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SUBJECT: Forwards "Design Reverification Program," Vols 1 & 2, final assessment rept.Results of program will be presented to NRC in late Oct 1983.

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SUBJECT: Forwards "Design Verification Program" [illegible] Final assessment rept, results of program will be presented to [illegible] on date of [illegible].

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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

Docket No. 50-397

September 27, 1983

Mr. Harrold R. Denton, Director
Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: WNP-2 Design Reverification Program

References:
a) Letter, G.D. Bouchey to H.R. Denton, "Nuclear Project No. 2 - Verification of Design and Construction Adequacy," dated October 22, 1982.

b) Letter, R. L. Ferguson to W.J. Dircks, "WNP-2 Plant Verification Program for WNP-2," dated November 24, 1982.

c) Letter, H.R. Denton to R.L. Ferguson, "Design Verification Program for WNP-2," dated December 28, 1982.

d) Letter, G.D. Bouchey to A. Schwencer, "Nuclear Project No. 2 - Qualification of Engineers Assigned to the WNP-2 Reverification Reviews," dated January 13, 1983.

References (a) and (b) described the Supply System programs for assuring that WNP-2 is designed and constructed in accordance with our commitments. One element of that overall program was an in-depth design reverification review of three reactor systems to provide added assurance of WNP-2 design adequacy. Reference (c) indicated your acceptance of the program proposed by the Supply System and requested additional information regarding the qualifications and independence of the engineers assigned to perform the design reviews. Reference (d) supplied the requested resumes and independence certifications.

Enclosed are copies of the final assessment report which provides the results of the WNP-2 Design Reverification Program. A meeting is being scheduled with NRC staff in late October, 1983, to present the results of the program.

If questions arise regarding the WNP-2 Design Reverification Program, you may contact Dr. G. D. Bouchey, (509)372-5359.

G C Sorensen

G. C. Sorensen, Acting Manager
Nuclear Safety and Regulatory Programs

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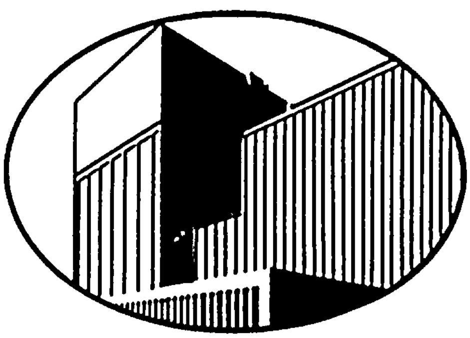
Volume II: Appendices to Final Assessment Report



September 1983

Washington Public Power Supply System
Richland, Washington 99352

830992

APPENDIX 1

WNP-2 Requirements and Design Reverification

Final Assessment Report

List of Potential Finding Reports

APPENDIX 1

LIST OF POTENTIAL FINDING REPORTS

PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-1		X		3.1	The B&R criteria document does not include requirements for all design input areas identified on the requirements reverification checklist.
HPCS-2		X		3.2.3.1.D	The equipment piece number for diesel engine cooling water heat exchanger is not consistent on all drawings.
HPCS-3		X		3.2.3.6.A	The diesel air start system is not totally redundant as described on the Flow Diagram.
HPCS-4		X		3.2.3.10.A	Current calculation revisions were not used as the basis for subsequent calculations.
HPCS-5		X		3.2.3.10.B	B&R and alternate calculations do not agree on the diesel exhaust pressure drop.
HPCS-6			X	3.1	Cold working of instrument tubing.
HPCS-7		X		3.2.3.2	Detail B showing HPCS Instrumentation is missing from Flow Diagram.
HPCS-8			X	3.2.3.5.B	HPCS/RCIC condensate storage level instrumentation separation is questioned.
HPCS-9		X		3.1	FSAR does not state the correct ASME Code Classification for the HPCS diesel cooling water heat exchanger.
HPCS-10			X	3.1	FSAR states that all fuel oil piping is ASME III whereas some is B31.1.
HPCS-11			X	3.2.3.6.B	Calculations that justify condensate storage level transfer setpoint not found.
HPCS-12			X	3.1	FSAR does not state the piping material requirements specified in the ECD.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

LIST OF POTENTIAL FINDING REPORTS

PFR No.	Classification** F	Classification** O	Classification** NV	Review Area*	Description
HPCS-13			X	3.1	Different sections of the engineering criteria document do not agree on piping corrosion allowance.
HPCS-14		X		3.1	FSAR does not agree with ASME piping code effective date specified in the ECD.
HPCS-15	X			3.2.3.6.A	B&R calculation on emergency water volume for HPCS pump suction is inconsistent with other calculations and the design events.
HPCS-16			X	3.2.3.7.A	HPCS relief valve design does not incorporate GE design specifications for double flange gaskets.
HPCS-17		X		3.2.4.3	The DSA diesel engine exhaust system line size does not correspond to manufacturers recommendations.
HPCS-18		X		3.2.4.3	The diesel fuel oil system does not meet NFPA Std. 37 requirements.
HPCS-19	X			3.2.4.3	No air box drain collection tank is provided for the HPCS diesel.
HPCS-20		X		3.1	Design requirement documents and FSAR values for vital piping damping coefficient do not agree.
HPCS-21	X			3.2.6.4.B	There is clearance between the attached parts of two snubbers where gaps are not allowed.
HPCS-22		X		3.2.5.3	No design calculations traceable to the HPCS pump support anchor bolts were found.
HPCS-23			X	3.2.5.1.D	Design procedures covering aspects of tne Instrumentation Installation Contractor design process were considered inadequate.
HPCS-24		X		3.2.5.1.D	Improper stress intensification factors were used in the analysis of PI Line X-73a.
HPCS-25		X		3.2.5.1.D	Evaluation of local stresses caused by weld attachments for PI Line X-73a was considered to be inadequate.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-26		X		3.2.5.1.D	No faulted conditions stress evaluation was found for PI Line X-73a.
HPCS-27		X		3.2.5.1.A	Loads used in the design of pipe supports for M200-2 piping system are not current.
HPCS-28		X		3.2.6.4.A	Potential restraint to thermal expansion of PI Line X-73a was identified.
HPCS-29		X		3.2.4.5	HPCS-FE-7 is installed with pressure taps and attached instruments located at the top rather than horizontally as suggested by good engineering practice.
HPCS-30		X		3.2.3.4	There are ambiguities in the piping code specifications for the CST to HPCS pump suction piping.
HPCS-31		X		3.2.3.5.B	Discrepancies in separation criteria were noted in the BRI documentation.
HPCS-32		X		3.2.4.6	GE specifications for instrument setpoint, accuracy, drift and range are not consistent.
HPCS-33			X	3.2.4.4	Instrument tubing match line elevations disagree between two isometric drawings.
HPCS-34			X	3.2.4.5	There is a discrepancy between the flange bore and pipe ID for HPCS-FE-7.
HPCS-35		X		3.2.4.6	The nameplate and ranges specified in the instrument data sheet for DPIS-9 do not agree.
HPCS-36			X	3.2.6.2	There is a discrepancy between GE and BRI recommendations upstream and downstream straight pipe run for orifice flowmeters.
HPCS-37		X		3.2.6.2.C	Discrepancies between the GE and BRI requirements for impulse line slope and instrument elevation are noted.
HPCS-38		X		3.2.6.2.B	HPCS-LS-2A was tagged with a tag identifying the level switch as HPCS-LS-2B.
HPCS-39		X		3.2.3.7.B	The instrument line for the suppression pool level switch is not orificed to provide containment isolation per RG 1.11.

** F - Finding
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NV - Not Valid

*Corresponds to Report Section Number

LIST OF POTENTIAL FINDING REPORTS

PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-40			X	3.2.6.2	The specified and nameplate ranges of HPCS-PS-12 do not agree.
HPCS-41		X		3.2.3.2	The valve interlock control function for HPCS-LS-2A is correctly shown in the GE specifications and FCD but not shown on tne GE P&ID or BRI Flow Diagram.
HPCS-42			X	3.2.3.4	The seismic classification of HPCS suction piping from the CST is incorrect.
HPCS-43		X		3.2.4.3	There are discrepancies in the BRI calculations which sized restrictive orifice HPCS-RO-4.
HPCS-44		X		3.2.3.1.C	The calculated pressure drop for HPCS diesel starting air system exceeds manufacturers recommendations.
HPCS-45			X	3.2.3.4	There are ambiguities in FSAR Table 3.2-1 on code class groups for the HPCS system.
HPCS-46	X			3.2.4.7	The adjustable range for Breaker 4-41 short circuit tripping does not meet the GE specification.
HPCS-47		X		3.2.4.7	The relay element connected to Breaker 4-41 does not permit proper coordination.
HPCS-48		X		3.2.3.3	As-built data was not used in BRI voltage drop calculation 2.06.03 for TR4-41.
HPCS-49	X			3.2.4.7	Ground fault alarm relays on Bus SM-4 will not function reliably.
HPCS-50		X		3.2.3.3	The effect of simultaneous starting of 480V and 4KV motors was not considered in BRI Voltage Drop Calculation 2.06.03, Rev. 5.
HPCS-51		X		3.2.3.6.C	The present design does not include the required degraded voltage protection and auto return to standby.
HPCS-52		X		3.2.3.3	The vendor print file for TR 4-41 contains two contradicting drawings.
HPCS-53		X		3.2.3.3	No fault duty calculation was provided for MC-4A.

** [illegible] Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-54		X		3.2.6.3	One of the bolts is missing from the HPCS pump grounding lug connection.
HPCS-55		X		3.2.3.7.A	There is an equipment piece number discrepancy between FSAR Table 6.2-16 and B&R Drawing M520 for several valves.
HPCS-56		X		3.2.5.2.B	Local pipe stress from a welded attachment lug for pipe support 910-N was not calculated adequately.
HPCS-57		X		3.2.5.2.A	Miscellaneous errors exist in the design calculation for pipe support HPCS-66.
HPCS-58	X			3.2.5.1.B	There is an error in the piping design guide.
HPCS-59		X		3.2.5.1.B	Calculation 8.14.64A does not correctly calculate the functional capability stress of the piping system.
HPCS-60			X	3.2.5.1.B	The pipe crack evaluation appears to be incomplete for BRI Calculation 8.14.64A.
HPCS-61			X	3.2.5.1.B	The displacement summaries for branch pipe connections do not include rotations.
HPCS-62		X		3.2.5.1.B	The load data source for chugging, SRV and LOCA jet direct loads are not referenced in BRI calculation 8.14.64A.
HPCS-63		X		3.2.5.1.B	There are documentation problems with seismic analysis input calculations for BRI Calculation 8.14.64A.
HPCS-64		X		3.2.5.1.B	Improper revisions were made to support load tables in BRI Calculation 8.14.64A.
HPCS-65		X		3.2.5.1.B	The thermal displacements at branch connections were not correctly summarized.
HPCS-66	X			3.2.5.1.B	Support design loads were incorrectly reported for HPCS-910N in BRI Calculation 8.14.64A.

** F - Finding
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*Corresponds to Report Section Number

PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-67		X		3.2.5.1.B	Errors were found in revised thermal expansion computer runs in BRI Calculation 8.14.64A.
HPCS-68		X		3.2.5.1.B	Some stress intensification factors were not included in the stress analysis in BRI Calculation 8.14.64A.
HPCS-69		X		3.2.5.1.B	An incorrect mass was used in the computer model of valve HPCS-V-15.
HPCS-70			X	3.2.5.1.B	The physical properties of HPCS-V-15 used in the computer model did not come from the referenced drawings.
HPCS-71			X	3.2.5.1.B	Various errors were made in the thermal expansion analysis in BRI Calculation 8.14.64A.
HPCS-72			X	3.2.5.1.B	Emergency condition temperatures were not considered in the thermal expansion analysis in BRI Calculation 8.14.64A.
HPCS-73				N.A.	This number was not used.
HPCS-74		X		3.2.5.1.A	Valve nozzle end loads and accelerations are not evaluated per requirements of the ECD.
HPCS-75				N.A.	This number was not used.
HPCS-76				N.A.	This number was not used.
HPCS-77			X	3.2.5.1.A	The SSE response spectra for mass point 40 (BRI Calculation 8.14.82) is not included in referenced document.
HPCS-78			X	3.2.5.1.A	The stress index, C2, used for the 3/4" elbowlet is lower than that required by ASME Section III.
HPCS-79		X		3.2.5.1.A	An additional weight of 1047 pounds was added to the 12" HPCS-V-5.
HPCS-80			X	3.2.5.1.A	HPCS-V-76 was modeled using a weight 400 pounds less than the drawings indicate.

** F - Finding
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PFR No.	Classification** F	O	NV	Review Area*	Description
HPCS-81	X			3.2.5.1.A	Incorrect scales were used for ADLPIPE response spectra input.
HPCS-82		X		3.2.5.2.D	Thermal loads used for design of HPCS-52 do not match those in the applicable pipe calculation.
HPCS-83		X		3.2.5.1.C	Elbow dimensions used in the analysis of small bore line DE-1738-1 are in error.
RFW-1		X		3.4.6.3	RFW-TE-41A had been improperly terminated in the field.
RFW-2		X		3.4.4.3.B	RFW line "A" temperature element installed orientation does not correspond to orientation shown on pipe isometric.
RFW-3		X		3.4.6.3	The signal cable for RFW-TE-41A was incorrectly labeled.
RFW-4		X		3.4.4.3.D	The wrong type of flow element was selected for RFW-FE-15.
RFW-5			X	3.4.4.1	RFW-V-32A was not specified to be testable with low pressure air as required by 10CFR50 Appendix J.
RFW-6	X			3.4.4.1.B	The feedwater heater relief valve capacity is not sufficient to provide relief for all hypothetical events.
RFW-7		X		3.4.4.2.A	Motor operator for RFW-V-65 is supplied with Class 1E power per PED 218-E-2858 but Drawing E-528, Sheet 27 has not been updated.
RFW-8		X		3.4.6.3	The air operator extension shaft of RFW-V-32A interferes with RWCU inlet line to header "A".
RFW-9		X		3.4.4.3.C	Inconsistencies are noted on the elementary and other electrical drawings for RFW-V-32A.
RFW-10		X		3.4.4.3.D	Upstream straight piping section length for RFW-FE-1A is insconsistent with ECD requirements.
RFW-11	X			3.4.4.3.D	Downstream straight piping length requirements for RFW-FE-1A is inconsistent with the ECD.

** F - Finding
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LIST OF POTENTIAL FINDING REPORTS

PFR No.	Classification** F	O	NV	Review Area*	Description
RFW-12			X	3.4.4.3	Connecting pipe size and pressure loss documentation inconsistencies are noted for RFW-FE-1A.
RFW-13			X	3.4.4.3	RFW-FE-1A is not installed as shown on GE drawings.
RFW-14			X	3.4.4.3	System flushing and protection screening for RFW-FE-1A is not installed.
RFW-15			X	3.4.6.3	The RFW-FE-1A pressure tap configuration and connections are not installed per manufacturers recommendations.
RFW-16		X		3.4.4.3.D	RFW-FE-1A calibration curve anomolies.
RFW-17		X		3.4.6.3	RFW-DPT-803A signal loop wiring and instrument rack tubing runs are not labeled in accordance with contractor requirements.
RFW-18			X	3.4.3.4	Documentation inconsistencies were found in the review of RFW-V-32A containment isolation requirements.
RFW-19		X		3.4.3.4	Loss of signal lock-up interlocks for RFW-DT-1A, DT-1B and-FCV-10 have not been implemented in accordance with GE recommendations.
RFW-20			X	3.4.3.4	The BRI elementary diagram does not show the required interlock between V-112B and DPS-4.
RFW-21	X			3.4.4.1.B	Control valve cavitation problems exist with some valves.
RFW-22		X		3.4.4.1.B	There are inconsistencies and design input errors in the sizing calculation for RFW-FCV-15.
NL-1		X		3.4.5.3	Vendor approved nozzle loads did not include flange deadweights for RFW-P-1A and 1B.
RHR-1		X		3.3.4.3.C	All required cable types were not listed in Class 1E list.
RHR-2		X		3.3.3.1.C	The BRI wiring design for several RHR valves did not follow GE requirements.

** [illegible] Finding
O - Observation
NV - Not Valid

*Corresponds to Report Se[illegible]on Number

PFR No.	Classification** F	O	NV	Review Area*	Description
RHR-3		X		3.3.3.5	Containment isolation valve limit switches prematurely indicate valve closure.
RHR-4		X		3.1	FSAR incorrectly states that seismic reevaluation is supplemented by NUREG-0800.
RHR-5			X	3.1	No design requirement was found to match FSAR commitment for vertical cable tray run fire breaks.
RHR-6	X			3.3.3.4	RHR-FC -64B was not included in the remote shutdown system design as required by specification 22A3085.
RHR-7		X		3.3.3.4	Remote shutdown system design specification 22A3085, Para. 4.1.1 is not met in that a new common point was created.
RHR-8		X		3.3.3.3	BRI drawing E503-8, Rev. 23 shows RHR-P-3 in Division B instead of Division 2.
RHR-9		X		3.3.3.1.C	The GE documentation for RHR-V-3B throttling are contradictory.
RHR-10	X			3.3.3.1.D	The second level undervoltage relays will cause bypass of the 115 kV source and will lockout the shed ESF loads.
RHR-11		X		3.3.3.1.D	Feeder loads for MC-7BB and 7BA are missing from the MC-7B load calculation.
RHR-12		X		3.3.3.1.D	Feeder circuit breaker for MC-7BB may be set too low.
RHR-13		X		3.3.4.1.A	There is a discrepancy in the RHR-FCV-64 operating time specifications.
RHR-14		X		3.3.4.2.A	RHR-FIS-10B is overranged.
RHR-15			X	3.3.3.1.D	V-4B is missing from Drawing E528-36; V47B is missing from E528-37. Fuse and thermal overload sizes are not included on the E-528 drawing for RHR-V-4B and RHR-V-47B.
RHR-16		X		3.3.4.3.C	The voltage drop from E-SL-81 to MC-8BB is larger than the 3% recommended by BRI criteria.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

PFR No.	Classification**			Review Area*	Description
	F	O	NV		
RHR-17		X		3.3.4.2.A	RHR-FT-1 impulse lines are not routed as shown with the flow diagram.
RHR-18		X		3.3.4.2.B	The documentation (GE) for RHR-FI-5 does not agree with the installed instrument indicating scales.
RHR-19			X	3.3.4.B	RHR-MO-24B and 64B were ordered specifying the wrong environmental class.
RHR-20		X		3.3.4.1.C	A cavitation check was not included in BRI Calculation 5.17.13 for RHR-RO-1B.
RHR-21		X		3.3.4.1.C	A cavitation check was not included in BRI Calculation 5.17.26 for RHR-RO-3B.
RHR-22		X		3.3.4.3.C	Cable 2M8BA-20 is not sized for derated conditions.
RHR-23		X		3.3.5.2.B	Heat exchanger drawings do not match the calculations.
RHR-24	X			3.3.5.2.B	Heat exchanger installation does not reflect the calculation and installation specification requirements.
RHR-25	X			3.3.5.2.B	Due to increased loadings, the anchor bolt analysis is incomplete.
RHR-26		X		3.3.5.2.A	The original calculations were not updated or referenced to supporting calculations.
RHR-27			X	3.3.5.2.A	A buckling analysis was not performed as required by design criteria.
RHR-28		X		3.3.5.2.A	The anchor bolt analysis for the upper lateral supports is incomplete.
RHR-29		X		3.3.5.2.A	Assumed future (design) hanger loads must be verified against the actual hanger loads.
RHR-30		X		3.3.4.3.A	Motor starters and TR-8-81 are subjected to over voltages (SM-8 side of the 480 V system).
RHR-31			X	3.3.4.3.B	Documentation discrepancies for the fuse and overload heater sizes for three valves were noted.
RHR[illegible]				N.A.	To be inclu[illegible] in Pipe and Support Addendum.

