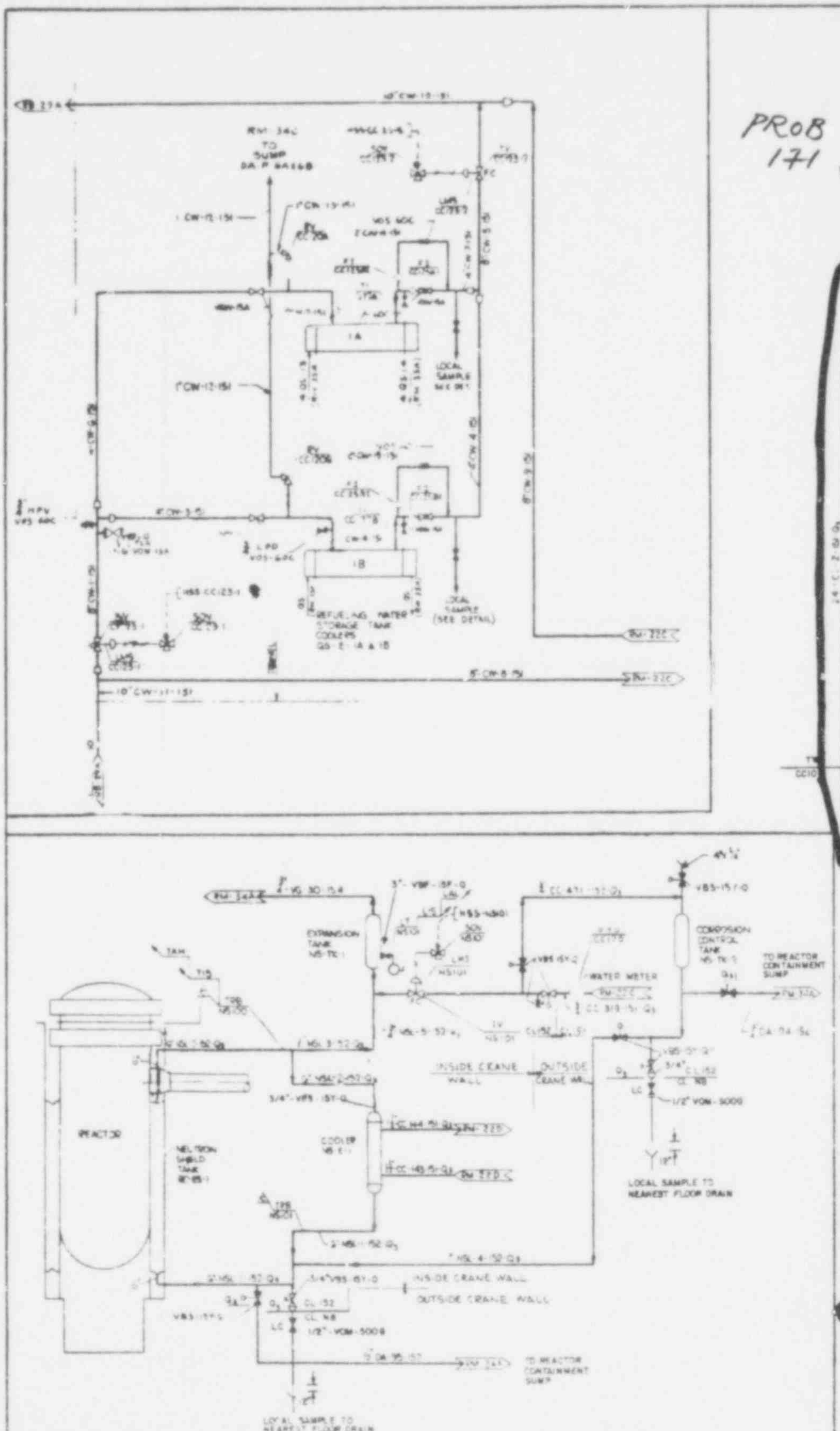
RM. No.	FIG. No.	RM. No.	FIG. No.
41 A	6.3-1	50 A	10.3-6
41 B	6.3-2	51 A	9.11-1
43 A	11.2-3		11.2-7
43 B	11.2-1	52 A	4.5-1