** [illegible] Finding
O - Observation
NV - Not Valid

*Corresponds to Report Se[illegible]on Number

PFR No.	Classification** F	O	NV	Review Area*	Description
RHR-33	X			3.3.6	Lugs on the heat exchanger are not shimmed per the GE specifications.
RHR-34				N.A.	This number was not used.
RHR-35		X		3.3.4.3.A	Fuse/circuit breaker coordination information is missing.
EQ-1		X		3.5.5.2	HPCS-MO-4 is not listed in QID file identified on the Class 1E list.
EQ-2		X		3.5.5.2	The QID file referenced for HPCS-RO-4 did not contain the required design certification documentation.
EQ-3		X		3.5.5.1	QID file for HPCS-42-4A7C does not include required qualification data.
EQ-4			X	3.5.5.2	There is no in-situ pull/deflection operability test record for valve RHR-FCV-64B in the QID file.
EQ-5				N.A.	Number not used.
EQ-6				N.A.	Number not used.
EQ-7		X		3.5.5.6	Confirmation is required for existence of low pressure isolation alarm and procedure to isolate auxiliary steam system.
EQ-8				N.A.	Number not used.
EQ-9		X		3.5.5.2	The dynamic qualification levels identified in the QID file for HPCS-LS-2A are less than the required inputs.
EQ-10		X		3.5.5.6	Computer runs for the HVAC cooldown phase of HELB environments are not documented in the calculation file.
EQ-11	X			3.5.5.6	EQ environment calculation predicts peak pressures across RWCU heat exchanger room (R510) walls exceeding FSAR design values.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

LIST OF POTENTIAL FINDING REPORTS
(Page 12 of 13)

PFR No.	Classification** F	Classification** O	Classification** NV	Review Area*	Description
EQ-12			X	3.5.5.6	Subcompartment pressure analysis does not consider a door in Room R408.
EQ-13	X			3.5.5.6	Non-conservative isolation valve closure characteristics assumed in RCIC line break analysis.
EQ-14	X			3.5.5.6	A non-conservative time delay was assumed for generating RWCU break isolation signal.
EQ-15	X			3.5.5.6	HELB calculations for EQ profiles did not specifically address single failure criteria.
EQ-16	X			3.5.5.6	Normal HVAC ductwork may not retain its integrity to support post-HELB cooldown.
EQ-17		X		3.5.5.1	There are discrepancies between the model numbers on the Class 1E/SRM lists and the installed components.
FP-1		X		3.5.3.3	Several dedicated cables that require protection were not listed in the E-948 cable tray node summaries.
FP-2			X	3.5.3	Thermolag fire barrier is applied to an empty tray that is not required to be lagged.
FP-3			X	3.5.3	Cable spreading room pentration curbs shown on M-576 are not shown on S-906.
FP-4		X		3.5.3.2	Note 7 on M521 SH2 should not apply to RHR-V-40.
WL-1		X		3.5.6.2	Main steam tunnel north wall load combinations are not verified.
WL-2	X			3.5.6.2	FSAR criteria incorrectly applied to the main steam tunnel north wall deflection calculation.
WL-3		X		3.5.6.1	Attachment loads were not considered in BRI design calculation for the main steam tunnel north wall.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

PFR No.	Classification** F	O	NV	Review Area*	Description
WL-4		X		3.5.6.2	Main steam tunnel north wall minimum reinforcing steel inconsistent with FSAR. The minimum reinforcing steel ratios used in the main steam tunnel are not consistent with FSAR descriptions but do meet ACI 318-1971 requirements.
WL-5			X	3.5.6.2	Jet impingement load factors were not properly considered in calculating the dynamic loading of the main steam tunnel north wall.
PB-1	X			3.5.4.1.B	Material allowables used for approval of loads and/or stresses for PWS-2-1 are not traceable.
PB-2		X		3.5.4.1.C	Field walkdown of HPCS pipe break location identified more potential targets than those cited in the B&R calculation.
PB-3			X	3.5.4.1.D	Post-accident damage sequence differs from that postulated in the original B&R calculation.
PB-4			X	3.5.4.1.B	As-built strut size is smaller than the size specified in BRI calculation 8.01.52.
PB-5				N.A.	Number not used.
PB-6			X	3.5.4.2.B	Field walkdown of RWCU pipe break location identified more potential targets than cited in the BRI calculation.
PB-7			X	3.5.4.1.E	Process deficiencies in potential target resolution were noted.

** F - Finding
O - Observation
NV - Not Valid

*Corresponds to Report Section Number

SECTION A - REQUIREMENTS REVERIFICATION

A.1 Mechanical

A.1.1 Specifications

BRI Documents:

B & R Engineering Criteria Document, Rev. 11.

B & R Tech. Memos 443, Rev. A; 526, Rev. A; 308, 667, 1010, 148, 156, 653, 776, 785, 845.

General Electric Documents:

22A1843, HPCS System Design Specification, Revision 4.

22A1843AU, HPCS System Design Specification Data Sheets, Revision 4.

731E931, P&ID - HPCS System, Revision 7.

731E932AD, Process Diagram - HPCS System, Revision 3.

731E950AD, Flow Control Diagram - HPCS System, Revision 2.

GEK-71334, Hanford 2 Operation and Maintenance Instruction HPCS System, July 1978.

22A3095, Pressure Integrity of Piping Design Specification.

22A3095AD, Pressure Integrity of Piping Design Specification Data Sheet.

22A3790, System Design Pressures Design Specification.

22A3062, Mechanical Codes and Standards Design Specification.

22A2625, System Criteria and Applications for Protection Against Dynamic Effects of Pipe Break Design Specification.

22A2988, Separation Criteria, Revision 6.

22A7416, Separation Criteria, February 1981.

3316-031, Instruction Manual - HPCS Diesel Generator.

21A8657, Rev. 3, Valves.

21A8658, Rev. 1, Electric valve actuaters.

21A9347AF, Rev. 1, Instrumentation and Electric equipment.

22A2625, Rev. 1, Protection against pipe whip.

22A2702AB, Rev. 1, Seismic design.

22A2817, Rev. 3, Residual heat removal.

22A2817AY, Rev. 0, Data sheet for 22A2817.

22A3007, Rev. 1, Testability of instrumentation and controls.

22A3008, Rev. 5, Equipment environmental data.

22A3039, Rev. 1, Process instrumentation.

22A3062, Rev. 2, Mechanical codes and standards.

22A3095AD, Rev. 1, Data sheet for 22A3095.

22A3730, Rev. 0, RHR heat exchanger.

22A3730AB, Rev. 0, Data sheet for 22A03730.

22A3797, Floor response spectra.

22A5267, Rev. 1, Regulatory requirements.

22A7416, Rev. 1, Electrical separation.

21A8658, General Requirements MOV Actuation.

22A2703E, Radiation Sources.

22A2703F, Radiation Sources.

22A2707, Water Quality.

22A2708, Water Sampling.

22A2710, Standby AC Power.

22A2711, Plant DC Power.

22A2719AB, RFP Turbine Responses.

22A2719, FW Flow Measurement and Control.

22A2800, Rated Steam Output Curve.

22A2801, GE Reactor System Heat Balance Rated.

22A2802, GE Reactor System Heat Balance - 105% Rated.

22A2887, Nuclear Boiler System.

22A2907, Feedwater Control System.

22A3061, Rev. 0, Electrical Codes and Standards.

22A3790, Feedwater System Description.

22A3046, Rev. 1, Core Standby Cooling System Network.

A.1.2 Westinghouse Thermal Performance Data

AB095-1554, 1205849 KW, Maximum Calculated Not Guaranteed

AB095-1555, 115745 KW, Maximum Guaranteed

AF111-0330, No. 5 Extraction

AF111-0331, No. 6 Extraction

AE111-0572, Nos. 4 and 5 Extraction Zone Enthalpy

AE111-0573, No. 6 Extraction Zone Enthalpy

A.1.3 Codes and Standards

ASME Boiler and Pressure Vessel Code, 1971 Edition with Addenda through Winter 1973.

ANSI-B.31.1, Power Piping Code, 1973 Edition with Addenda through Winter 1973.

AISC Manual of Steel Construction, Seventh Edition, 1970.

WNP-2 FSAR with Amendments through 26, November 1982, Sections 1.2, 3.1, 3.2, 3.5, 3.11, 5.2, 6.1, 6.2, 6.3, 9.5, Appendix F, 14.2.

A.2 Instruments and Controls (Generic Design Requirements Applicable to HPCS, RHR and RFW Systems)

A.2.1 Specifications

BRI Documents:

BRI Design Criteria, Section G "Instrumentation and Control". Paragraphs 4.0, 4.4, 6.0, 7.4.2, Page G-45, Paragraph 2, Paragraph 7.4.1

General Electric Documents:

22A3039, Rev. 1, March 26, 1973, "Process Instrumentation". Sections: Paragraph 4.3.4.2.

22A3061, Rev. 0, September 3, 1971, "Electrical Codes and Standards".

22A3062, Rev. 0, March 10, 1971, "Mechanical Codes and Standards".

22A3095, Rev. 0, July 17, 1972, "Pressure Integrity of Piping and Equipment Pressure Parts". Sections: Paragraph #A3.3

22A3790, Rev. 0, May 31, 1973, "System Design Pressures".

22A3059, Rev. 1, November 6, 1972, "Definition of Piping Interfaces - Reactor Coolant Pressure Boundary".

22A2702A, Rev. 1, January 7, 1971, "Seismic Design" Design Specification.

21A8696, Rev. 0, May 10, 1971, "Seismic Requirement for Class I Instrumentation".

21A8658, Rev. 1, May 17, 1971, "General Requirements for Motor Operated Valve Actuators". Purchase Requisition.

22A3008, Rev. 5, April 8, 1977, "BWR Equipment Environmental Interface Data". Sections: Paragraph 3.1, 3.2, 4.1, 4.2, and 4.5.

22A3095 AD, Rev. 0, September 26, 1973, "Design Requirements for Pressure Integrity of Piping and Equipment Pressure Parts - Data Sheet".

22A2718, Rev. 5, April 10, 1974, "Special Wire and Cable".

22A3067, Rev. 2, October 12, 1972, "Mechanical Equipment Separation". Paragraph 4.5

22A7416, Rev. 0, "Electrical Equipment, Separation for Safeguards System". Specification February 19, 1982.

22A2988, Rev. 6, June 20, 1975, "Electrical Equipment; Separation for Safeguards Systems". Plant Requirements. Paragraphs: 4.3.3.1, 4.3.3.1.1, 4.3.3.1.2, SHT 10 Table IV, 4.4.1, 4.4.3, 4.4.3.4, 4.4.4, SHT 17, Table 3.

22A2625, Rev. 2, March 9, 1973, "Dynamic Effects/Pipe Break". Design Guide.

A.2.3 Contracts

Contract 42 Tech. Spec. Div. 15

Contract 215 Tech. Spec. Div. 50

Contract 220 Tech. Spec. Div. 50 Page 50A-16, Page 50A-34A, Page 50A-37, 38

A.3 RHR System - Design Requirements (I&C Section)

A.3.1 Specifications

BRI Documents:

Engineering Design Criteria, Section G

General Electric Documents:

22A2817, Rev. 3, November 27, 1973, "Residual Heat Removal System - System Design Specification", Section 4.3, 4.1.2, 4.1.2.4, 4.5.

22A2817AY, Rev. 0, October 31, 1974, "Residual Heat Removal System - System Design Specification - Data Sheet", Sections 2.1, 4.4, and 4.6.

22A3008, Rev. 5, April 8, 1977, "BWR Equipment Environmental Interface Data".

22A3041, Rev. 1, March 14, 1971, "Essential Components".

22A3185, Rev. 1, February 4, 1975, "Piping Interfaces".

22A2711, Rev. 3, January 9, 1974, "Plant D-C Power".

22A2718, Rev. 5, April 10, 1974, "Special Wire and Cable".

22A7416, Rev. 0, March 3, 1982, "Electrical Equipment, Separation for Safeguards System".

22A3007, Rev. 1, December 1, 1971, "Engineering Safeguards Systems, Criterion for Testability of Instrumentation and Controls".

22A3061, Rev. 0, September 3, 1971, "Electrical Codes and Standards".

22A3067, Rev. 2, October 12, 1972, "Mechanical Equipment Separation".

22A2710A, Rev. 7, September 9, 1974, "Standby A-C Power".

22A3095, Rev. 0, July 17, 1972, "Pressure Integrity of Piping and Equipment Pressure Parts".

22A3095AD, Rev. 0, September 26, 1973, "Design Requirements for Pressure Integrity of Piping and Equipment Pressure Parts - Data Sheet".

20A4756, Rev. 1, December 28, 1970, "Logic Symbols".

22A3059, Rev. 1, November 6, 1972, "Definition of Piping Interfaces Reactor Coolant Pressure Boundary".

22A2707, Rev. 5, May 28, 1974, "Water Quality".

22A2749, Rev. 1, June 24, 1975, "Cleaning of Piping and Equipment".

22A3790, Rev. 0, May 31, 1973, "System Design Pressures".

22A3039, Rev. 1, March 26, 1973, "Process Instrumentation".

MPL A62-4310, "Qualification Testing of Instrument and Control Devices Classified as Essential.

21A8696, Rev. 0, May 10, 1971, "Seismic Requirements for Class I Instrumentation". Sections SHT 2, 3.

22A3062, Rev. 2, March 10, 1971, "Mechanical Codes and Industrial Standards".

22A3746, Rev. 1, January 21, 1974, "System Design Specification - Local Instrument Panels".

22A2702A.

A.3.2 Contracts

Contract 42, Division 15, Sections 15A, B, and C

Contract 58, Division 50

Contract 59, Division 16, Section 16A

Contract 59, Division 50

Contract 215, Division 50

Contract 218, Division 50

Contract 220, Division 50

A.4 HPCS System - Design Requirements (I & C Section)

A.4.1 Specifications

BRI Documents:

Engineering Design Criteria, Section G, Paragraph 4.0, 4

General Electric Documents:

22A1483, Rev. 4, February 19, 1974, "High Pressure Core Spray System", Sections 3.1, 3.2, 3.3, 4.3.1, 4.3.1.2, 4.3.1.3, 4.3.1.5, 4.5.

731E932AD 11 P&ID, HPCS System", SHTS 1 and 2.

22A3039, Rev. 1, March 26, 1973, "Process Instrumentation" System Design Specification.

22A3061, Rev. 0, September 3, 1971, "Electrical Codes and Standards".

22A3062, Rev. 2, March 10, 1971, "Mechanical Codes and Standards".

22A3095, Rev. 0, July 17, 1972, "Pressure Integrity of Piping and Equipment Pressure Parts", Section 4, Table A.

22A3790, Rev. 0, May 31, 1973, "System Design Pressures".

22A3059, Rev. 1, June 24, 1975, "Cleaning of Piping and Equipment".

22A1483AU, Rev. 4, August 13, 1979, "High Pressure Core Spray System", Design Specification Data Sheet.

22A8696, Rev. 0, May 10, 1971, "Seismic Requirements for Class I Instrumentation", Sections: SHTS 2, 3.

A.4.2 <u>Contracts</u>:

Contract 42 Tech. Spec. Div. 15

Contract 215 Tech. Spec. Div. 50

Contract 220 Tech. Spec. Div. 50

A.5 RFW System - Specific Design Requirements (I&C Section)

A.5.1 Specifications

BRI Documents:

Engineering Design Criteria, Section G

General Electric Documents:

22A2907, Rev. 3, March 28, 1974, "Feedwater Control System (Steam Turbine Driven Reactor Feed Pumps)", System Design Specification, Sections 5.3, 4.3.2.2, 3.1.3.2, 3.3, 4.3.2.

22A2907AB, Rev. 1, August 16, 1971, "Feedwater Control System (Steam Turbine Driven Feed Pumps)" Design Specification, Section 4.1.3.

22A2719, Rev. 2, June 15, 1973, "Feedwater Flow Measurement and Control" Specification, Section 4.4.1.1.

22A2719AB, Rev. 0, July 26, 1971, "Feedwater Flow Measurement and Control" BWR Plant Requirements, Section 2.3.

22A3790, Rev. 0, May 31, 1973, "System Design Pressures".

22A2887, Rev. 6, January 29, 1979, "Nuclear Boiler System", Design Specification.

22A3095, Rev. 0, July 17, 1972, "Pressure Integrity of Piping and Equipment Pressure Parts", Sections: SHT 10, D2, SHT 95, SHT 90, 91; Table I, SHT 98 Comment #1.

238X241AD, Rev. 9, "Feedwater Control System - Master Parts List".

DL807E160TC, Rev. 0, June 15, 1978, "Device List and System Elementary Diagram--Feedwater Control System".

22A3041, Rev. 1, March 14, 1972, "Essential Components", Design Specification.

239X241AD, Rev. 9, "Feedwater Control System (Turbine)" Master Parts List.

PL368X482, Rev. 7, "Reactor Feedwater Document List".

22A3095AD, Rev. 0, September 26, 1973, "Design Requirements for Pressure Integrity of Piping and Equipment Pressure Parts - Data Sheet", Sections: SHT 20 A2.1, SHT 98 Paragraph C.

22A3059, Rev. 1, November 6, 1972, "Definition of Piping Interfaces - Reactor Coolant Pressure Boundary".

22A2707, Rev. 5, May 28, 1974, "Water Quality.

22A2887AB, Rev. 4, "Nuclear Boiler System--REVAB" System Design Specification.

22A86796, Rev. 1, March 7, 1978, "Seismic Requirements for Essential Instrumentation", Purchase Specification, Sections: SHT's 2, 3.

21A8657, Rev. 3, May 20, 1975, "General Requirements for Valves".

22A2988, Rev. 6, June 20, 1975, "Electrical Equipment, Separation for Safeguards Systems". Plant Requirements, Paragraphs: 4.3.3.1, 4.3.3.1.1, 4.3.3.1.2, SHT 10 Table IV, 4.4.1, 4.4.3, 4.4.3.4, 4.4.4, SHT 17 Table 3.

22A3067, Rev. 2, October 12, 1972, "Mechanical Equipment Separation", Paragraph 4.5.

22A2271AS, Rev. 1, November 30, 1978, "Preoperational Test Program", Pre-op Test Specifications.

22A3838, Rev. 1, March 8, 1976, "Recommended Prerequisites for Pre-Operational Testing". Preoperational Test Specification.

A.6 Electrical

A.6.1 Specifications

BRI Documents:

B&R Engineering Criteria Document, Rev. 11, March 16, 1982, Plus Project Criteria Advance Changes dated up to November 1, 1982, Sections D and F.

TM-330, Rev. N/A, June 28, 1972, "Medium Voltage Switchgear Basis".

TM-427, Rev. 1, February 21, 1973, "Control and Secondary Wiring Internal to Switchgear, Panels, and Similar Enclosures".

TM-443, Rev. A, March 29, 1973, "Systems Description, High Pressure Core Spray System".

TM-510, Rev. N/A, May 3, 1973, "Motor Control Center Basis".

TM-526, Rev. A, June 28, 1973, System Description, Residual Heat Removal System".

TM-671, Rev. N/A, July 5, 1974, "Contract #2 - PVC Cables".

TM-990, Rev. 1, March 11, 1977, "MCC - PCU Insulated Control Wiring".

TM-1129, Rev. N/A, August 11, 1978, "Class 1E Motor Operated Valves".

System Description #72, Rev. 0, September 25, 1975, "Feedwater System".

EM-79-006, Rev. N/A, January 2, 1979, "MCC Master List".

General Electric Documents:

21A8658, Rev. 1, May 17, 1971, "General Requirements for Motor Operated Valve Actuators - Purchase Specification".

21A9222, Rev. 2, January 11, 1974, "Electric Motors, General - Purchase Specification".

21A9222DM, Rev. 5, December 14, 1979, "Motors, Vertical (RHR) - Purchase Specification".

22A1483, Rev. 4, February 19, 1974, "HPCS System - Design Specification".

22A1483AU, Rev. 4, August 13, 1979, "HPCS System - Data Sheet".

22A2710A, Rev. 7, September 9, 1974, "Standby AC Power - BWR Requirements".

22A2711, Rev. 3, January 9, 1974, "Plant DC Power - Design Specification".

22A2817, Rev. 3, November 27, 1973, "RHR System - Design Specification".

22A2817AY, Rev. 2, October 31, 1974, "RHR System - Data Sheet".

22A3008, Rev. 5, April 8, 1977, "BWR Equipment Environmental Interface Data - Design Specification".

22A3038, Rev. 6, February 5, 1979, "Motor List, Electric - Design Specification".

22A3061, Rev. 0, September 3, 1971, "Electrical Codes and Standards - Design Specification".

22A5267, Rev. 1, May 2, 1979, "Regulatory Requirements and Industrial Standards - Design Bases".

22A7416, Rev. 0, February 19, 1981, "Electrical Equipment, Separations for Safeguards Systems - Plant Requirement".

22A2907, Rev. 3, March 28, 1974, "Feedwater Control System - Design Specification".

22A2907AB, Rev. 1, August 16, 1971, "Feedwater Control System - Data Sheet".

A.6.2 Supply System Documents

Supply System EDI-4.8, Rev. 0, September 22, 1981, "Acceptance Criteria for WNP-2 Safety Related Equipment Qualification".

A.6.3 Contracts

Contract #35, Sect. 15A, "Miscellaneous Pumps and Motors".

Contract #41A, Sect. 15A, "Nuclear Valves".

Contract #41B, Sect. 15A, "Nuclear Valves".

Contract #47A, Sect. 16A, "Medium Voltage Switchgear".

Contract #49, Sect. 16A, "Motor Control Centers".

Contract #62A, Sect. 16A, "Electrical Cable".

Contract #62B, Sect. 16A, "Electrical Cable".

Contract #62C, Sect. TP, "Electrical Cable".

Contract 218, Sect. 16A, "Electrical Installation".

A.7 Engineering Mechanics

A.7.1 Specifications

BRI Documents:

PSDG M400 through M411 - "Pipe Support Design Guide and Work Procedures" for WNP-2, Sections M400 through M411, Rev. 7, 9/16/82.

Burns and Roe, Inc. Design Guide, Rev. 0 (For piping stress analysis only, WNP-2).

TM 429 - B&R, Inc. Technical Memorandum No. 429, "Piping Loads on Equipment", 12/19/72.

TM 443 - B&R, Inc. Technical Memorandum No. 443, "System Description High Pressure Core Spray System", Rev. A, 5/4/73.

TM 482 - B&R, Inc. Technical Memorandum No. 482, "Seismic Loading for Class II Seismic Piping", 3/23/73 .

TM 1181 - B&R, Inc. Technical Memorandum No. 1181, "SRV Discharge Loads: Drywell", 9/17/80 .

TM 1223 - B&R, Inc. Technical Memorandum No. 1223, "Annulus Pressurization --- Building Response", 2/17/81.

TM 1226 - B&R, Inc. Technical Memorandum No. 1226, "Piping System Evaluation for Hydrodynamic Loads", Rev. 2, 10/30/81.

TM 1237 - B&R, Inc. Technical Memorandum No. 1237, "Chugging Loads", 7/1/81.

Engineering Criteria Document, Rev. 11, 3/16/82.

TM 1240 - B&R, Inc. Technical Memorandum No. 1240, "Functional Capability Criteria for WNP-2 Piping", Rev. 1, 2/2/82.

TM 1248 - B&R, Inc. Technical Memorandum No. 1248, "LOCA Chugging Loads on WNP-2 Submerged Structures", 11/25/81.

TM 1253 - B&R, Inc. Technical Memorandum No. 1253, "SRV Loads: Displacements", 1/13/82.

TM 1254 - B&R, Inc. Technical Memorandum No. 1254, "SRV Discharge Loads; Wetwell".

TM 1257 - B&R, Inc. Technical Memorandum No. 1257, "Structural Response Spectra", 3/5/82.

TM 1263 - B&R, Inc. Technical Memorandum No. 1263, "Hydrodynamic Loads to be Used for the DAR, Rev. 3 Assessment", 4/20/82.

TM 1059 - B&R, Inc. Technical Memorandum No. 1059, "Load Capacity of Primary Containment Weld Pads", Rev. 1, 1/31/78.

TM 1085 - B&R, Inc. Technical Memorandum No. 1085, "Pipe Break Outside of Containment - Structural Effects", 4/6/78.

TM 1020 - B&R, Inc. Technical Memorandum No. 1020, "Regulatory Guide 1.46; Recommendation Concerning Implementation", Rev. 1, 10/19/77.

TM 1151 - B&R, Inc. Technical Memorandum No. 1151, "Criteria for Pipe Break and Missile Redundancy Evaluation Outside Primary Containment", 6/27/79.

TM 1210 - B&R, Inc. Technical Memorandum No. 1210, "Statistically Derived Allowables for Expansion Bolts", 10/17/80.

TM 1271 - B&R, Inc. Technical Memorandum No. 1271, "QC II Equipment Nozzle Allowable Loads", 6/14/82.

DWG M520 - B&R, Inc. Drawing No. M520, "Flow Diagram, HPCS and LPCS Systems, Reactor Building", Rev. 27.

DWG M521 - B&R, Inc. Drawing No. M521, "Flow Diagram, Residual Heat Removal System", Rev. 35.

DWG M200-112 - Drawing "Residual Heat Removal System", Rev. 4.

DWG M200-150 - Drawing "Residual Heat Removal System", Rev. 7.

General Electric Documents:

22A1483 - General Electric Design Specification, "High Pressure Core Spray System", Rev. 4, 2/19/74.

22A2817 - General Electric Design Specification, "Residual Heat Removal System", Rev. 3, 11/27/73.

22A2887 - General Electric Design Specification, "Nuclear Boiler System", Rev. 6.

22A3790 - General Electric System Design Specification, "System Design Pressures", Rev. 0, 5/31/73.

22A3797 - General Electric Design Analysis, "Floor Response Spectra, Primary Containment", Rev. 1, 5/22/75.

761E428 - Heat Exchanger Outline Drawing, Rev. 2.

NEDO 21061 - General Electric Report, "Dynamic Forcing Functions Information Report", Rev. 3.

22A3095AD - General Electric Data Sheet, "Pressure Integrity of Piping and equipment Pressure Parts", Rev. 0.

22A3170 - General Electric Certified Design Specification, "Piping, Main Steam and Recirculation", Rev. 0.

731E932 Drawing - Process Diagram and Data Sheet for HPCS System, Rev. 3.

731E966 Drawing - Process Diagram and Data Sheet for RHR System, Rev. F.

A.7.2 Supply System Documents

Report WPPSS-74-2-R3 - Protection Against Pipe Breaks Outside Containment.

Report WPPSS-74-2-R1 - Protection Against Pipe Breaks Inside Containment.

A.7.3 Contract Specifications

C215 - Specification 2808-#215, "Mechanical Equipment Installation and Piping", Contract No. 215.

C215 15B - Section 15B, "Piping Systems", of C215 Spec.

C215 15Q - Section 15Q, "Pipe Supports", of C215 Spec.

C220 15E - Specification 2808-#220, "Instrumentation Installation", Contract No. 220, Section 15E, "Piping and Tubing Supports".

C208 - Specification C-0208, "Small Diameter Piping and Pipe Support Criteria", Rev. 1, Modification 4, 5/29/81.

A.7.4 Codes and Standards

ASME Sec. III - ASME Boiler and Pressure Vessel Code, Section III, Div. 1, 1971 Edition through Winter, 1973 Addenda.

ASME NB-3000 - Article NB-3000, "Design", of ASME Sec. III.

ASME NC-3000 - Article NC-3000, "Design", of ASME Sec. III.

ASME ND-3000 - Article ND-3000, "Design", of ASME Sec. III.

ASME NF-3000 - Article NF-3000, "Design", of ASME Sec. III.

ANSI B31.1 - American National Standard Code for Pressure Piping, "Power Piping", 1973 Edition through Winter, 1973 Addenda.

ANSI B31.1 - 101 - Section 101, "Design Conditions", of ANSI B31.1.

ANSI B31.1 - 102 - Section 101, "Design Criteria", of ANSI B31.1.

ANSI B31.1 - 104 - Section 101, "Pressure Design of Components", of ANSI B31.1.

AISC Manual - American Institute of Steel Construction, Inc. "Manual of Steel Construction", 7th Edition, 1970.

AISC Spec. - AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", 2/12/69.

ANS-58.2 - ANSI N176, "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture", Dec. 1979.

A.7.5 NRC Documents

NRC Topical Report 7/17/80 - NRC Topical Report, "Evaluation of Topical Report - Piping Functional Capability criteria", 7/17/80.

NRC RG 1.29 - NRC Regulatory Guide 1.29, "Seismic Design Classification", Rev. 3.

NRC RG 1.46 - NRC Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment", Rev. 0.

NRC RG 1.48 - NRC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components", Rev. 0.

NRC RG 1.60 - NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants", Rev. 1.

NRC RG 1.61 - NRC Regulatory Guide 1.61, "Damping for Seismic Design of Nuclear Power Plants", Rev. 0.

NRC RG 1.92 - NRC Regulatory Guide 1.92, "Combining Modal Response and Spatial Components in Seismic Response Analysis", Rev. 1.

10 CFR 50 - Title 10, Chapter 1, "Code of Federal Regulations - Energy", Part 50.

NRC I&E 79-02 - NRC Inspection and Enforcement Bulletin No. 79-02 "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts", Rev. 2, 11/8/79.

NRC SRP 3.6.1 - NRC Standard Review Plan, Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment"...

NRC SRP 3.6.2 - NRC Standard Review Plan, Section 3.6.2 "Determination of Break Locations and Dynamic Effects Associated with the Postulated Piping Failures.

SECTION B HPCS SYSTEM REVIEW REFERENCES

B.1 Mechanical Discipline References

B.1.1 Specifications

General Electric:

22A1483, Rev. 4, "Design Specification - High Pressure Core Spray System", MPL# E22-4010 dated February 19, 1974.

22A1483AU, Rev. 4, "Design Specification Data Sheet - High Pressure Core Spray System", MPL# E22-4010, dated August 13, 1979.

22A3095AD, Rev. 1, "Design Specification Data Sheet - Pressure Integrity of Piping and Equipment Pressure Parts", MPL# A62-4030 dated September 26, 1973.

22A2702AB, Rev. 1, "Design Specification Data Sheet 2, Seismic Design", MPL# A62-4090, dated January 11, 1972.

22A3067, Rev. 3, "Design Specification - Mechanical Equipment Separation", MPL# A62-4350, dated August 21, 1975.

21A1740, Rev. 3, "Purchase Specification - Valve, Gate", MPL# E22-F004, dated 1/13/72.

21A1884, Rev. 2, "Purchase Specification - Valve Data Specification, Valve, Gate", MPL# E22-F004, dated 1/14/75.

21A9243, Rev. 0, "Purchase Specification - Auxiliary Pumps for Boiling Water Reactors", MPL# E22-C001, dated May 1, 1973.

21A9243DE, Rev. 2, "Purchase Specification Data Sheet - High Pressure Core Spray Pump", MPL# E22-C001, dated October 29, 1973.

21A1880, Rev. 1, "Purchase Specification - Valve Data Specification, Valve Gate", MPL# E22-F012, Dated April 18, 1972.

21A1736, Rev. 3, "Purchase Specification - Valve Data Specification - Valve, Gate", MPL# E22-F012, dated January 25, 1972.

B&R Contract Specifications

2808-215, Section 15A, 15B, 15F, 15G
2808-69, Section 15A
2808-213

PDM Contract Purchase Specification,

PUSP-16713-3 Rev. B, dated August 7, 1980.

B&R Engineering Criteria Document, Sections E, F and I

B.1.2 Calculations

B&R Calculations:

5.19.01, Rev. 0, "HPCS Pipe Sizing", June 15, 1971.

5.19.02, Rev. 0, "HPCS System - Preliminary Line Sizing" December 12, 1971.

5.19.07, Rev. 0, "HPCS Piping Schedule", April 18, 1973.

5.19.08, Rev. 0, "Restrictors - HPCS System", September 12, 1975.

5.19.10, Rev. 0, "ECCS Minimum NPSH Calculations - R.G. 1.1", Rev. 0, November 10, 1976.

5.19.11, Rev. 4, "Pressure Drop Calculation HPCS System", August 20, 1981.

5.19.12, Rev. 0, "HPCS System - Water Leg Low Pressure Alarm", September 6, 1979.

5.19.13, Rev. 1, "Sizing of HPCS Emergency Water Volume", September 15, 1981.

5.19.14, Rev. 0, "NPSH of HPCS Pump - Maximum Allowable Suppression Pool Temperature", September 10, 1980.

10.04.71, Rev. 0, "WPPSS Hanford No. 2 Condensate Tank NRC Question 211.61", June 10, 1979.

10.04.72, Rev. 0, "WPPSS NP#2 - Analyze Vortex Formation at the HPCS/RCIC Suction Inlet in the CST", August 25, 1980.

B.1.3 Technical Memoranda

B&R

Technical Memorandum 443 Rev. A "System Description High Pressure Core Spray System", March 12, 1973.

B.1.4 Manuals

General Electric Operation and Maintenance Instructions:

GEK 71334, "High Pressure Core Spray System", July 1978.

VPF 3238-842-2, Rev. B, "Instruction Manual - Motor-Operated Gate Valves, GE Order 205-AE204, Darling E-5310" dated June 21, 1980.

VEL-HO-1 "Velan Operation and Maintenance Manual - Check Valves".

VPF 3069-30-3, "Ingersoll-Rand O&M Manual - High Pressure Core Spray Pump", March 19, 1975.

B.1.5 Drawings

General Electric Drawings:

731E931AD Rev. 0, "P&ID-HPCS System" MPL# E22-1010, dated July 30, 1974.

731E931 Rev. 7, "P&ID - HPCS System" MPL# E22-1010, dated May 22, 1974.

731E932AD Rev. 3, "Process Diagram-HPCS System", MPL# E22-1020, dated October 22, 1978.

B&R Drawings:

M520 Rev. 33	M732, Rev. 18
M711 Rev. 22	M626, Rev. 5
M712 Rev. 29	SM197, Rev. D
M713 Rev. 29	SM193, Rev. E
M714 Rev. 25	SM191, Rev. E
M715 Rev. 27	SM183, Rev. E
M716 Rev. 27	SM136, Rev. D
M718 Rev. 33	SM135 Rev. D
S798 Rev. 31	M567 Rev. 6
S796 Rev. 14	M200, Sht. 132 Rev. 5
S795 Rev. 41	M200, Sht. 100 Rev. 7A

M744 Rev. 7 M200, Sht. 101 Rev. 9
M527 Rev. 37 M200, Sht. 2 Rev. 5
M569

PDM Drawings:

E37 Rev. A2 E61, Rev. B

CB&I Drawings:

72-4396-2 Rev. 6, 72-4396-1A Rev. 7 72-2647-1 Rev. 8
72-2647-123 Rev. 7

Zurn Drawings:

I-80120-A (B&R Transmittal 213B-1231B)

Anchor-Darling Drawings:

94-13262 Rev. E 94-13306 Rev. C 94-13401 Rev. B

Velan Drawing:

P2-2767-N-2 Rev. L

J. E. Lonergan Drawing:

A-2647 Rev. A

Ingersoll-Rand Drawings:

D-12X20KD86XEZ D-12X20KD0321XZC

Permulit Drawing:

556-30530 Rev. 9

Isometric Drawings:

HPCS-629-1.4	HPCS-629-5.7	HPCS-630-1.4
HPCS-630-5.6	HPCS-630-7.10	HPCS-630-11.12
HPCS-630-13.19	HPCS-630-20.23	HPCS-630-24.25
HPCS-630-26.28	HPCS-630-29.30	HPCS-630-31.33
COND-351-1.9	COND-351-10.15	HPCS-632-1.3
HPCS-633-1.2	HPCS-1458-1	HPCS-1458-2
HPCS-1458-3	HPCS-1458-4	HPCS-1459-1
HPCS-1459-2	HPCS-1460-1	HPCS-1461-1
HPCS-2569-1	HPCS-1644-1	HPCS-2568-1
HPCS-2570-1	HPCS-1958-1	HPCS-2571-1
D-220-X-78		

B.1.6 <u>WNP-2 FSAR</u>

Sections 3.2, 6.1, 6.2, 6.3, 15

B.1.7 <u>Other References</u>

"High Pressure Core Spray System Design Reverification Plan", Revision 1, System design Engineering, WPPSS, dated February 20, 1983.

SDEI-3.5 "Design Reverification", Revision 3, System Design Engineering Instruction 3.5, WPPSS, dated December 8, 1982.

TDP 3.4 "Preparation, Verification, and Control of Calculations", June 8, 1982

"WNP-2 Plant Verification Report" WPPSS, dated June 1982

VPF-3069-91-1, "Certified Test Report for HPCS-P-1", April 10, 1975

Form N-5, Data Reports for Field Installation of Nuclear Power Plant Components, Component Supports and Appertenances (by HPCS System Line and Code Class)

Crane Technical Paper No. 410, "Flow of fluids Through Valves, Fittings, and Pipe", Crane Co., Twentieth Printing - 1981.

NEDM-20363-13, "Hydraulic Analysis Procedures for BWR Piping Systems", GE, September 1975.

AEC-TR-6630, "Handling of Hydraulic Resistance, Coefficients of Local Resistance and of Friction", I.E. Idel'chik, 1960.

B.2 Mechanical Diesel Discipline References

B.2.1 Specifications

22A1483, Rev. 4, High Pressure Core Spray Systems Design Specifications.

22A1483AU, Rev. 4, HPCS Design Specification Data Sheet.

21A1848AB, Engine Generator for HPCS Purchase Specification Data Sheet.

21A1848, Engine Generator for HPCS Purchase Specification.

21A1776, Rev. 1, HPCS Diesel Service Water Pump Purchase Specification.