541 142

13

REVISION 1

PROB 171



NEUTRON SHIELD TANK COOLING WATER SUB-SYSTEM

RM. No. LEGEND

541 143

AMENDMENT 6
12/7/73

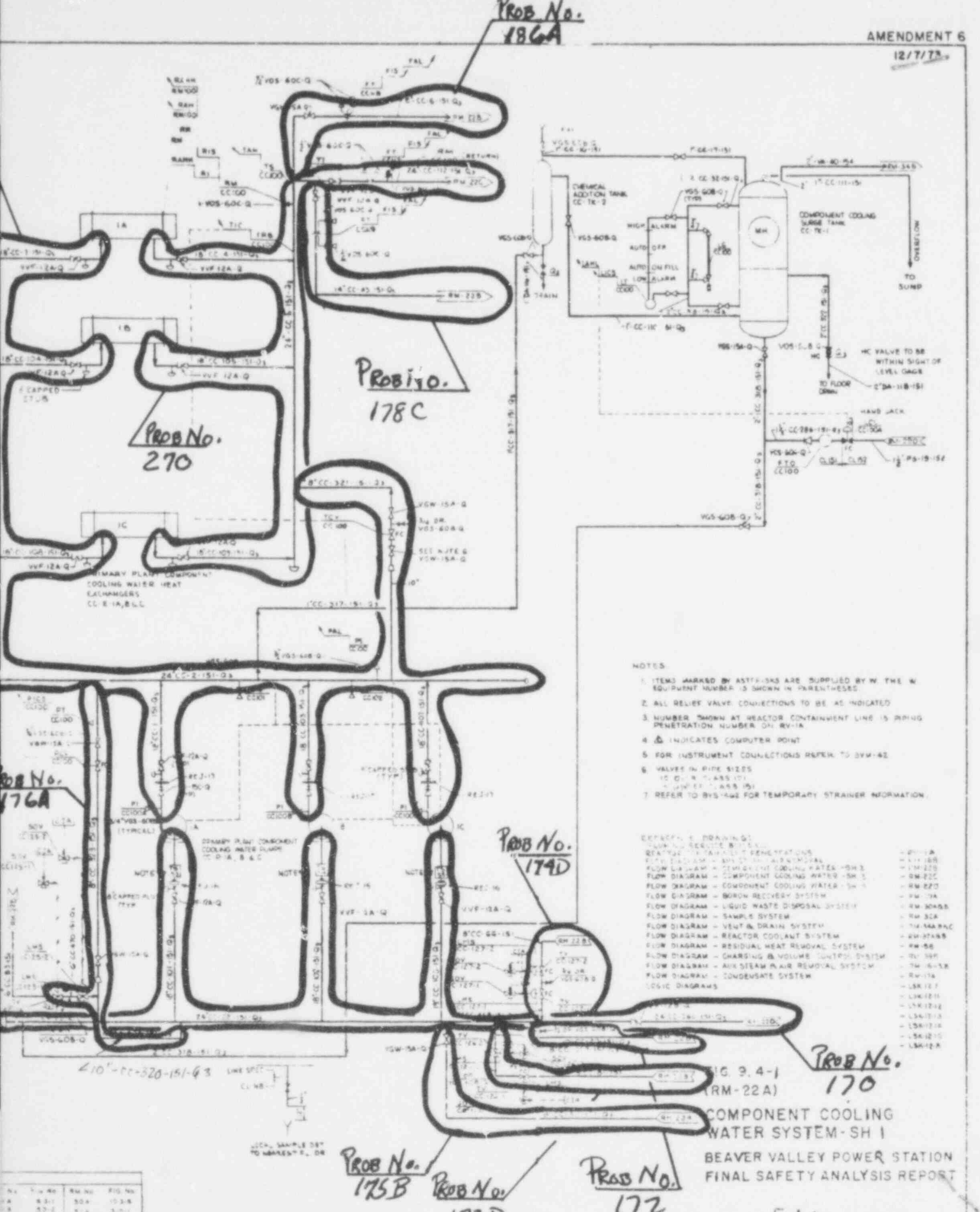
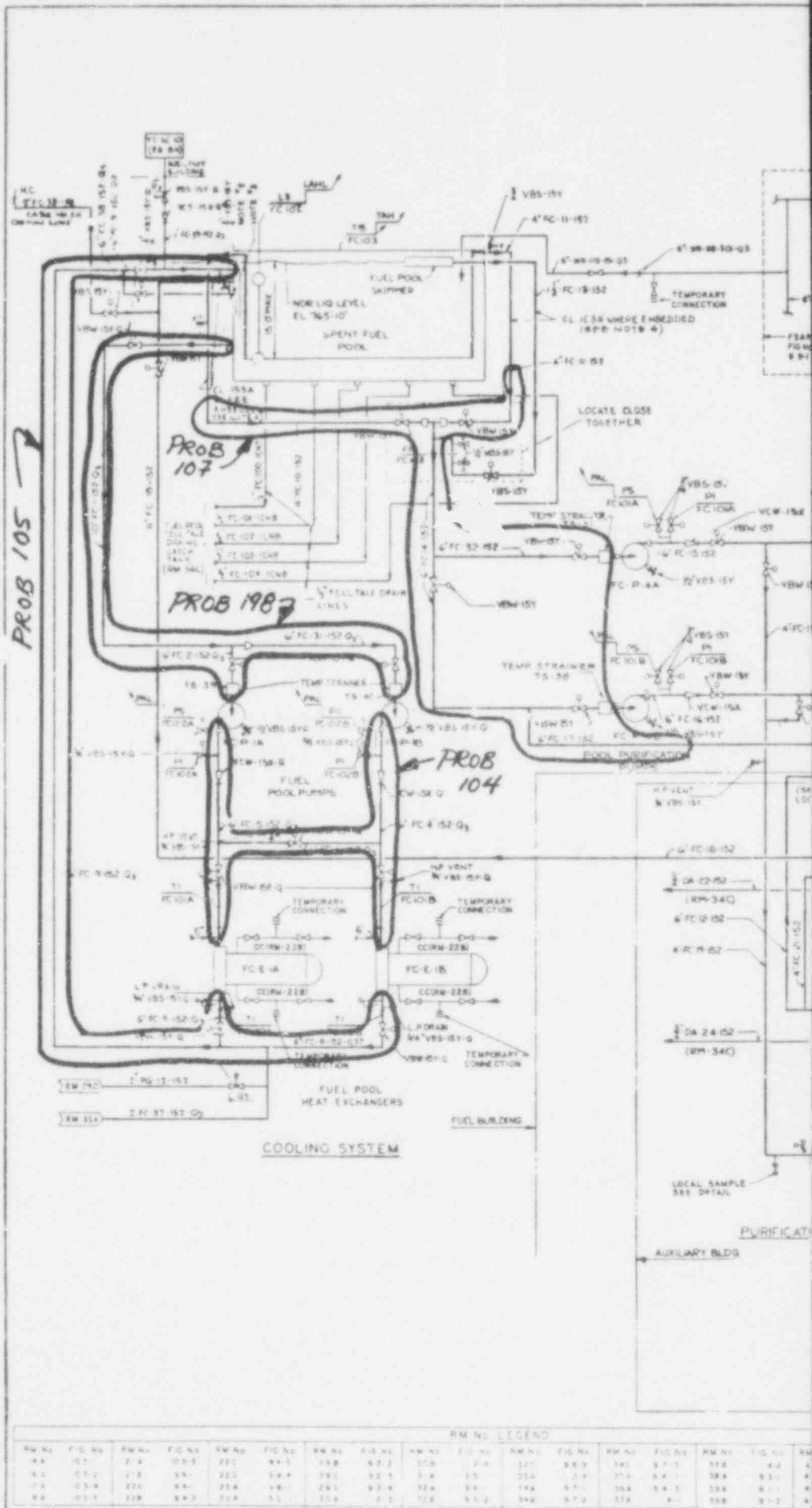