21A1776AD, Rev. 1, HPCS Diesel Service Water Pump Purchase Specification Data Sheet.

A990 GEAPPD, Thermxchanger Exchanger Specification Sheet.

Contract 215, Material Specifications.

B&R Engineering Criteria Document.

B.2.2 Calculations

B&R Nuclear Calculations:

5.43.01, "Diesel Engine System Calculations" Rev. 0, 2/19/74.

5.43.02, "Diesel Oil Tanks (Storage and Day Tanks) Capacity Verification", Rev. 0, 8/9/79.

B.2.3 Technical Memoranda

B&R HPCS Diesel Generator Technical Memorandum:

TM-0558	D.G. Synchro - Check Relays
TM-0586	Emergency D.G. Operation
TM-0775	Diesel Generator Loading
TM-0746	Interlocking for Diesel Generator
TM-0608	Diesel Generator Cooling Water System
TM-1053	Standby D.G. Light Load Operation
TM-1066	Gas Disper. & Met Analysis
TM-0817	System Description, D.G. Systems
TM-0443 (Rev. A)	System Description

B.2.4 Manuals

Instructions/Parts Manual for Hanford II Diesel Generator, Contract No. 205-AD583, PSD IWO No. A-990, by Power Systems Division, Book One, Sections 8, 10, 14 MI 1748 Rev. B, Doc. No. 3316-031

Pacific Pumps Instruction Manual
CVI 2-2E22-13-11

B.2.5 Drawings

Isometric Drawings:

DE-797-1.5	Rev. 7	1/25/83
DE-789-1.3	Rev. 5	10/12/82
DE-1738-1	Rev. 7	12/2/82
DE-2836-1	Rev. 5	12/1/82
DSA-4275-1	Rev. 5	12/10/82
DSA-4396-1	Rev. 4	10/12/82
DSA-4396-2	Rev. 5	9-21-82
DSA-2536-1	Rev. 1	11/19/82

Isometric Drawings (Contd.)

DSA-2537-1	Rev. 4	8/6/82
DSA-2537-2	Rev. 5	7/24/82
DSA-2537-3	Rev. 5	8/2/82
DSA-2537-4	Rev. 5	11/29/82
DSA-2537-5	Rev. 4	8/19/82
DO-448-1B		
DO-1620-1		
DO-1620-2		
DO-2530-3		
DO-2531-1		
DO-2531-3		
DO-2532-1		
DO-2532-2		
DO-2532-3		
DO-2533-1		
DO-2533-2		
DO-2533-3		
DO-2675-1		
DO-2797-1		
DO-4328-1		
DCW-2510-1		
DCW-2510-2		
DW-1965-11		

B&R Flow Diagram M-512

GE Piping Diagrams for HPCS Diesel Engine Generator

A990D08001	HPCS Diesel Engine Generator Air Intake Piping Schematic
A990D09001	HPCS Diesel Engine Generator Exhaust System Piping Schematic

A990F03001	HPCS Diesel Engine Generator DLO Schematic Diagram
A990F04001	HPCS Diesel Engine Generator Jacket Water System with Heat Exchanger Schematic Diagram
A990C06002	Fuel Oil Schematic Lister SR1A Diesel Engine
A990F06001	HPCS Diesel Engine Generator F.O. Schematic Diagram
A990F07001	HPCS Diesel Engine Generator Air Start System Schematic Diagram
A990F02001	HPCS Diesel Engine-Generator Assembly

B.2.6 Other References

I&E Bulletins - HPCS Diesel Generator

Emergency D.G. Lube Oil Addition and Onsite Supply,	80-04
Potential D.G. Turbocharger Problem	79-12
Degradation of Fuel Oil Flow to the Emergency D.G.	77-15
Emergency D.G. Lube Oil Cooler Failures	80-11

Standards of Tubular Exchanger Manufacturers Association

DEMA Standard Practices for Low and Medium Speed Stationary Diesel Engines

ASTM Standards, Part 17, Classification of Diesel Fuel Oils

National Fire Protection Association Standards 30, 37, and 70

SLT-57.2-5 (Rev. 0) HPCS Diesel Engine Jacket Cooling Water Flush and Fill

B.3 I&C Discipline References

B.3.1 Specifications

General Electric:

22A1483	Rev. 4	Design Specification, HPCS System
22A1483AU	Rev. 4	HPCS System Data Sheet
22A2988	Rev. 6	Electrical Separation (See 22A7416)
22A3008	Rev. 5	BWR Equipment Environmental Interface Data
22A3039	Rev. 1	Process Instrumentation
22A3067	Rev. 3	Mechanical Equipment Separation
22A3095	Rev. 0	Process Integrity of Piping and Equipment
22A3746	Rev. 1	Local Instrument Panels
22A7416	Rev. 0	Electrical Equipment Separation

Burns and Roe:

Design Criteria Sections F and G

B.3.2 Calculations

Burns and Roe:

5.51.051	Target Determination Pipe Break Outside Containment
8.01.203	Pipe Break Locations

B.3.3 Technical Memorandum

Burns and Roe:

TM-1151, Criteria for Pipe Break and Missile Redundancy Evaluation

Letter BRBEC-F-82-3752, dated October 21, 1982
Letter BRBEC-F-83-2174, dated March 22, 1983

B.3.4 Manuals

General Electric

GEK 71334	July 1978	High Pressure Core Spray System, O&M
GEK 71337	June 1978	Vendor Supplied Instruments, O&M

Dragon Valves, Inc.

12583	Rev. 0	Excess Flow Check Valve Instrument Manual

B.3.5 Drawings

General Electric

127D1840TC	Rev. 2	HPCS Instrument Panel Arrangement
163C1043TC	Rev. 1	HPCS Instrument Panel Piping Diagram
731E931AD	Rev. 7	HPCS P&ID
731E950AD	Rev.	HPCS FCD
234A9309TC	Rev. 3	Instrument Data Sheets
807E172TC	Rev. 19	Elementary Diagrams HPCS System

Burns and Roe

7E015	Rev. 2	Electrical Wiring Diagram HPCS-V-1
7E016	Rev. 2	Electrical Wiring Diagram HPCS-V-4
7E017	Rev. 1	Electrical Wiring Diagram HPCS-V-10
7E018	Rev. 1	Electrical Wiring Diagram HPCS-V-11
7E019	Rev. 2	Electrical Wiring Diagram HPCS-V-12
7E020	Rev. 2	Electrical Wiring Diagram HPCS-V-15

7E021	Rev. 1	Electrical Wiring Diagram HPCS-V-23
7E025	Rev. 1	Electrical Wiring Diagram Controls Sheet 1
7E02	Rev. 1	Electrical Wiring Diagram Controls Sheet 1
S 709	Rev. 26	Structural, Reactor Building
E522	Rev. 16	Elem. Diag. Isolation Valves
E535-18A	Rev. 9	Connection Wiring Diag. MC4A
-18B	Rev. 7	Connection Wiring Diag. MC4A
E536-2C	Rev. 12	Connection Wiring Diag. Term. Box and Misc.
-5B	Rev. 12	Connection Wiring Diag. Term. Box and Misc.
E537-IV	Rev. 3	Connection Wiring Diag. Term. Box and Misc.
-3A	Rev. 9	Connection Wiring Diag. Control Room Term. Cab.
-4B	Rev. 8	Connection Wiring Diag. Control Room Term. Cab.
-26A	Rev. 7	Connection Wiring Diag. Control Room Term. Cab.
E539-2	Rev. 10	Connection Wiring Diag. Reactor I&C
-14	Rev. 10	Connection Wiring Diag. Reactor I&C
-21	Rev. 8	Connection Wiring Diag. Reactor I&C
E540-4	Rev. 5	Connection Wiring Diag. Motor Op. Valves
-6	Rev. 10	Connection Wiring Diag. Motor Op. Valves
M520	Rev. 32	Flow Diagram, HPCS and LPCS Systems
M527	Rev. 44	Flow Diagram, Condensate Supply System
M530	Rev. 32	Flow Diagram, Nuclear Boiler Recirculation System
M543	Rev. 33	Flow Diagram, Containment Cooling and Purging
M567	Rev. 6	General Arrangement Reactor Bldg. El 422'and 441'
M568	Rev. 22	General Arrangement Reactor Bldg. El 471' and 501'
M619-VI	Rev. 5	Instrumentation Contract 220, General Notes
M609	Rev. 10	Instrument Process AZ 0° to 180°
M619-6	Rev. 4	Instrument Conn. Diag. H22-P009
-19	Rev. 4	Instrument Conn. Diag. H22-P024

M623	Rev. 8	Instrument Process Plan El 501'
M624	Rev. 8	Instrument Process Plan El 512'
M625	Rev. 11	Instrument Process Plan El 522'
M626	Rev. 7	Instrument Process Plan El 541'
M627	Rev. 13	Instrument Process Plan El 560'
M628	Rev. 17	Instrumentation Process Penetration Schedule
M629	Rev. 7	Instrumentation Process Partial Plans and Sections
M734	Rev. 24	Miscellaneous Piping Plan and Sections at El 471'

Johnson Controls

B-220-007.0-H22P024	Rev. 3	Line Identification List (20 sheets)
B-220-X-73	Rev. 2	Line Identification List X-73
-86A	Rev. 1	Line Identification List X-86A
-86B	Rev. 3	Line Identification List X-86B
-87A	Rev. 2	Line Identification List X-87A
-87B	Rev. 3	Line Identification List X-87B
D-220-007.0-H22P024	Rev. 3	Tube Erection Isometrics (20 Sheets)
D-220-7.1-X-732-1	Rev. 2	Process Instrument Line X-73a
-1A	Rev. 1	Process Instrument Line X-73a
-1B	Rev. 1	Process Instrument Line X-73a
-1C	Rev. 1	Process Instrument Line X-73a
D-220-X-73	Rev. 0	Process Instrument Line X-73
-86A	Rev. 1	Process Instrument Line X-86A
-86B	Rev. 2	Process Instrument Line X-86B
-87A	Rev. 1	Process Instrument Line X-87A
-87B	Rev. 1	Process Instrument Line X87B

D-220-3500-250-CMS-LT-1&2, Rev. 1, Local Instrument Installation

E-220-5500-RB-441	Rev. 6	Tube Routing React. Bldg. El. 441'
-471	Rev. 6	Tube Routing React. Bldg. El. 447'
-501	Rev. 5	Tube Routing React. Bldg. El. 501'
-522	Rev. 6	Tube Routing React. Bldg. El. 522'
-548	Rev.	Tube Routing React. Bldg. El. 548'

Bovee and Crail

HPCS-630-7.10	Rev. 8	Discharge from HPCS-P-1 to RPV

Gilbert/Commonwealth

COND-4631-1	Rev. 3	RCIS-HPCS Switchgear Standpipe
-2	Rev. 2	RCIS-HPCS Switchgear Standpipe
-3	Rev. 3	RCIS-HPCS Switchgear Standpipe
-4	Rev. 3	RCIS-HPCS Switchgear Standpipe
-5	Rev. 3	RCIS-HPCS Switchgear Standpipe
-6	Rev. 3	RCIS-HPCS Switchgear Standpipe

Dragon Valves, Inc.

C-12583	Rev. E	Excess Flow Check Valve

Daniel Industries

C-2629		ANS Orifice Flange, Upstream
C-2630		ANS Orifice Flange, Downstream

B.3.6 Contract Specifications

21A9376	Rev. 1	Flow Orifice Assembly
21A9376AJ	Rev. 1	Flow Orifice Assembly Data Sheet
21A9417AB	Rev. 0	I.D.S. HPCS System
2808-220		Johnson Controls

B.3.7 Other References

Drawing Control Log, Dated 2-13-83

WNP-2 NUREG-0588, Environmental Equipment Qualification Report

B.4 Electrical Discipline References

B.4.1 Specifications

B&R Engineering Criteria Document, Rev. 11, March 16, 1982, plus Project Criteria Advance Changes dated up to April 6, 1983, Section D and Appendix 3, "WNP-2 Electrical Separation Practices", Rev. 2, March 21, 1983.

21A1884, Rev. 2, April 23, 1975, "HPCS Gate Valve Data-Purchase Specification".

21A8658, Rev. 1, May 17, 1971, "General Requirements for Motor Operated Valve Actuators".

21A9222, Rev. 2, January 11, 1974, "Electrical Motors, General - Purchase Specification".

21A9222DL, Rev. 3, July 18, 1980, "HPCS Pump Vertical Motor Data Sheet - Purchase Specification".

21A9300, Rev. 3, July 25, 2978, "Switchgear Electrical Metal Enclosed for HPCS - Purchase Specification".

21A9300AD, Rev. 3, August 17, 1974, "Metal Enclosed Electrical Switchgear - Purchase Specification - Data Sheet".

21A9301AJ, Rev. 3, November 26, 1974, "HPCS Motor Control Center - Purchase Specification - Data Sheet".

22A1483, Rev. 4, August 7, 1974, "HPCS Design Specification".

22A1483AU, Rev. 2, August 13, 1979, "HPCS Design Specification - Data Sheet".

22A2710A, Rev. 7, September 9, 1974, "Standby AC Power - BWR Requirements".

22A7416, Rev. 0, March 5, 1981, "Electrical Equipment Separation for Safeguards System BWR Plant Requirements Specification".

22A3008, Rev. 5, April 8, 1977, "BWR Equipment Environmental Interface Data - Design Specification".

22A3061, Rev. 0, September 3, 1971, "Electrical Codes and Standards - System Design Specification".

22A7416, Rev. 0, March 5, 1981, "Electrical Equipment, Separation for Safeguards Systems".

238X185AD, Rev. 13, December 29, 1981, "HPCS Parts List".

22A3038, Rev.6, "List of Electric Motors - Design Specification".

B.4.2 <u>Calculations</u>

2.02.02, Rev. 1, "Main Plant One-Line Auxiliary Load Calculations".

2.02.07, Rev. 2, "Motor Control Centers - Load Calculations".

2.02.16, Rev. 1, "Load Summary - Major Plant Operating Modes".

2.02.18, Rev. 0, "Volt. Switchgear Load Study".

2.03.02, Rev. 5, "Main One-Line Short Circuit Calculation".

2.03.07, Rev. 2, "480 V Switchgear Short Circuit Calculations".

2.03.09, Rev. 0, "Motor Control Center Short Circuit Calculations".

2.05.01, Rev. 3, Battery and Battery Charger Calculation 250 VDC, 125 VDC and 24 VDC Systems".

2.06.03, Rev. 5, "Main One-Line Voltage Drop Calculations".

2.06.07, Rev. 1, "Service and Diesel Generator Building Feeder and Voltage Drop Calculation".

2.06.10, Rev. 1, "Service and Diesel Generator Building Feeder and Voltage Drop Calculation".

2.06.17, Rev. 0, "4160/6900 V. Motor Feeder Cable - Voltage Drop".

2.07.01, Rev. 2, "High Voltage Cable Sizing - Ampacities and Conduits".

2.07.05, Rev. 0, "Cable Sizing - 4.16 and 6.9 KV - Short Circuit Capacity".

2.07.09, Rev. 3, "125 VDC System Cable Sizing for Circuit Breakers".

2.07.10, Rev. 0, "D.C. System Cable Sizing for Voltage Drop Calculation".

9.21.02, Rev. 0, "Reactor Building - Emergency Cooling and Critical Area Cooling System".

9.24.00, Rev. 6, "HVAC - Diesel Generator Building".

B.4.3 Technical Memoranda

B&R Technical Memo #1060, Rev. 3, January 22, 1980, "Voltage Drop Study".

B.4.4 Manuals

CVI 49-00,25, Issue 1, "ITE Instruction Manual" for Motor Control Centers.

VPF 3395-27, "Installation Operation and Maintenance and Instruction Manual for HPCS Metal Clad Switchgear".

VPF 3390-12, "Switchgear Equipment Instruction Manual".

CVI 2-02E22-09, 10, Issue 1, "ICS Manual".

AEF 62B-00-0112, "Instrumentation Control Power Coaxial and Triaxial Cables Installation Instruction Manual".

B.4.5 Drawings.

Burns and Roe

E WD-7E-003,	Standby Water Leg PP. HPCS-P-3 (E22-C003)	Rev. 1,	05/17/82
E WD-72-0016	M.O.V. HPCS-V-4 (E22-F004)	Rev. 2,	08/31/82
E 502-2	Main One Line Diagram	Rev. 19	01/19/83
E 503-9	Aux. One Line Digaram	Rev. 16	12/18/82
E 514-6	Relay Settings 4.16 KV Switch-gear SH-4	Rev. 6	12/18/82
E 517-9	4160 V SWGR, Elem. Diag.	Rev. 18	02/02/83
E 550	Cable Schedule - Power	Rev. 26	03/17/83
E 553-1	Class 1E Electrical Equip List	Rev. 4	03/28/83
E 558-2	Turb. Gen. Bldg. Grounding Plans and Details	Rev. 4	04/12/82
E 662-1	Reactor Bldg. Grounding Plans and Details Sh. 1	Rev. 11	10/07/82

E 680	Reactor Bldg. El 422'3" Power Conduit and Tray Plan	Rev. 17	12/30/82
E 785	Diesel Gen. Bldg. El. 441'-0" Power and Tray Plans	Rev. 32	01/21/83

PED 218-E-4533

807E183TC, Sheets 1 to 6, overall Rev. "HPCS Power Supply Elementary Diagram".

807E172TC, Sheets 1 to 8, overall Rev. "HPCS Elementary Diagram".

992C349BC, Rev. 4, "HPCS Pump Motor Outline".

731E302AD, Sheets 1 to 3, overall Rev. "HPCS Power Supply One Line Diagram".

VPF 3395-9, Rev. , "HPCS Motor Control Unit Wiring Diagrams".

VPF 3395-10, Rev. , "HPCS Motor Control Unit Wiring Diagrams".

VPF 3395-11, Rev. , "HPCS Motor Control Unit Wiring Diagrams".

VPF 3395-2, Rev. , "480V Motor Control Center for HPCS".

VPF 3395-2, Rev. , "480V Motor Control Center for HPCS Bill of Material".

147C1614, Rev. 1, "HPCS Transformer Outline".

5528, Rev. , "HPCS Transformer Nameplate Detail".

0123D3805, Rev. , "HPCS Metal Clad Switchgear Interconnection".

B.4.6 Memoranda

NEDO-10905, Rev. 3, "Topical Report HPCS Power Supply Unit and Amendments".

GEWP-2-81-189, HPCS Relay Settings.

EM-79-006, Rev. 0, January 2, 1979, "MCC Master List".

B.4.7 Contract Specifications

Contract 2, Division 2, Section 2A, "Nuclear Steam Supply System".

Contract 35, "Miscellaneous Pumps and Motors", Division 15, Section 15A.

Contract 49, Division 16, Section 16A, "Motor Control Centers".

Contract 62A, Division 16, Section 16A, "Electrical Cable".

Contract 62B, Division 16, Section 16A, "Electrical Cable".

B.4.8 Other References

IEEE 141-1976, "The Red Book - Recommended Practice for Electric Power Distribution for Industrial Plants".

IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations".

IEEE 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations".

IEEE 308-1971, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations".

NEMA-MG-1-1978, "Motor and Generator Standards".

NEMA-ICS-1-108, "Service and Installation Conditions".

NEMA-ICS-2-321, "AC General-Purpose Class A Magnetic Controllers for Induction Motors Rated in Horsepower 600 Volts and Less, 50 and 60 Hertz".

NEMA-ICS-2-322, "AC General-Purpose Motor Control Centers".

NEMA-ICS-2-327, AC General-Purpose Class A Magnetic Controllers for Induction Motors, Rated in Full-Load and Locked-Rotor Current, 600 Volts and Less, 50 and 60 Hertz".

IPCEA Pub. No. S-68-516, Interim #2, "Cables Rated 5000 Volts and Less and Having Ozone-Resistant Ethylene-Propylene-Rubber Integral Insulation and Jacket".

ICEA Pub. No. P-54-440 (Second Edition), "Ampacities - Cables in Open-Top Cable Trays".

NFPA-70-1981, "National Electrical Code".

IEEE Conference Paper C72-121-7, "IEEE Flame Test Report".

Washington Public Power Supply System WNP-2 Class 1E Equipment List (WNP-2 C1E List) dated 3/2/83.

Crane-Deming Pumps - Test No. T-552B - Item #13 - HPCS Water Leg Pump.

Westinghouse "Report of Test Form for Induction Motors", Form A-2, Induction Motor L-980864-03-A dated 5/4/76.

Gould/Shawmut Bulletin AT-620R, "UL Class RK-5 Current Limiting Fuses".

WNP-2 Master Equipment List (MEL).

Okonite Bulletin 721.1 "Engineering Data for Copper and Aluminum Conductor Electrical Cables".

Okonite Bulletin 776, "Okonite Cable".

Cable Pull Slips - Fishback/Lord - for Cables 3HPCS-0030, 3HPCS-0080, 3HPCS-0340.

WNP-2 FSAR Appendix F.

Darling Valve and Manufacturing Company, "Test Data Report", Shop Order No. E5310-4-1, Customer P. O. 205-AE204, Valve Tag No. E22-F004, dated 9-20-74.

Anchor-Darling Valve Co. letter "Certification", P.O. No. 205-AE204, MPL No. E22-F004. Dated 10-7-74.

B.5 Engineering Mechanics Discipline References

B.5.1 Specifications/Codes/Guides

B&R Engineering Criteria Document, Sections E and I, Appendies A and 2 Rev. 3.

ASME B&PV Code, Section III, 1971 Edition, including Addenda through Winter, 1973.

B&R Piping Design Guide, Rev. 3.

WNP-2 FSAR.

U.S. NRC Standard Review Plan 3.6.2.

G.E. Design Specification 22A2887, Rev. 6, for the Nuclear Boiler System.

G.E. Document # 22A1483, Rev. 4 (HPCS System Design Specification).

ASME B&PV Code Case N-122(1745) 03/01/76.

Westinghouse Structural Analysis Program PIPSAN.

Contract 208 Specification, "Small-Diameter Piping and Pipe Support Criteria" Section.

B&R Memo EM-82-322, 6/28/82.

WRC Bulletin 107, March 1979, Revision.

ASME Code Case N-318, Rev. 0.

TPIPE User's Manual, Version G/C 4.3.

G/C Engineering Handbook, User's Manual, Rev. 7.

B&R Project Engineering Directive, PED-C0208-0689.

Johnson Controls, Inc. Design Guide for WNP-2.

AISC Steel Construction Manual, 7th Edition.

General Electric Document 21A9243 DE, Rev. 1, "Specification and Data Sheet - HPCS Pump".

General Electric "Operating Instructions for HPCS Pump", 3/19/75.

Building Code Requirements for Reinforced Concrete, ACI 318-71.

ANSI B31.1 Power Piping Code, 1973 Edition, W73 Addenda.

Contract 208 Specification, "Small-Diameter Piping and Pipe Support Criteria" section.

B.5.2 Calculations

B&R Calculation 8.14.64A, Rev. 0.

Burns and Roe, Inc. Calculation # 8-70-02 (Thermal Expansion).

B&R McDonald Douglas STRUDL run #1219, 68 pages, November 1, 1982.

B&R McDonald Douglas STRUDL run #2016, 65 pages, October 29, 1982.

B&R Calculation 8.15.65 for HPCS-52.

B&R Calculation 8.15.225 for HPCS 901N.

Gilbert/Commonwealth Calculation No. DE-1738-1, Rev. 7.

G/C Calculation #00010, Rev. 1, Design Guide for Shear Lugs.

Pipe Support Calculation No. JCI-220-CLC-961, Rev. 1.

U-Bolt Calculation No. JCI-220-CLC-529, Rev. 2.

Pipe Stress Calculation NUPIPE Run X-73AIN, T-20, 83/03/24.

Base Plate Calculation JCI-220-CLC-997, Rev. 1.

Piping Analysis Program Summary, X-73a IN, 3/28/83.

Pipe Support Calculation, DE-1738-11 and 11A, Rev. 8.

Pipe Stress Analysis, TPIPE run BC2RFWC, Rev. 7, Run 5.

G/C Calculation No. 0000-12, Rev. 0.

Pipe Support Calculation No. 8.15.1133, Rev. 3.

NUPIPE Printout "X73AIN AS-BUILT CONFIG", 82/09/08.

JCI Calculation 220-CLC-4119, Rev. 0.

JCI Piping Analysis Program Summary, File X-73a, Trial #20.

B&R Calculation 8.14.82 11/10/82.

Pipe Support Calculation No. 8.15.1076, Rev. 2.

Pipe Stress Calculation No. 8.14.82, Support Load Summary Sheets, Rev. 9.

B&R Calculation No. 6.17.22, Book SV-72.

B.5.3 Technical Memoranda

T.M. 1226, Rev. 3, "Piping System Evaluation for Hydrodynamic Loads".

T.M. 1240, Rev. 1, "Functional Capability Criteria for WNP-2 Piping".

Tech. Memo #1253 (SRV Displacements).

Tech. Memo #1181, Rev. 1 (SRV Response Spectra).

Tech. Memo #1257, Rev. 2 (Seismic and Hydrodynamic Response Spectra).

Tech. Memo #1283 (Reduction of SRV Loading).

B.5.4 Manuals

"Weldolet Stress Intensification Factors", Bonney Forge, 1976.

"NAVCO Piping Datalog", Edition No. 10, 1974, National Valve and Manufacturing Company, Pittsburgh, PA.

"ANSYS Rev. 4 User's Manual", Rev. A, 2/1/82, Swanson Analysis Systems, Inc.

ADLPIPE User Manual, Rev. J, 11/18/82, issued by CDC.

TPIPE User's Manual, Version G/C 4.3.

Project Engineering Directive 220-M-0853, 07/23/82.

ADLPIPE Input Preparation Manual, May 1981 Revision.

ADLPIPE Reference #16, "Lumped Mass Location", March, 1975.

B.5.6 Drawings

B&R Stress Isometric, M200-Sh. 100, Rev. 7A.

Bovee and Crail Construction Isometric, HPCS-629-1.4, Rev. 9FO (as-built).

Pittsburgh-Des Moines Steel Co., AB-E68, Rev. N, 6/9/82, Wetwell Piping.

B&R Flow Diagram, M-520, Rev. 33.

G.E., 731E932AD, Rev. 3, HPCS Operating Conditions.

PDM, D-101, Rev. K, Penetration X-31 Details.

Anchor/Darling, 2621-3, Rev. B, Valve HPCS-V-16.

Anchor/Darling, 94-13473, Rev. A, Valve HPCS-V-15.

PDM, AB-E150, Wetwell Piping and Support Details.

PDM, AB-E118-31, Wetwell Piping and Support Details.

PDM, AB-E12, Wetwell Piping and Support Details.

B&R, S-795, Penetration Dimensions.

B&R Support Detail Drawing, HPCS-900N.

B&R Support Detail Drawing, HPCS-901N.

B&R Support Detail Drawing, HPCS-52.

B&R Project Engineering Directive, PED-C0208-0689.

M200-SHT2-1, Rev. C (HPCS Isometric).

M200-SHT2-2, Rev. A (HPCS Supports Orientation).

S795 (X-6 Penetration Detail).

M601, Rev. 20 (Valve List).

HPCS-63, Rev. 3 (Support Detail).

HPCS-912N, Rev. 1 (Support Detail).

HPCS-911N, (Support Detail).

HPCS-910N, Rev. 1 (Support Detail).

HPCS-918N, Rev. 2 (Support Detail).

HPCS-919N, Rev. 2 (Support Detail).

HPCS-66, Rev. 3 (Support Detail).

HPCS-904N, Rev. 2 (Support Detail).

HPCS-906N, Rev. 1 (Support Detail).

HPCS-64, Rev. 2 (Support Detail).

HPCS-907N, Rev. 1 (Support Detail).

HPCS-908N, Rev. 2 (Support Detail).

Pipe Isometrics, B&R M200-606, Rev. 4.

Pipe Isometric, G/C DE-1738-1, Rev. 8.

Flow Diagram, B&R M512, Rev. 27.

Valve drawing, Borg-Warner # 38020, Rev. F.

Pipe Support Drawing, B-220-670-35, Rev. 1.

Pipe Fabrication Isometric, D-220-7.1-X-73a, Rev. 2.

Pipe Isometric and Support Drawings, DE-1738-1, 3 sheets.

Pipe Support Drawing, HPCS-910N, Rev. 3Fo.

Pipe Fabrication Isometric, HPCS-630-26.28, Rev. 8.

JCI D-220-7.1-X-73a, As-Built.

JCI Pipe Support Drawings, as listed in D-220-7.1-X-73a.

B&R Blow Diagram, M520, Rev. 27.

Dragon Valve Drawing C-10580, Rev. 0.

Pipe Fabrication Drawing HPCS-630-29.30, Rev. 7.

Pipe Support Standard Drawings H501 (5 sheets), H502 Rev. 0, H503 Rev. 0.

B&R Drawing S-701, Rev. 9.

B&R Drawing S-702, Rev. 6.

B&R Drawing S-749, Rev. 17.

B&R Drawing S-750, Rev. 21.

B&R Drawing S-660, Rev. 28.

Ingersoll-Rand Pump Drawing C-12X20KD86X2-H, Rev. 6.

General Electric

167E2054, Rev. 0 (Nozzle Thermal Transients).

731E932AD, Rev. 3 (HPCS Thermal Modes).

761E716 (RPV Nozzle Allowable Loads).

Bovee and Crail

HPCS-630-31.33, Rev. 9 (Construction Drawings).

HPCS-630-29.30, Rev. 8 (Construction Drawings).

HPCS-630-26.28, Rev. 9FO (Construction Drawings).

Velan Dwg. #P2-2767-N-2, Rev. L (Valve HPCS-V-5).

Velan Dwg. #P2-3311-N-16, Rev. F (Valve HPCS-V-51).

Anchor/Darling Dwg. #2652-3, Rev. B (Valve HPCS-V-76).

B.5.7 Memoranda

U.S. NRC Memo, "Evaluation of Topical Report - Piping Functional Capability Criteria", R. L. Tedesco from J. P. Knight, 7/17/80.

Memo from J. Braverman to R. E. Snaith on 2/10/81 (Seismic Anchor Motions).

Memo from D. Bagehi to R. E. Snaith on 2/1/82 (Seismic Anchor Motions).

B.5.6 Other

Velan Valve Stress Report # SR-6335.

SECTION C RHR SYSTEM REVIEW REFERENCES

C.1 Specifications

21A3757, Rev. 0, GE Purchase Specification for Relief Valves on RHR Heat Exchangers.

21A3757AA, Rev. 2, GE Purchase Specification Data Sheet for Tube Side Relief Valves on RHR Heat Exchangers.

21A3757AD, Rev. 2, GE Purchase Specification Data Sheet for Shell Side Relief Valves on RHR Heat Exchangers.

21A8657, Rev. 3, GE Purchase Specification, General Requirements for Valves.

21A8658, Rev. 1, GE Purchase Specification, General Requirements for Motor Operated Valve Actuators.

21A8706, Rev. 3, GE Purchase Specification, Heat Exchanger Materials for General Electric Design.

21A9222, Rev. 2, GE Purchase Specification, General Requirements for Electric Motors.

21A9222DM, Rev. 5, GE Purchase Specification Data Sheet for Vertically Mounted RHR System Motor.

21A9243, Rev. 0, GE Purchase Specification for Auxiliary Pumps for Boiling Water Reactors.

21A9243DJ, Rev. 3, GE Purchase Specification Data Sheet for RHR Pumps.

21A9347AF, Rev. 1, GE Purchase Specification Data Sheet, General Requirements for Instrumentation and Electric Equipment.

21A9376, Rev. 1, GE Purchase Specification for Flow Orifice Assembly.

21A9388AB, Rev. 0, GE Purchase Specification, Instrument Data Sheet for the RHR System.

21A9425, Rev. 1, GE Purchase Specification for RHR Heat Exchangers.

21A9425AB, Rev. 1, GE Purchase Specification Data Sheet for RHR Heat Exchangers.

22A2707, Rev. 5, GE Design Specification, BWR Plant Requirements for Water Quality.

22A2710A, Rev. 7, GE Design Specification, BWR Plant Requirements for Standby AC Power.

22A2711, Rev. 3, GE Design Specification, BWR Plant Requirements for DC Power.

22A2714AB, Rev. 1, GE Design Specification, BWR Plant Requirements for Ventilating, Cooling and Heating.

22A2750, Rev. 4, GE Design Specification for Inservice Inspection.

22A2750AD, Rev. 1, GE Quality Assurance Data Sheet for Inservice Inspection.

22A2817, Rev. 3, GE System Design Specification for the RHR System.

22A2817AY, Rev. 0. GE System Design Data Sheet for the RHR System.

22A2988, Rev. 6, GE BWR Plant Requirements, Separation of Electric Equipment for Engineered Safeguard Systems (see also 22A7416 Rev. 0).

22A3007, Rev. 1, GE System Design Specification, Testability Criterion for Instrumentation and Controls in Engineered Safeguard System.

22A3008, Rev. 5, GE Design Specification, Environmental Interface Data for BWR Equipment.

22A3038, Rev. 6, GE Design Specification, Data Listing for Electric Motors to be Supplied by the APED of GE (see also 21A9222).

22A3039, Rev. 1, GE System Design Specification for Process Instrumentation.

22A3061, Rev. 0, GE System Design Specification, Electrical Codes and Standards.

22A3062, Rev. 2, GE System Design Specification, Mechanical Codes.

22A3067, Rev. 3, GE System Design Specification for Mechanical Equipment Separation.

22A3085, Rev. 3, GE Design Specification for the Remote Shutdown System.

22A3095, Rev. 0, GE System Design Specification, Pressure Integrity of Piping and Equipment Pressure Parts.

22A3095AD, Rev. 1, GE System Design Data Sheet, Pressure Integrity of Piping and Equipment Pressure Parts.

22A3730, Rev. 0, GE System Design Specification for RHR Heat Exchangers.

22A3730AB, Rev. 0, GE System Design Data Sheet for RHR Heat Exchangers.

22A3746, Rev. 1, GE System Design Specification for Local Instrument Panels.

22A5233, Rev. 0, GE Installation Specification for RHR Heat Exchangers.

22A5267, Rev. 1, GE System Specification on Regulatory Requirements, Industrial Standards and Design Bases.

22A7416, Rev. 0, GE BWR Requirements for Separation of Electrical Equipment in Engineered Safeguard Systems (see also 22A2988 Rev. 6).

234A9407TC, Rev. 4, GE Instrument Data Sheets for the RHR System.

249A1401TC, Rev. 1, GE Instrument Data Sheets for the Remote Shutdown System.

B&R Engineering Criteria Document:

Section D Electrical Engineering Criteria
Section E Mechanical Engineering Criteria
Section F Chemical and Nuclear Engineering Criteria
Section G Instrumentation and Control Engineering Criteria
Section H Technical Standards Applicability List
Section I Piping and Pipe Support Criteria

C.2 B&R Design Calculations

2.02.02, Rev. 1, Main Plant Oneline Auxiliary Load Calculations.

2.02.07, Rev. 2, Motor Control Center Load Calculations.

2.02.18, Rev. 0, 480 V Switchgear Load Study.

2.03.02, Rev. 5, Main Oneline Short Circuit Calculation.

2.03.07, Rev. 2, 480 V Switchgear Short Circuit Calculation.

2.03.09, Rev. 0, Motor Control Center Short Circuit Calculations.

2.03.11, Rev. 0, Fault Calculations for Paralleling DG1 and DG2.

2.06.03, Rev. 5, Main Oneline Voltage Drop Calculations.

2.06.05, Rev. 3, Reactor Building Feeder Voltage Drop Calculation.

2.06.10, Rev. 1, Service and Diesel Generator Building Feeder Voltage Drop Calculations.

2.06.17, Rev. 0, 4.16 KV and 6.9 KV Motor Feeder Cable Voltage Drop Calculation.

2.07.01, Rev. 2, High Voltage Cable Sizing, Ampacities and Conduits.

2.07.05, Rev. 0, 4.16 KV and 6.9 KV Cable Sizing, Short Circuit Capacity.

2.07.09, Rev. 3, 125 V DC System Cable Sizing for Circuit Breakers.

2.07.10, Rev. 0, DC System Cable Sizing for Voltage Drop.

2.12.00, Rev. 5, Relay Setting Time Current Characteristic Curves.

2.12.14, Rev. 1, 4.16 KV Switchgear Relay Settings.

5.17.13, Rev. 0, Flow Restrictor Sizing, RHR System.

5.17.19, Rev. 1, RHR System Pressure Drop Calculations.

5.17.20, Rev. 0, Effectiveness Calculation for RHR Heat Exchanger.

5.17.26, Rev. 0, RHR Testline Orifice Sizing Calculation.

5.17.29, Rev. 1, RHR LPCI Line Orifice Sizing Calculation.

6.19.19, Rev. 0, Pages 38 through 56A, Structural Calculation, Reactor Building, Interior Walls at Elevation 572.0 ft.

6.19.34, Rev. 2, Sheets 1 through 9 (Pages 62 to 70C), Structural Calculation, Reactor Building, Equipment Foundations.

7.00.55, Rev. 3, Minimum Flow Control Valve Sizing Calculations.

8.14.127B, Rev. 6, Structural Design Calculation, Anchor Group 36.

8.15.213, Rev. 4, Review/Redesign Calculation for Piping Support RHR-HGR-184 (PS-1, Y), Node 163.

8.15.2341, Rev. 2, Review/Redesign Calculation for Piping Support RHR-HGR-436 (PS-6, Y) Node 1220.

9.21.02, Rev. 0, Reactor Building, Emergency Cooling and Critical Area Cooling System.

9.32.00, Rev. 3, HVAC for Control Room, Cable Spreading Room and Critical Switchgear Room.

C.3 Technical Memoranda

TM 151, Rev. 0, B&R Technical Memorandum, RHR Heat Exchanger Leak Investigation.

TM 181, Rev. 0, B&R Technical Memorandum, Shielding Requirements for the RHR System.

TM 194, Rev. 1, B&R Technical Memorandum, RHR Heat Exchanger Leak Investigation.

TM 327, Rev. 0, B&R Technical Memorandum, Shielding Requirements for the RHR Heat Exchanger Rooms.

TM 420, Rev. 4, B&R Technical Memorandum, Electric Cable - Listing of Outside Diameter, Weight, Pulling Tension and Bending Radius.