FIG. 9.4-1
(RM-22A)
COMPONENT COOLING WATER SYSTEM-SH 1

BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

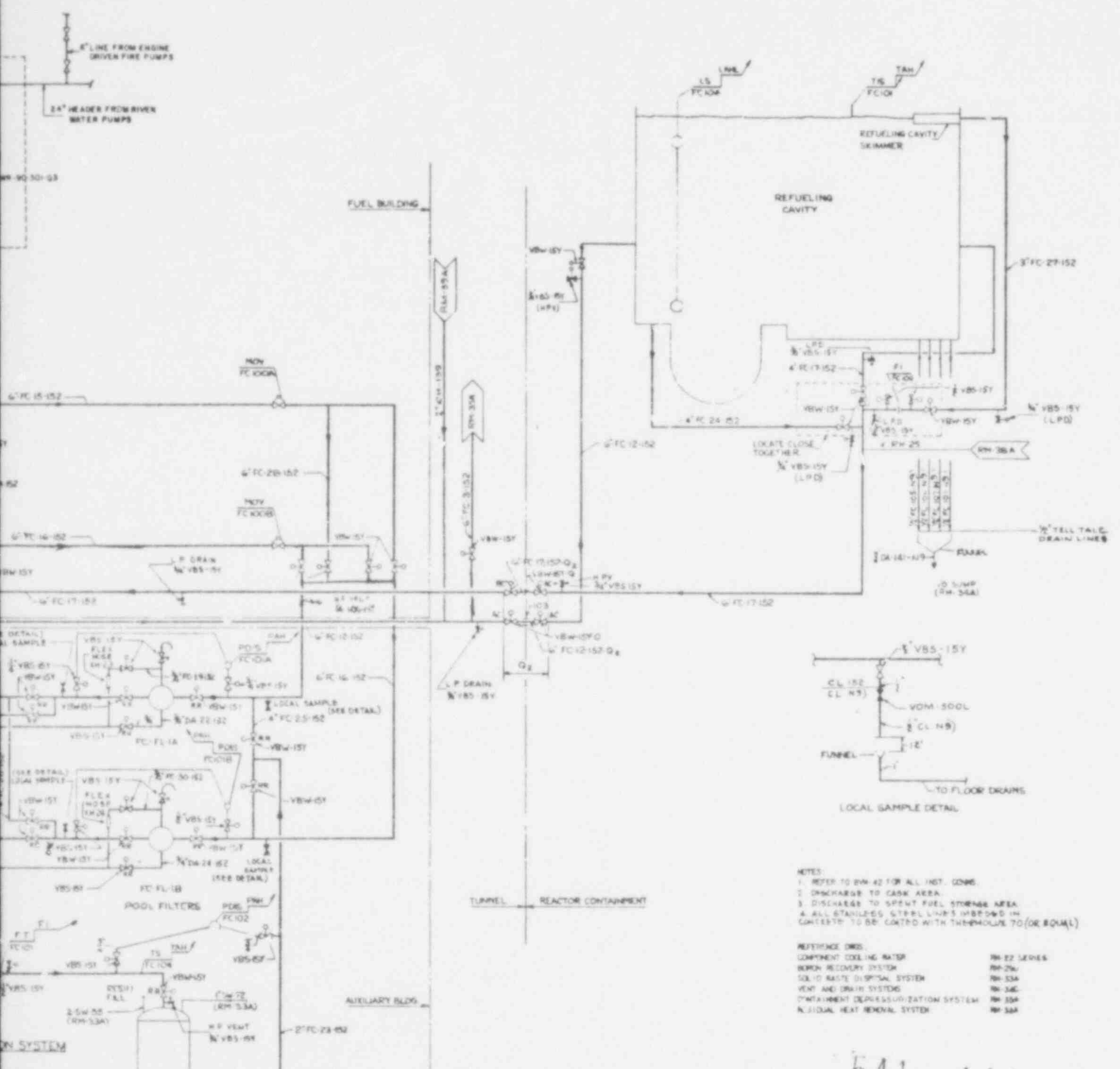
541 144 3

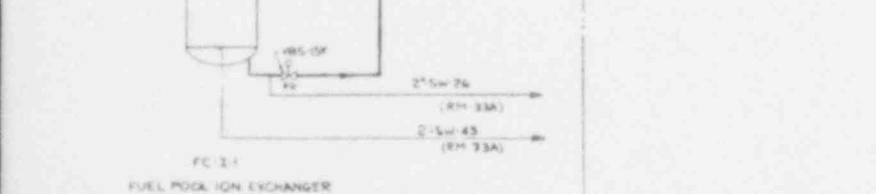
REVISION 1



FUEL POOL SKIMMER
NOR LIQ LEVEL EL 765-10'
SPENT FUEL POOL
LOCATE CLOSE TOGETHER
TEMPORARY CONNECTION
PROB 107
PROB 105
PROB 198
PROB 104
TEMP STRAINER
FUEL POOL PUMPS
POOL PURIFICATION PUMPS
FUEL POOL HEAT EXCHANGERS
FC-E-1A
FC-E-1B
FUEL BUILDING
COOLING SYSTEM
LOCAL SAMPLE SEE DETAIL
AUXILIARY BLDG
RM No LEGEND

541 145





NOTES:
1. REFER TO RM-42 FOR ALL INST. CONNS.
2. DISCHARGE TO CASK AREA.
3. DISCHARGE TO SPENT FUEL STORAGE AREA.
4. ALL STAINLESS STEEL LINES IMBEDDED IN CONCRETE TO BE COATED WITH THERMOLUX 70 (OR EQUAL)

REFERENCE DWGS.