TM 526, Rev. A, B&R Technical Memorandum, System Description for the RHR System.

TM 563, Rev. 0, B&R Technical Memorandum, RHR Heat Exchanger Leakage Investigation.

TM 610, Rev. 0, B&R Technical Memorandum, RHR System Relief Valve Sizing.

TM 1000, Rev. 0, B&R Technical Memorandum, Actuation of RHR Heat Exchanger Relief Valves.

TM 1016, Rev. 0, B&R Technical Memorandum, Cavitation in the RHR System.

TM 1060, Rev. 3, B&R Technical Memorandum, Voltage Droop Study.

TM 1129, Rev. 0, B&R Technical Memorandum, Class 1E Motor Operated Valves.

TM 1131, Rev. 0, B&R Technical Memorandum, Design Changes for Line RHR [illegible] (16).

TM 1232, Rev. 0, B&R Technical Memorandum, Service Water Requirements.

C.4 Vendor Manuals

GEK-71330, July 1978, Operation and Maintenance Instructions for the Remote Shutdown System.

GEK-71336, July 1978, Operation and Maintenance Instructions for the Residual Heat Removal System.

GEK-71337, June 1978, Operation and Maintenance Instructions for Vendor Supplied Instruments

CVI 47A-00, 131, Issue 1, Operation and Maintenance Instructions plus Parts Catalog for Medium Voltage Metal Clad Switchgear.

CVI 49-00, 25, Issue 1, ITE Instruction Manual for Motor Control Centers.

CVI 2-02E12-08, Sheet 10, Issue 1, Operation and Maintenance Manual for RHR Pumps (Ingersoll Rand).

C.5 Drawings

C.5.1 Mechanical and Nuclear

M-151, Rev. 0, B&R General Arrangement Drawing, Ground Floor (Elevation 441.0 ft).

M-152, Rev. 0, B&R General Arrangement Drawing, Mezzanine Floor (471.0 ft).

M-153, Rev. 0, B&R General Arrangement Drawing, Operating Floor (501.0 ft).

M-154, Rev. 0, B&R General Arrangement Drawing, Reactor Duilding Floor Plans at 422.25 ft, 510.5 ft, 522.0 ft, 548.0 ft, 572.0 ft, and 606.88 ft.

M-155, Rev. 0, B&R General Arrangement Drawing, Reactor Building Vertical Sections.

M-159, Rev. 0, B&R Equipment List for General Arrangement Drawings.

M-501, Rev. 21, B&R Chart of Flow Diagram Symbols.

M-521, Sheet 1 and 2, Rev. 39, B&R Flow Diagram of the Residual Heat Removal System.

M-524, Sheet 1 and 2, Rev. 37, B&R Flow Diagram of the Standby Service Water System.

197R567, Rev. 3, GE Piping and Instrument Symbols.

731E961AD, Sheet 1 and 2, Rev. 4, GE Piping and Instrumentation Diagram for the RHR System.

731E966, Rev. 6, GE Process Diagram for the Residual Heat Removal System.

731E966AD, Sheet 1, Rev. 2, Sheet 2, Rev. 0, GE Process Data Sheet for the Residual Heat Removal System.

762E481, Rev. 5, GE Assembly Drawing, RHR Heat Exchanger.

762E483, Rev. 3, GE Drawing of RHR Heat Exchanger Channel.

762E484, Rev. 4, GE Drawing of RHR Heat Exchanger Tube Bundle.

762E485, Rev. 3, GE Drawing of RHR Heat Exchanger Tube Sheet.

921D280, Rev. 0, GE Instrument Symbols.

105D4981, Rev. 2, GE Drawing of RHR Heat Exchanger Channel Cover.

105D4984, Rev. 3, GE Drawing of RHR Heat Exchanger Baffle Plate.

137C7572, Rev. 0, GE Installation Drawing for RHR Heat Exchanger Relief Valve.

M-200, Sheet 106, Rev. 5, B&R Isometric Diagram with RHR-V-4B, RHR-V-6A, RHR-P-2B Suction.

M-200, Sheet 107, Rev. 5, B&R Isometric Diagram with RHR-HX-1B Inlet, RHR-V-47B, RHR-V-48B, RHR-V-89, RHR-V-116, RHR-V-115, RHR-FCV-64B, RHR-RO-1B, RHR-V-18B.

M-200, Sheet 112, Rev. 4, B&R Isometric Diagram with RHR-FE-14B, RHE-FIS-10B, RHR-V-3B.

M-200, Sheet 113, Rev. 4, B&R Isometric Diagram with RHR-V-42B, RHR-V-53B, RHR-V-17B, RHR-V-16B.

M-200, Sheet 150, Rev. 7, B&R Isometric Diagram with RHR-V-27B, RHR-V-24B, RHR-V-172B, RHR-RO-3B.

M-701, Rev. 24, Reactor Building Layout at Elevation 422.25 ft..

M-702, Rev. 21, Reactor Building Layout at Elevations 441.0 ft and 444.0 ft.

M-703, Rev. 18, Reactor Building Layout at Elevation 471.0 ft.

M-704, Rev. 22, Reactor Building Layout at Elevation 501.0 ft.

M-705, Rev. 23, Reactor Building Layout at Elevation 522.0 ft.

M-706, Rev. 32, Reactor Building Layout at Elevation 548.0 ft.

M-707, Rev. 15, Reactor Building Layout at Elevation 572.0 ft.

M-708, Rev. 28, Reactor Building Layout Details at Various Elevations and Vertical Sections.

M-709, Rev. 31, Reactor Building, Vertical Sections.

C.5.2 Instrumentation and Control

197R567, Rev. 3, GE Piping and Instrument Symbols.

731E961AD, Rev. 4, 2 Sheets, GE Piping and Instrumentation on Diagram for the RHR System.

731E966, Rev. 6, GE Process Diagram for the RHR System.

731E999, Rev. 5, GE Functional Control Diagram for the RHR System.

762E280AD Rev. 0, GE Functional Control Diagram for the Remote Shutdown Panel.

807E170TC, Rev. 14, GE Elementary Diagram for the RHR System.

807E151TC, Rev. 10, GE Elementary Diagram for the Remote Shutdown Panel.

105D4947AD, Rev. 1, GE IED for the Remote Shutdown Panel.

127D1812TC, Rev. 3, GE Tubing Diagram for Rack H22-P021.

127D1841TC, Rev. 3, GE Arrangement Drawing for Rack H22-P021.

828E191TC, Rev. 9, GE Connection Diagram for Rack H13-P618.

828E289TC, Rev. 5, GE Connection Diagram for Rack H22-P021.

828E466TC, Rev. 8, GE Arrangement Drawing for Remote Shutdown Panel.

828E482TC, Rev. 8, GE Connection Diagram for Remote Shutdown Panel.

921D280, Rev. 0, GE Instrument Symbols.

145C3008, Rev. 8, GE Differential Pressure Switch Diagram, Purchased Part.

145C3011, Rev. 8, GE Diagram for Differential Pressure Switch.

159C4540, Rev. 6, GE Diagram for Meter, Model 180.

163C1183 Rev. 5,, GE Diagram for Differential Pressure Transmitter.

9E003, Rev. 2, B&R Electric Wiring Diagram for RHR-P-2B.

9E004, Rev. 1, B&R Electric Wiring Diagram for RHR-P-2B.

9E010, Rev. 1, B&R Electric Wiring Diagram for RHR-P-3.

9E017, Rev. 2, B&R Electric Wiring Diagram for RHR-V-3B.

9E019, Rev. 2, B&R Electric Wiring Diagram for RHR-V-4B.

9E022, Rev. 2, B&R Electric Wiring Diagram for RHR-V-6B.

9E034, Rev. 1, B&R Electric Wiring Diagram for RHR-V-24B.

9E038, Rev. 1, B&R Electric Wiring Diagram for RHR-V-27B.

9E047, Rev. 1, B&R Electric Wiring Diagram for RHR-V-47B.

9E049, Rev. 2, B&R Electric Wiring Diagram for RHR-V-48B.

9E057, Rev. 1, B&R Electric Wiring Diagram for RHR-FCV-64B.

E-522, Rev. 16, B&R Elementary Diagram for Isolation Valve Status Display Panel.

E537, Sheet 6C, Rev. 11, B&R Connection Wiring Diagram for Control Boards.

E539, Sheet 16, Rev. 10, B&R Connection Wiring Diagram for RHR System.

E539, Sheet 20, Rev. 8, B&R Connection Wiring Diagram for RHR System.

E-697, Rev. 32, I&C Conduit and Tray Diagram at Elevation 501.0 ft.

M-153, Rev. 0, B&R General Arrangement Drawing, Operating Floor (501.0 ft).

M-154, Rev. 0, B&R General Arrangement Drawing, Reactor Building Floor Plans at 422.3 ft, 510.5 ft, 522.0 ft, 548.0 ft, 572.0 ft, 606.9 ft.

M-155, Rev. 0, B&R General Arrangement Drawing, Reactor Building Vertical Sections.

M-159, Rev. 0, B&R Equipment List for General Arrangement Drawings.

M-501, Rev. 21, B&R Chart of Flow Diagram Symbols.

M-521, Sheet 1 and 2, Rev. 39, B&R Flow Diagram of the RHR System.

M-568, Rev. 23, B&R Radiation Zone Drawing for Reactor Building at Elevations 471.0 ft and 501.0 ft.

M-706, Rev. 33, B&R Piping Plan, Reactor Building at Elevation 548.0

M-735, Rev. 26, B&R Piping Plan, Reactor Building at Elevation 501.0 ft.

M-807, Rev. 18, B&R HVAC Plans, Reactor Building at Elevation 501.0 ft.

M200, Sheet 107, Rev. 5, B&R Piping Diagram, Contract 215.

M200, Sheet 112, Rev. 4, B&R Piping Diagram, Contract 215.

M619, Sheet 15, Rev. 7, B&R Tubing Connection Diagram, Contract 220.

M619, Sheet 16, Rev. 7, B&R Tubing Connection Diagram, Contract 220.

D-220-0090-H22-P021, Rev. 1, JCI Diagram.

D-220-3500-5.0-RHR-FT-1, Rev. 1, JCI Diagram.

E-220-5500-RB-501, Rev. 5, JCI Drawing.

52A8654, Rev. D, Fisher Controls Drawing of Limitorque Actuated Control Valve.

C.5.3 Electrical

9E003, Rev. 2, B&R Electric Wiring Diagram for Pump RHR-P-2B.

9E017, Rev. 1, B&R Electric Wiring Diagram for RHR-V-3B.

9E034, Rev. 1, B&R Electric Wiring Diagram for RHR-V-24B.

9E057, Rev. 1, B&R Electric Wiring Diagram for RHR-FCV-64B.

E501, Rev. 9, B&R Electrical Symbol List.

E502, Sheet 2, Rev. 19, B&R Main Oneline Diagram, Emergency Buses.

E503, Sheet 7, Rev. 25, B&R Auxiliary Oneline Diagram, Motor Control Centers.

E503, Sheet 8, Rev. 23, B&R Auxiliary Oneline Diagram, Motor Control Centers.

E503, Sheet 12, Rev. 23, B&R Auxiliary Oneline Diagram, Motor Control Centers.

E514, Sheet 8, Rev. 2, B&R Diagram, Relay Settings for 4.16 KV Switchgear, SM-8.

E517, Sheet 3, Rev. 12, B&R Elementary Diagram for 4.16 KV Switchgear.

E517, Sheet 4, Rev. 8, B&R Elementary Diagram for 4.16 KV Switchgear.

E517, Sheet 9, Rev. 17, B&R Elementary Diagram for 4.16 KV Switchgear.

E517, Sheet 10, Rev. 13, B&R Elementary Diagram for 4.16 KV Switchgear.

E517, Sheet 13, Rev. 9, B&R Elementary Diagram for 4.16 KV Switchgear.

E517, Sheet 18, Rev. 1, B&R Elementary Diagram for 4.16 KV Switchgear.

E518, Sheet 6, Rev. 11, B&R Elementary Diagram for 480V Switchgear.

E519, Sheet 1A, Rev. 4, B&R Elementary Diagram for Valve Control.

E528, Sheet 25, Rev. 1, B&R MCC Equipment Overload Summary for MCC-MC-7B-A.

E528, Sheet 35, Rev. 3, B&R Overload Summary for MCC-MC-8B.

E528, Sheet 36, Rev. 1, B&R Overload Summary for MCC-MC-8B-A.

E528, Sheet 37, Rev. 0, B&R Overload Summary for MCC-MC-8B-B.

E533-21VH-5, Rev. 2, Bill of Material for Electrical Devices, 4.16 KV Switchgear, SM-8.

E550, Rev. 35, Power Cable Schedule.

E551, Rev. 38, Control Cable Schedule.

E558, Sheet 2, Rev. 4, Turbine Generator Building, Grounding Plans and Details.

E680, Rev. 18, Reactor Building at Elevation 422.25 ft, Power Conduit and Tray Plan.

E681, Rev. 11, Reactor Building at Elevation 441.0 ft, Power Conduit and Tray Plan.

E682, Rev. 35, Reactor Building at Elevation 471.0 ft, Power Conduit and Tray Plan.

E684, Rev. 31, Reactor Building at Elevation 522.0 ft, Power Conduit and Tray Plan.

E685, Rev. 20, Reactor Building at Elevation 548.0 ft, Power Conduit and Tray Plan.

E686, Rev. 25, Reactor Building at Elevation 572.0 ft, Power Conduit and Tray Plan.

E745, Sheet 1, Rev. 18, Radwaste and Control Building at Elevation 437.0 ft, Power Conduit and Tray Plan.

E747, Sheet 1, Rev. 37, Radwaste and Control Building at Elevation 467.0 ft, Power Conduit and Tray Plan.

E915, Rev. 12, Reactor Building at Elevation 422.25 ft, Location Plan for Cable Tray Nodes.

E916, Rev. 5, Reactor Building at Elevation 441.0 ft, Location Plan for Cable Tray Nodes.

E917, Rev. 10, Reactor Building at Elevation 471.0 ft, Location Plan for Cable Tray Nodes.

E191, Rev. 7, Reactor Building at Elevation 522.0 ft, Location Plan for Cable Tray Nodes.

E922, Sheet 2, Rev. 6, Reactor Building Sections, Location Plan for Cable Tray Nodes.

E922, Sheet 4, Rev. 7, Reactor Building Sections, Location Plan for Cable Tray Nodes.

E927, Sheet 1, Rev. 10, Radwaste and Control Building at Elevation 437.0 ft, Location Plan for Cable Tray Nodes.

E929, Rev. 9, Radwaste and Control Building at Elevation 467.0ft, Location Plan for Cable Tray Nodes.

E934, Sheet 2, Rev. 10, Cable Spreading Room in Radwaste and Control Building, Location Plan for Cable Tray Nodes.

E935, Sheet 4, Rev. 7, Section 4-4 of Radwaste and Control Building, Location Plan for Cable Tray Nodes.

M-521, Sheet 1 and 2, Rev. 39, B&R Flow Diagram of the Residual Heat Removal System.

922C302FD, Rev. 6, Outline for Induction Motor RHR-M-2B.

C.5.4 Structural

M-151, Rev. 0, B&R General Arrangement Drawing at Elevation 441.0 ft (Ground Floor).

M-152, Rev. 0, B&R General Arrangement Drawing at Elevation 471.0 ft (Mezza Floor).

M-153, Rev. 0, B&R General Arrangement Drawing at Elevation 501.0 ft (Operating Floor).

M-154, Rev. 0, B&R Reactor Building Floor Plans at Elevations 422.25 ft, 510.5 ft, 522.0 ft, 548.0 ft, 572.0 ft, 606.88 ft.

M-155, Rev. 0, B&R General Arrangement Drawing, Reactor Building Vertical Sections.

M-159, Rev. 0, B&R Equipment List for General Arrangement Drawing.

M-501, Rev. 21, B&R Chart of Flow Diagram Symbols.

M-521, Sheet 1, Rev. 39, B&R Flow Diagram of the Residual Heat Removal System.

M-521, Sheet 2, Rev. 39, B&R Flow Diagram of the Residual Heat Removal System.

H-501, Sheets 1, 2, 3, all Rev. 0, B&R Construction Tolerances, Piping and Pipe Supports.

S-660, Rev. 28, B&R Drawing, Structural Anchor Bolt Schedule.

S-722, Rev.16, B&R Drawing, Reactor Building Details at Elevation 572.0 ft.

S-769, Rev. 7, B&R Drawing, Reactor Building Details.

S-772, Rev. 40, B&R Drawing, Reactor Building Equipment Foundations Sheet 2.

S-794, Sheet 1, Rev. 24, Structural Drawing of Primary Containment.

S-1000, Sheet 1, Rev. 19, List of Reactor Building Piping Restraints.

S-1062, Rev. 5, Load Table for Piping Supports in Primary Containment.

761E428, Rev. 2, GE Drawing, Residual Heat Removal System.

762E481, Rev. 5, GE Drawing of the RHR Heat Exchanger.

762E484, Rev. 4, GE Drawing of Tube Bundle for RHR Heat Exchanger.

762E485, Rev. 3, GE Drawing of Tube Sheet for RHR Heat Exchanger.

105D4984, Rev. 3, GE Drawing of Baffle Plate for RHR Heat Exchanger.

M-200, Sheet 107, Rev. 5, B&R Isometric Diagram with RHR-HX-1B Inlet.

M-200, Sheet 112-1, Rev. 5B, B&R Isometric Diagram with RHR-FE-14B (at Elevation 565.5 ft).

M-200, Sheet 112-2, Rev. A, Data Sheet for M-200 Sheet 112-1.

M-200, Sheet 150, Rev. 7A, B&R Isometric Diagram with RHR-V-24B.

M-701, Rev. 19, B&R Drawing, Reactor Building Floor Plans at Elevation 422.25 ft, Vertical Sections..

M-702, Rev. 21, B&R Drawing, Reactor Building Layout at Elevations 441.0 ft and 444.0 ft.

M-703, Rev. 18, B&R Drawing, Reactor Building Layout at Elevation 471.0 ft.

M-704, Rev. 22, B&R Drawing, Reactor Building Layout at Elevation 501.0 ft.

M-705, Rev. 23, B&R Drawing, Reactor Building Layout at Elevation 522.0 ft.

M-706, Rev. 32, B&R Drawing, Reactor Building Layout at Elevation 548.0 ft.

M-707, Rev. 15, B&R Drawing, Reactor Building Layout at Elevation 572.0 ft.

M-708, Rev. 21, B&R Drawing, Various Reactor Building Sections and Details.

RHR-184 S0068, Sheet 1OF4, Rev. 4, B&R Drawing of Piping Support RHR-HGR-184.

C.6 Memoranda (General)

EM-79-006, B&R Engineering Memorandum, MCC Master List, January 2, 1979.

EM-79-238, B&R Engineering Memorandum, MCC Master List Revisions, March 22, 1979.

GEBR-2-81-182, GE Letter to B&R, on Increased Loads.

GEBR-2-81-189, GE Letter to B&R, on Increased Loads.

C.7 Contract Specifications

Contract 2, Division 2, Section 2A, Nuclear Steam Supply System.

Contract 41A, Division 15, Section 15A, Nuclear Valves.

Contract 41B, Division 15, Section 15A, Nuclear Valves.

Contract 42, Fisher Controls, Incd..

Contract 42A, Division 15, Section 15B, Control Valves, Quality Class I.

Contract 47A, Division 16, Section 16A, Metal Clad Switchgear.

Contract 49, Division 16, Section 16A, Motor Control Centers.

Contract 62A, Division 16, Section 16A, Electrical Cable.

Contract 62B, Division 16, Section 16A, Electrical Cable.

Contract 215, Division 15, Section 15B, Piping Systems, Section 15F, Valves, Section 15G, Specialties.

Contract 220, Johnson Controls, Incd.

C.8 Other Documentation Utilized for Investigation

C.8.1 Test Data

2993-112-1, Rev. 0, Ingersoll Rand Pump Test Data, Curve N-621, Pump Serial Number 047-3111, dated December 26, 1974.

2993-117-1, Rev. 1, Ingersoll Rand Pump Test Data, Curve N-155, Speed-Torque Characteristic of Centrifugal Pump, Start With Open Discharge.

2997-24, Rev. 1, Curve 388-AA-578, Speed-Torque-Current Curves for Induction Motor, RHR Pump Motors for Hanford II, B&R File 41A-00-0073 Rev. 3, Limitorque Corporation, Master Certification Sheet 1.

B&R 41B-00-0108, Limitorque Motor Data.

WPPSS QA EEI-02-KNC-80-022, Test Report, Limitorque Valve Actuator Qualification for Nuclear Power Station Services, Report B0058, Test per IEEE Standards 382-1972, 323-1974, 344-1975, by Limitorque Corporation, dated January 11, 1980.

Cable Pull Slips for Cables 2SM8-50 and 2M8BA-20.

B&R 41A-00-8496, Motor Test Report for RHR-MO-3B.

QA Film 02-003-1254, Anchor Valve Company, Certified Operation Test Report for RHR-MO-24B.

QA Film 02-009-322, Report of Test Certification for RHR-MO-64B.

QA Film 02-009-323, Fisher Conrol Company, Manufacturer Certification for RHR-MO-64B.

WPPSS SLT EDS-8, System Lineup Test for RHR-P-2B, 4 Pages, dated May 28, 1981 and October 19, 1981.

WPPSS SLT EDS-1, System Lineup Test for RHR-P-2B, dated January 8, 1982.

C.8.2 Standards and Regulatory Guides

IEEE-141-1969, The Red Book, Recommended Practice for Electric Power Distribution in Industrial Plants.

IEEE-279-1971, Criteria for Protection Systems in Nuclear Power Generating Stations.

IEEE-308-1974, Criteria for Class 1E Power Systems in Nuclear Power Generating Stations.

IEEE-323-1971, Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE-323-1974, Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE-382-1972, Type Test of Class 1E Electric Valve Actuators for Nuclear Power Generating Stations.

IEEE-383-1974, Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations.

NEMA-MG-1-1978, Motor and Generator Standards.

NEMA-ICS-1-108, Service and Installation Conditions.

NEMA-ICS-2-321, AC General Purpose Class A Magnetic Controllers for Induction Motors, Rated in Horsepower, 600 V and less, 50 and 60 Hz.

NEMA-ICS-2-322, AC General Purpose Motor Control Centers.

NEMA-ICS-327, AC General Purpose Class A Magnetic Controllers for Induction Motors, Rated in Full Load and Locked Rotor Current, 600 V and less, 50 and 60 Hz.

IPCEA-S-68-516, Interim Publication 2, Cables Rated 5.0 KV and Less, Having Ozone Resistant Ethylene-Propylene-Rubber Integral Insulation and Jacket.

IPCEA-P-54-440, 2nd Ed., Ampacities of Cables in Open Cable Trays.

NFPA-70-1981, National Electrical Code.

ANSI-C37.04-1979, Rating Structure for AC High Voltage Circuit Breakers, Rated on a Symmetrical Current Basis.

ANSI-C37.06-1979, Preferred Ratings and Related Required Capabilities for AC High Voltage Circuit Breakers, Rated on a Symmetrical Current Basis.

ANSI-C37.010-1972, Application Guide for AC High Voltage Circuit Breakers, Rated on a Symmetrical Current Basis.

ANSI-C37.010-1979, Application Guide for AC High Voltage Circuit Breakers, Rated on a Symmetrical Current Basis.

RG-1.131, Regulatory Guide, Qualification Tests of Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations.

C.8.3 Miscellaneous Other Documentation

WPPSS WNP-2 Class 1E Equipment List, dated March 2, 1983 and January 4, 1983.

GE ESM Book 3, GE Electrical Equipment Specification Manual, Application Guide for Systems and Utilization Equipment.

WX-AD-32-262, Westinghouse Application Data 32-262 for Type DHF Circuit Breakers.

PPM 10.25.13, WNP-2 Plant Procedure Manual, Electrical Maintenance Programs and Procedures, Westinghouse High Voltage Circuit Breakers.

238X184AD, Rev. 7, Parts List for Residual Heat Removal System.

SECTION D RFW SYSTEM REVIEW REFERENCES

D.1 Mechanical References

D.1.1 Design Specifications

General Electric Design Specifications

22A719, Rev. 0, Feedwater Flow Measurement and Control.

22A2800, Rev. 1, Rated Steam Output Curve.

22A2801, Rev. 1, Reactor System Heat Balance - Rated.

22A2802, Rev. 1, Reactor System Heat Balance - 105% of Rated.

22A2887, Rev. 6, Nuclear Boiler System.

22A3007, Rev. 5, BWR Equipment Environmental Interface Data.

22A3067, Rev. 3, Mechanical Equipment Separation.

22A3095AD, Rev. 1, Pressure Integrity of Piping and Equipment, Press. Parts.

22A2907, Rev. 3, FW Control System (Steam Turbine Driven RFW Pumps).

22A2907AB, Rev. 1, Feedwater Control System.

Burns and Roe Engineering Criteria Document

Section E - Mechanical Engineering Criteria

Section F - Nuclear Power Engineering Design Criteria

Section G - Instrumentation and Control Criteria

Section I - Process Piping and Pipe Supports

Westinghouse Thermal Performance Data

Heat Balances.

AB095-1554-1205849 KW, Maximum Calculated, Not Guaranteed.

AB095-1555-1154745 KW, Maximum Guaranteed.

Industry Standards

Heat Exchanger Institute Std. for Closed FW Htrs, 1st Ed., 1968.

American Petroleum Institute Std. RP-520.

D.1.2 Calculations

4.20.04 - Feedwater System - From Reactor Feed Pumps to the Reactor Vessel, 11-16-76.

4.25.01 - Reactor Feedwater System Pressure Drop Calc. , 3-13-78.

5.07.72 - Pressurization of M.S. Tunnel From an M.S. Line Break, 5-13-79.

5.07.73 - Pressurization of M.S. Tunnel From an F.W. Line Break, 8-14-79.

7.00.50, Sht. 5 - RFW-V-115A, B Flow Control Valve Sizing 5-18-72.

Sht. 6 - COND-V-149, Control Valve Sizing, 1-25-72.

Sht. 6A - RFW-FCV-15, Control Valve Sizing, 3-11-83.

Sht. 5A, Rev. 1 - RFW Resizing of RFW-PCU-15, 3-22-83.

D.1.3 Technical Memorandums

TM 667 - Feedwater Delivery System 6-26-74.

TM 1010 - Operation of Feedwater Delivery System, 4-29-77.

D.1.4 Manuals

Anchor Darling Valve Operation and Maintenance Manual, AVC-198.

Southwest Engineering Manual for Feedwater Heaters.

Velan Valve Instruction Manual.

Ingersoll-Rand Reactor Feedwater Pump Manual.

Delaval Reactor Feedpump Turbine Drive Instruction Book.

D.1.5 Drawings

Burns and Roe

M504, Rev. 40, Condensate and Reactor Feedwater Flow Diagram.

M506, Rev. 40, Misc. Drains, Vents and Sealing Systems.

M529, Rev. 35, Nuclear Boiler, Main Steam Flow Diagram.

M645, Rev. 15, RFW and Cond. Piping Sections.

M200-27, Rev. 6, FW Piping In Containment: Line A.

M200-28, Rev. 5, FW Piping In Containment: Line B.

M200-334, Rev. 6, FW Piping, RFW Pumps to #6 Htr and Condenser.

M200-335, Rev. 7, FW Piping, RFW Pumps to Reactor.

M200-341, Rev. 3, Cond., L.P. Htrs 5A and 5B to RFW Pumps.

Bovee and Crail Isometrics

COND-385-1.4, Rev. 6, Seal Water to RFW Pumps 1A and 1B
-385-5.6, Rev. 3, Seal Water to RFW Pumps 1A and 1B

RFW-413-1.5, Rev. 10, From FW Pump 1A to Condenser
-6.8, Rev. 6, From FW Pump 1A to Condenser
-414-1.5, Rev. 10, FW Pump 1B to Condenser
-6.8, Rev. 6, FW Pump 1B to Condenser
-415-1.5, Rev. 7, Recirc. Line, HP Htrs. to Condenser
-6.7, Rev. 5, Recirc. Line, HP Htrs. to Condenser
-8.10, Rev. 6, Recirc. Line, HP Htrs. to Condenser
-11.12, Rev. 7, Recirc. Line, HP Htrs. to Condenser
-13.14, Rev. 6, Recirc. Line, HP Htrs. to Condenser
-416-1.5, Rev. 5, From FW Pump 1A and 1B to HP Htrs. 6A and 6B
-6.9, Rev. 5, FW Pumps 1A and 1B to HP Htrs. 6A and 6B
-10.12, Rev. 9, FW Pumps to HP Htrs. 6A and 6B
-13.14, Rev. 7, FW Pump to HP Htrs. 6A and 6B

Bovee and Crail Isometrics (Cont'd)

-417-1.3, Rev. 5, HP Htr. 6A to Flow Meter
-4.5, Rev. 3, HP Htr. 6A and 6B to Flow Meters
-6.8, Rev. 3, HP Htr. 6A and 6B to Flow Meters
-9.10, Rev. 2, HP Htr. 6A and 6B to Flow Meters
-11.13, Rev. 2, HP Htr. 6A and 6B to Flow Meters
-418-1.2, Rev. 10, Flow Element to Cont. (Line A)
-3, Rev. 4, Flow Element to Cont. (Line A)
-4, Rev. 7, Cont. to Reactor Vessel (Line A)
-5.6, Rev. 5, Cont. to Reactor Vessel (Line A)
-7.8, Rev. 5, Cont. to Reactor Vessel (Line A)
-9.10, Rev. 7, Cont. to Reactor Vessel (Line A)
-11.12, Rev. 6, Cont. to Reactor Vessel (Line A)
-13, Rev. 6, Cont. to Reactor Vessel (Line A)
-419-1.2, Rev. 8, Flow Meter to Cont. (Line B)
-3, Rev. 4, Flow Meter to Cont. (Line B)
-4, Rev. 4, Cont. to Reactor Vessel (Line B)
-5.7, Rev. 7, Cont. to Reactor Vessel (Line B)
-8.9, Rev. 7, Cont. to Reactor Vessel (Line B)
-10.11, Rev. 5, Cont. to Reactor Vessel (Line B)
-12.13, Rev. 7, Cont. to Reactor Vessel (Line B)
-479-1.3, Rev. 2, FW Pump 1B to Hp Htrs. 6A and 6B
-480-1.4, Rev. 4, Bypass Line, RFW Pump Disch. to Hx6A Disch.

Vendor Drawings

CCI Control Valve, Dwg. #921901077, Rev. H.

Anchor Darling Valve Dwg. #3084-3, Rev. A.

Fisher Control Dwg. #52A8558, Rev. C.

I-R Pump Curve Dwg. #49413.

I-R Seal Injection Control Dwg. #2636-C-18C.

I-R CN Pump Dwg. #C-18X17CN500X4B.

I-R CN Pump Parts List, Dwg. #C-18X17CN500X4.

Velan Dwg. #P2-3319-N-33, Rev. J.

D.1.6 Memoranda

WPBR-73-891, Containment Isolation Valves, 12-11-73.

BRWP-74-365, Containment Isolation Valves, 4-10-74.

WPBR-74-460, Containment Isolation Valves, 4-19-74.

EN-RLH-81-05, Containment Iso. and Testability Eval., 10-12-81.

D.1.7 Contract Specifications

Cont. No.	Award Date	Item
2808-10	1-14-72	Feedwater Heaters
2808-11A	2-18-72	Reactor Feed Pumps
2808-41A	12-3-73	Nuclear Valves
2808-41B	12-3-73	Nuclear Valves
2808-42A	5-13-74	Misc. Control Valves, Controllers and Acc.
2808-215	5-13-74	Mechanical Equipment Installation

B&W Equipment Spec. #08-1004-352-00 (RFW-FCV-15)

D.1.8 Reports

Anchor Darling Valve Design Report: 24"-900# Check Valves.

Anchor Darling Material Certification Report for RFW-V-32A.

CCI Material Certification Report (RFW-FCV-15).

Velan Certificate of Compliance (RFW-V-65A).

D.2 Electrical References

D.2.1 Design Specifications

B&R Engineering Criteria Document, Section D, Electrical Engineering Criteria.

B&R Engineering Criteria Document, Appendix 3, Electrical Separation Practices, Rev. 1, 12-22-82.

D.2.2 Calculations

2.02.02 (Main Plant Bus Load Calculations), Rev. 1, DL 6/15/81.

2.02.07 (Motor Control Centers Load Calculations), Rev. 1, DL 10-12-76.

2.03.07 (480 Volt Switchgear Short Circuit Calculations), Rev. 2, DL 1/20/77.

2.03.09, (MCC Short Circuit Calculations), Rev. 0, DL 1/24/78.

2.06.03, (Computer Run) - (Main One Line Voltage Drop Calculations), Rev. 5, DL 1/18/80.

2.06.05 (Reactor Building Feeder and Voltage Drop Calculations), Rev. 3, DL 2/8/77.

2.06.06 (Turbine Generator Building, Feeder and Voltage Drop Calculations), Rev. 1, DL 12/16/74.

2.06.10 (480 Volt MCC Voltage Drop Calculation and Cable Sizing), Rev. 1, DL 4/30/74.

2.12.00 (Relay Setting Time Current Characteristic Curves), Rev. 5, DL 9/15/82.

2.12.12 (480 Volt Switchgear Relay Settings Motor Data), Rev. 1, DL 11/30/76.

D.2.3 Technical Memorandum/Engineering Memo

EM-79-006, Rev. 0, 1/2/79, MCC Master List.

Tech. Memo 1060, Rev. 2, Voltage Drop Study.

B&R Engrg. Memo EM-79-239, Rev. 0, 3/22/79, MCC Master List Revision.

D.2.4 Manuals

ITE Imperial Corporation, Rowan Controller Manual.

Reactor Feed Pump drive Turbine (Delaval), 2808-12.

Limitorque Manual, SMDI-170.

D.2.5 Drawings

The following B&R drawings with revision numbers listed were reviewed:

EWD-72E-001, MOV RFW-V-65A (B22-F065A), Rev. 1, 7/22/82.

EWD-72E-013, MOV RFW-V-109, Rev. 1, 2/3/83.

EWD-72E-015, MOV RFW-V-112A, Rev. 1, 7/22/82.

EWD-72E-037, Turb. RFW-DT-1A Turning Gear RFT-M-TNGA, Rev. 1, 7/22/82.

EWD-72E-039, Turb. RFW-DT-1A Main Oil Pump RFT-M-MOPA, Rev. 2, 8/31/82.

E502-2, Main One Line Diag., Rev. 19, 1/19/83.

E503-1, Aux. One Line Diag., Rev. 15, 3/21/83.

E503-6, Aux. One Line Diag., Rev. 26, 3/22/83.

E515-1, Breaker Setting 480V Swgr. SL-11 to SL-31, Rev. 1, 10/19/81.

E515-3, Breaker Setting 480V Swgr. SL-63 to SL-81, Rev. 2, 2/20/82.

E528-1, MCC Equip. Overload Summary MCC-MC-1A, Rev. 1, 12/17/82.

E528-2, MCC Equip. Overload Summary MCC-MC-1B, Rev. 2, 11/17/82.

E535-3A, Connection Wiring Diag. Motor Control Center, Rev. 9, 12/07/82.

E535-3B, Connection Wiring Diag. Motor Control Center, Rev. 10, 2/1/83.

E535-10A, Connection Wiring Diag. Motor Control Center, Rev. 11, 4/13/82.

E535-10B, Connection Wiring Diag. Motor Control Center, Rev. 13, 2/1/83.

E528-27, MCC Equip. Overload Summary MCC-MC-7C, Rev. 0, 12/17/82.

E537-19A, Connection Wiring Diag. Control Room Term. Cabinet, Rev. 6, 4/4/83.

E550, Cable Schedule - Power, Rev. 34, 12/7/82.

E558-2, Turb. Gen. Bldg. Grounding Plans and Details, Rev. 4, 4/12/82.

E902-3, Turb. Gen. Bldg. Grnd. Fl. El. 441'-0" Location Plan Cable Tray Nodes, Rev. 1, 7/16/75.

E918, Reactor Bldg. El. 501'-0" Location Plan Cable Tray Nodes, Rev. 11, 4/6/83.

E929, Radwaste and Control Bldg. El. 467'-0" Location Plan Cable Tray Nodes, Rev. 10, 4/6/83.

E933, Radwaste and Control Bldg. Misc. Elev's. Location Plan Cable Tray Nodes, Rev. 4, 4/6/83.

E935-4, Radwaste and Control Bldg. - Section "4-4" Locations Cable Tray Nodes, Rev. 8, 4/6/83.

Other Vendor Drawings Reviewed

B&R File No. 4900 0001, ITE Imperial Corp., MCC Layout for MCC-MC-1B.

B&R File No. 4900 0035, ITE Imperial Corp., MCC Layout for MCC-MC-7C.

B&R File No. 1200 0003, Console Oil Diagram (Delaval Turbine, Inc.).

B&R File No. 41A-00-0073, Limitorque Corp.

B&R File No. 43-00-0061, Walworth Co.

B&R File No. 43-00-0112, Walworth Co.

GE Motor for Turning Gear, DD-17271.

D.2.6 Memoranda

Included in Section D.2.3

D.2.7 Contract Specifications: (B&R)

i) Contract Specification 2808-12, Reactor Feed Pump Turbine - Bid Issue, BD-24.

ii) Contract Specification 2808-41, Nuclear Valves, Division 15, Section 15A.

iii) Contract Specification 2808-43, Standard Cast or Forged Steel Valves, Division 15, Section 15A.

iv) Contract Specification 2808-49, Motor Control Centers, Division 16, Section 16A.

v) Contract Specification 2808-62A and 62B, Electrical Cable.

D.2.8 Others

Vendor Drawings

Veelan Engrg. Co., Test Reports for RFW-MO-65A, (Veelan Order No. P2-3313-N).

Walworth Co., Test Report for RFW-MO-109, RFW-MO-112A, (Walworth Co., P.O. PP 32500, 5/25/77).

Delaval Certificate of Conformance for RFT-M-MOPA, RFT-M-TNGA.

Bussman Fuse Manufacturing, Part III, Component Protection for Electrical Systems.

Industry Codes and Standards

NEMA MG-1, Para. MG1-1.26 (Totally Enclosed Machine).

NEMA ICS-2-322.21 (Combination Motor Control Unit Ratings).