COMPONENT COOLING WATER	RM-22 SERIES
BORON RECOVERY SYSTEM	RM-26
SOLID WASTE DISPOSAL SYSTEM	RM-33A
VENT AND DRAIN SYSTEMS	RM-34C
CONTAINMENT DEPRESSURIZATION SYSTEM	RM-35A
RESIDUAL HEAT REMOVAL SYSTEM	RM-36A

No	FIG No	RM No	FIG No
7	9.5-1	50A	10.3-6
8	9.5-2	51A	8.15-1
A	11.2-3		11.2-7
B	11.5-1	57A	9.5-1

541 146

FIG 9.5-1
(RM-31A)

FUEL POOL COOLING AND PURIFICATION SYSTEM

BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

β

plant are subject to regulatory body approval, so this combination of requirements governed seismic design of B31.1 piping on nuclear plants.

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

The qualification of large piping systems of safety class categories is nearly always done by means of a computer analysis. A dynamic analytical model of 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 represented by springs connecting the masses. System damping is included as 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 appropriate seismic spectral motion on the computer.

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

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

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

When load combinations are complete, bending moments and stresses in the piping system are computed according to B31.1 equations. Basically, twice the 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 comparable to ASME III today. S_h is the tabulated value of allowable stress as provided by the Code, in the hot condition. In B31.1, S_h is based on the 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 temperatures up to 90 percent of yield strength because of the greater toughness and ductility of these materials. These values of allowable stress are the lowest in use for any piping in the United States. ASME III Class 1 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 respectively permit allowable stresses up to 72 percent of the ultimate

deflection. Extrapolating the curve of Figure H-3 to 0.5 inch deflection yields 10 percent damping.

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

8. CONCLUSIONS AND IMPLICATIONS FOR MODERN NUCLEAR PLANTS

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

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

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

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

9. REFERENCES

1. Cloud, R.L. et al. Editors, Pressure Vessels and Piping, Design and Analysis, Amer. Soc. of Mech. Engrs. N.Y., N.Y., 1972.

541 148

2. Biot, M.A., Analytical and Experimental Methods in Engineering Seismology, Trans ASCE 108 p 365-408, 1942.

3. Biggs, J.M., Introduction to Structural Dynamics, McGraw Hill Book Co., 1964.

4. Berkowitz, L., Seismic Analysis of Primary Piping Systems for Nuclear Generating Systems, Reactor and Fuel Processing Technology, Argonne Natl Lab, Fall, 1969.

5. Markl, A.R.C., Fatigue Tests of Piping Components, Trans ASME V 74 1952.

6. Brock, J.E., Expansion and Flexibility, Chap. 4, Piping Handbook, 5th Ed., King and Crocker (Eds.) McGraw Hill Publishing Co., 1967.

7. Markl, A.C.R., Piping Flexibility Analysis, Trans ASME, 1955, p 419.

8. Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis, Amer. Soc. of Mech. Engrs., 1979 (to be published).

9. Swiger, W.F., personal communication, May 1979.

10. Swiger, W.F., Notes on Plants Designed by Stone & Webster Which Have Experienced Large Earthquakes, 1979, Unpublished.

11. How Nuclear Piping Code Rules Will Influence Piping Design Today and Tomorrow, Heating, Piping, and Air Conditioning, June, 1970, p 69.

12. The Great Alaska Earthquake of 1964, Engineering, National Academy of Sciences, Washington, D.C., 1973.

13. San Fernando, California, Earthquake of February 9, 1971, Leonard Murphy, Sci. Coord, U.S. Dept. of Comm., NOAA, Washington, D.C., 1973.

14. Snyder, Arthur I., Damage to Mechanical Equipment as a Result of the Feb. 9, 1971 Earthquake in San Fernando, California, Seismic Design and Analysis, Amer. Soc. of Mech. Engrs., 1971.

15. Managua, Nicaragua Earthquake of Dec. 23, 1972, Earthquake Engineering Research Inst., Nov., 1973.

16. Bohm, George J. Damping for Dynamic Analysis of Reactor Coolant Loop Systems, Optical Meeting on Reactor Safety, Salt Lake City, Utah, March 1973, Conf-730304 Available NTIS.

17. Bush, Spencer, Battelle Northwest Laboratories, Personal Communication.