NEMA ICS-2-321.41 (Short Time Capability).

IPCEA - No. P-54-440, "Ampacities, Cables in Open Top Cable Trays".

NFPA 70-1981, "National Electric Code".

ANSI C37.04-1979 (American National Standard Rating Structure for AC High Voltage Circuit Breakers Rated on a Symmetrical Current Basis).

ANSI C37.010-1979 (American National Standard). IEEE Application Guide for AC High Voltage Circuit Breakers Rated on a Symmetrical Current Basis.

IEEE-279-1971 (Criteria for Protection Systems for Nuclear Power Generating Stations).

IEEE-308-1974 (Criteria for Class 1E Power Systems for Nuclear Power Generating Stations).

IEEE-323-1974 (Qualifying Class 1E Equipment for Nuclear Power Generating Stations).

IEEE-344-1975 (Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations).

IEEE-382-1974 (Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations).

IEEE-383-1974 (Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations.

IEEE-384-1977 (Criteria for Independence of Class 1E Equipment and Circuits).

R-G-1.75, Physical Independence of Electric Systems.

NUREG 0588, Category 2, (Environmental Qualification of Class 1E Equipment).

D.3 Instrumentation and Control References

D.3.1 Specifications (General Electric and Burns and Roe, Inc.)

22A2907, Rev. 3, "Feedwater Control System (Steam Driven Turbine Reactor Feed Pumps", 3/28/74.

22A2907AB, Rev. 1, "Feedwater Control System" Data Sheet, 8/16/71.

22A2719, Rev. 2, "Feedwater Flow Measurement and Control" Design Specification, Dated 7/26/71.

22A2719AB, Rev. 0, "Feedwater Flow Measurement and Control" BWR Plant Requirements Specification, 7/26/71.

732E120AD, "IED - Feedwater Control System, Turbine Feed Pumps", Rev. 3.

807E160TC, "Feedwater System" Elementary Diagram, Sheets 1, Rev. 12; 2, Rev. 12; 3, Rev. 10; 4, Rev. 12; 5, Rev. 8.

807E153TC, "Nuclear Boiler Process Instrumentation System" Elementary Diagram, Sheets: 1, Rev. 13; 1A, Rev. 10; 2, Rev. 11; 3, Rev. 3; 4, Rev. 12.

DL807E160TC, "Device List - System Elementary C34A", (6/15/78).

234A9304TC, "IDS - Feedwater Control System", Dated 7/6/73.

GEK-71337, "Instrumentation Manual for Vendor Supplied Instruments", (Feedwater Control System Device CVI Data), Volumes I, II, III, IV, V and VI.

22A3067, Rev. 3, "Mechanical Equipment Separation" System Design Specification, Dated 8/31/75.

22A7416, Rev. 0, "Electrical Equipment, Separation for Safeguards Systems" Design Specification, Dated 2/19/81.

22A3085, Rev. 3, "Remote Shutdown System" Design Specification, Dated 5/25/79.

22A3007, Rev. 1, "Engineering Safeguards Systems, Criteria for Testability of Instrumentation and Controls", 12/1/71.

22A8658, Rev. 1, "General Requirements for Motor Operated Valve Actuators", Dated 5/17/71.

GEK-71314, "Feedwater Control System, O and M Manual", Dated 9/78.

166B7135A, "Information Document - Feedwater Dynamic Analysis Data", Sheets: 1, Rev. C; 2, Rev. C; 3, Rev. C; 4, Rev. C; 5, Rev. C; 6, Rev. C; 7, Rev. C; 8, Rev. C; 9, Rev. C; 10, Rev. C; 10A, Rev. C; 11, Rev. C; 12, Rev. C; 13, Rev. C; 14, Rev. C; 15, Rev. C; 16, Rev. C; 17, Rev. C; 18, Rev. C

Burns and Roe Engineering Design Criteria, Section F, Table 7.4-3, Equipment Classifications.

22A3039, Rev. 1, "Process Instrumentation", 3/26/73, Design Specification Para. 4.2.2, 4.3.3, Figures 12, 1.8.10, Para. 4.2.4, 4.2.5.

22A3041, Rev. 1, "Essential Components", 3/14/77.

22A3746, Rev. 1, "Local Instrument Panels" Design Specification, 1/21/74.

22A3008, Rev. 5, "BWR Equipment Environmental Interface Data", (4/8/77), Design Specification.

239X241AD, "Feedwater Control System (Turbine Driven Reactor Feed Pumps) - Parts List", Rev. 10, Dated 6/4/80.

234A9301TC, Sheet 22, Rev. 1 (8/1/73), "IDS - Nuclear Boiler System".

22A3181AD, Rev. 0, "Flow Element (Main Steam Restrictor" System Design Specification and Data Sheet (11/13/73).

127D1835TC, Rev. 1 (7/19/73), "Main Steam Flow Instrument Panel A (H22-P015).

21A9387AB, Rev. 0, "IDS - Feedwater Control System - Turbine Drive" (9/17/71), Sheet 5.

21A9430, Rev. 0, "Main Steam Flow Element", (11/4/71).

22A2887AB, Rev. 4, Sheet 4, "Nuclear Boiler System Data Sheet" (1/10/75).

163C1029TC, "Piping Diagram - Main Steam Flow Instrument Panel A (H22-P015), Rev. 2 (7/22/77).

127D1845TC, Rev. 2 (7/22/77), "Connection Diagram - Main Steam Flow Instrument Panel A (H22-P015).

163C1183, Rev. 0, "Differential Pressure Transmitter Detail", 4/4/74.

127D1826TC, Rev. 4, "Arrangement, Reactor Vessel Level and Pressure Instrument Panel A (H22-P004)".

127D1814TC, Rev. 3, "Piping Diagram, Reactor Vessel Level and Pressure Instrument Panel A (H22-P004)".

127D1827TC, Rev. 2, "Electrical Diagram, Reactor Vessel Level and Pressure Instrument Panel A (H22-P004)".

117C-4928, Rev. B, "Feedwater Flow Meter Section - Purchased Part" (Shows C34-N001A, B as a double section in which each section is double flanged (flanged at both ends), dated 2/16/71.

761E443, Rev. 1, "Primary Steam Piping Nuclear Boiler - Purchased Part", Dated 2/8/70 (shows C34-N001A, B Specifications).

131C7598, Sheet 1, Rev. 1, "Flow Meter Section - Feedwater Control System", Dated 6/1/71 (C34-N001A, B specification drawing), shows C34N001A, B as a double section in which the sections are flanged together only. The outer ends are for welding.

21A9414, Rev. 1, "Feedwater Flow Meter Section" - Purchase Specific 1/7/71 (has calibration procedures and materials, etc. specification for C34-N001A and B) entire document.

21A9414AB, Rev. 2, "Feedwater Flow Section" - Purchase Specification Data Sheet, Dated 8/24/73, entire document.

328X154TC, Section A, Rev. 11, "Shipping Group Parts List - Nuclear Boiler Local Instrumentation".

238X178A1, Page 7, Rev. 22, "Nuclear Boiler System - Master Parts List" (shows B22-N041 temp. elements code, equipment and source classifications).

159C4520, Sheet 1, Rev. 6, "Temperature Element - Nuclear Boiler", (Details on B22-N041A or RFW-TE-41A).

159C4520, Sheet 2, Rev. 6, "Temperature Element - Nuclear Boiler", (More B22-N041A details).

22A2887, Rev. 6, "Nuclear Boiler System", 1/29/79, Para. 4.11.3.3, Design Specification.

22A2718, Rev. 5, "Special Wire and Cable", 4/10/74, Para. 2.13.2, 2.13.4 (gives wiring type criteria and lead resistance criteria).

828E185TC, Rev. 4, "Arrangement, Nuclear Steam Supply Shutoff Temperature Recorder VB".

22A3041, Rev. 1, "Essential Components", 3/14/72, Design Specification.

22A8696, Rev. 1, "Seismic Requirements for Essential Class I Instrumentation", 3/7/78.

22A2702A, Rev. 1, "Seismic Design", 1/7/71, Design Specification.

22A3059, Rev. 1, Cleaning of Piping and Equipment", 6/24/75.

248A9393, Rev. 0, "General Use, Controller Assembly Data Sheet".

GE-1, Feedwater Control System "Preoperational Test Instruction" (12/12/77), Rev. 0.

STI-23X, Feedwater Control System Tune-Up Procedure, "Startup Test Instructions" (6/10/81), Rev. 2.

GEZ-6894, "Hanford 2 Nuclear Power Station Control Systems Design Report", R. W. Polomik, S. T. Chow (2/80), Chapter 7.

22A4152, Rev. B, "Startup Test Program", Sht. 53 (shows Feedwater System Control response performance criteria).

22A2271AS, Rev. 1, "Preoperational Test Program" (shows Feedwater System).

22A2801, Rev. 1, "GE Reactor System Heat Balance - Rated" System Design Specification, Dated 1/24/72.

22A2802, Rev. 1, "GE Reactor System Heat Balance - 105% of Rated" System Design Specification, Dated 1/24/72.

22A2800, Rev. 2, "Rated Steam Output Curve" Design Specification, Dated 1/9/79.

22A3148, Rev. 1, "Heat Balance, Reactor System - 105% of Rated" Information Document, Dated 1/9/79.

22A3149, Rev. 1, "Heat Balance, Reactor System - Rated" Information Document, Dated 1/9/79.

P.O. 282-F9762, Rev. 0, "Temperature Element Product Quality Checklist", Dated 9/17/74

Burns and Roe Engineering Criteria Document, Rev. 11, 3/16/82 Section G. Instrumentation and Control, Section F Equipment Classification, Appendix 3, "WNP-2 Electrical Separation Practices", Rev. 1.

D.3.2 Calculations

7.10.02, Rev. 3, "Flow Element Sizing Calculations", 10/26/76, Sheet 8.

Alden Research Laboratories Worchester Polytechnic Institute, "Calibration - Two 24" Flow Nozzle Assemblies, Serial Numbers N-1031, N-1032. The Permutit Company Purchase Order Number L-58671-1565", Dated October, 1974, (Calibration Data for C34-N001A and C34-N001B).

Vickery - Simms #BC-N-1005-5, Orifice Bore Calculations.

D.3.3 Technical Memorandum

BRI Technical Memorandum 1010, "Operation of Feedwater Delivery System" (4/29/77), (with updated Exhibits and FE #166B7135A drawings).

BRI Technical Memorandum 667, "Feedwater Delivery System" (6/26/74).

BRI Technical Memorandum 572, "Feedwater Control System" (9/21/73).

BRI Technical Memorandum 308, Rev. A, "System Description - Condensate/ Reactor Feed" (10/6/72).

D.3.4 Manuals (Vendor)

Anchor Darling Valve Company, "Instrument Manual, Operator - Maintenance Instructions and Parts Catalog for WNP-2" (V-32A, B, V-10A, B), WPPSS CVI 0251B-00-75-1, 11/28/76.

Permutit Corporation Operating Instructions for C34-N001A and C34-N001B, Rev. 1, BRI AEF 02-11-0710.

Anchor Darling Co. Instruction Manual, Operator - Maintenance Instructions and Parts Catalog", CVI 02-41B-00, Sht. 75, Issue 1.

"Self Drag Flow Control Valve Operation and Maintenance Manual", Babcock and Wilcox CVI 02-42D-00, Sht. 12.

Woodward Governor Operation and Maintenance Manual Reactor Feedwater Turbines CVI 02-12-00, Sht. 16.

Fisher Technical Bulletin 62.1:546, dated 12/76, "Type 546, 546S and 546ST, Electro-Pneumatic Transducers.

D.3.5 Drawings

Burns and Roe Drawings

Mechanical

M151, Rev. 0, "General Arrangement - Ground Floor Plan".

M152, Rev. 0, "General Arrangement - Mezzanine Floor Plan".

M153, Rev. 0, "General Arrangement - Operating Floor Plan".

M154, Rev. 0, "General Arrangement - Reactor Building and Miscellaneous Plans".

M502, Rev. 27, "Main and Exhaust Steam System, Turbine Generator Building".

M504, Rev. 36, "Flow Diagram, Condensate and Feedwater System".

M506, Rev. 28A, "Flow Diagram Miscellaneous Drains, Vents and Sealing Systems, Turbine Generator Building".

M509, Rev. 16, "Flow Diagram - Turbine Oil Purification and Transfer System, Turbine Generator Building".

M529, Rev. 28, "Nuclear Boiler System - Flow Diagram".

M610, Rev. 5, "Installation of Thermowells and Sample Probes".

M200, Sheet 335, Rev. 7, "Reactor Feedwater Piping, RFW Pumps to Reactor", 5/16/80.

M543, Rev. 25, "Flow Diagram - Reactor Building Primary Containment Cooling and Purging System".

M617, Sht. 64A, Rev. 6, "IR-64 Legend"
Sht. 64B, Rev. 4, "Connection Diagram IR-64"
Sht. 64C, Rev. 7, "IR-64 Arrangement"
Sht. 64D, Rev. 4, "Connection Diagram IR-64"
Sht. 12A, Rev. 6, "Inst. Rack IR-12 Legend"
Sht. 12B, Rev. 4, "Inst. Rack IR-12 Arrangement"
Sht. 12C, Rev. 3, "Inst. Rack IR-12 Tubing Arrangement"
Sht. 12E, Rev. 2, "Inst. Rack IR-12 Wiring"
Sht. 12F, Rev. 4, "Inst. Rack IR-12 External Electrical Connections"
Sht. 12G, Rev. 0, "Inst. Rack IR-12 External Electrical Connections"
Sht. 12D, Rev. 5, "Inst. Rack IR-12 Tubing Arrangement Cont."

M619, Sht. 85, Rev. 5, "Inst. Rack IR-1B Connection Diagram"
Sht. 110, Rev. 4, "IR-12 Instrument Connection Diagram"
Sht. 112, Rev. 6, "IR-12 Instrument Connection Diagram"
Sht. 142, Rev. 9, "IR-64 Reactor Building Inst. Rack"
Sht. 104, Rev. 5, "Inst. Rack IR-9 Connection Diagram".

M621, Sht. 1, Rev. 5, "Panel/Console/Cabinet/Rack Classification List"
Sht. 4, Rev. 2, "Panel/Console/Rack List".

M620, Sht. 504-17, Rev. 0, "H.P. Heater Outlet Line M.O. Valve Control Logic Diagram"
Sht. 506-10, Rev. 1, "Reactor Feedwater Pump Turbine RFW-DT-1A Drain Valve Control Sch. and Logic Diagram".

M200-335, Rev. 7, "Reactor Feedwater Piping RFW Pumps to Reactors", 5/22/80.

M502, Rev. 27, "Flow Diagram - Main and Exhaust Steam System, T.G. Building", 2/25/83.

M504, Rev. 36, "Flow Diagram - Feedwater and Condensate System, T.G. Building", 1/14/83.

M506, Rev. 28A, "Flow Diagram - Misc. Drains, Vents and Sealing System T.G. Building", 1/28/83.

M509, Rev. 16, "Flow Diagram - Turbine Oil Purification and Transfer System T.G. Building", 12/10/82.

M529, Rev. 28, "Flow Diagram - Nuclear Blr. Main Steam System, Reactor Building", 3/4/83.

M610, Rev. 5, "Installation of Sample Probes and Thermowells", 10/25/82.

M617-12A, Rev. 6, "Instrument Rack IR-12 Legend", 5/26/82.

M617-12B, Rev. 4, "Dwg. Voided by PED 220-I-0772", 10/08/81.

M617-12C, Rev. 3, "Instrument Rack IR-12 Tubing Arrangement", 5/26/82.

M617-12D, Rev. 5, "Instrument Rack IR-12 Tubing Arrangement", 5/26/82.

M617-12E, Rev. 2, "Dwg. Voided by PED 220-I-0772", 11/13/81.

M617-12F, Rev. 4, "Dwg. Voided by PED 220-I-0772 Electrical Connections" 10/12/79.

M617-12G, Rev. 0, "Instrument Rack IR-12 External Electrical Connections".

M617-64A, Rev. 6, "Instrument Rack IR-64 Legend", 2/3/83.

M617-64B, Rev. 4, "Dwg. Voided by PED 220-I-0772", 12/18/81.

M617-64C, Rev. 7, "Instrument Rack IR-64 Tubing", 5/26/82.

M617-64D, Rev. 4, "Dwg. Voided by PED 220-I-0772", 12/28/81.

M619-85, Rev. 5, "IR-1B Reactor feed Pump 1B Instrument Rack", 3/14/83.

M619-142, Rev. 9, "IR-64 Reactor Building Instrument Rack El. 501'-0", Div. II", 3/14/83.

M620-504-17, Rev. 0, "H.P. Htr. Outlet Line M.O. Valve Control Logic Diagram", 9/7/76.

M620-506-10, Rev. 1, "Reactor Feedwater Pump Turbine RFW-DT-1A Drain Valve Control Schematic and Logic Diagram", 3/1/76.

M621-1, Rev. 5, "PNL Console Cabinet Rack List", 6/12/82.

M621-4, Rev. 2, "PNL Console Cabinet Rack List", 4/14/77.

Various Vendor Drawings

Control Components Inc. Drawing No. 9225, Rev. 11, "Self Drag Element 12" x 12" Angle Body" (1/6/77), BRI AEF #42D-00-0015 (RFW-FCV-10).

Woodward Governor Co. Drawing #9930-333, Sheet 2, "Control - 2301 Panel" (11-23-73).

Delaval Turbine Inc. Drawing C-72374, Sheets 9, Rev. 9; 13, Rev. 10, "Woodward Governor Schematic".

Delaval Turbine Inc. #CCA-2561, Rev. 2, "Reactor Feedpump Drives by Delaval Turbine Inc." (5/5/72), shows performance curves.

Ingersoll-Rand Inc. #49056, "Reactor Feed Pump curves" (7/10/72).

Johnson Controls Drawing #B-220-063.0, H22-P015, Sheet 1, Rev. 3, Sheet 1, Rev. 5, "Line Identification List", Rack H22-P015.

Johnson Controls Drawing #B-220-063.0, H22-P015, Sheet 2, Rev. 2; Sheet 3, Rev. 2; Sheet 4, Rev. 2; Sheet 5, Rev. 2; Sheet 5A, Rev. 0; Sheet 5B, Rev. 0; Sheet 5C, Rev. 0.

Permutit Corporation Drawing 556-27984, Rev. 6, "Outline and Assembly - Feedwater Flow Pipe Section, Size (24") 20.668" X 10.334" (directly references D-4 and C-1 and C-2), Dated 11/28/73.

Permutit Corporation Drawing #556-28016, Rev. 1, "Tube Bends Layout - For Feedwater Flow Element - Size 20.668" X 10.334 (24" - Sch. 120), (directly references C-1 and C-2), Dated 12/29/71.

Permutit Corporation Drawing #555-26992, Rev. 1, "Flow Straightener for 24" Sch. 120 Pipe - Project Hanford II", Dated 9/27/73.

Johnson Controls, Inc. Drawing #D-220-2000 - FX-6A, Rev. 0, "Local Flow Test Connection WPPSS Nuclear Project No. 2", Dated 5/16/79 (shows C34-N001A flow test connections and orientations).

Bovee and Crail Inc. Drawing #RFW-418-1.2, Rev. 11, "From Flow Meter to Reactor Vessel (Line "A"), (shows C34-N001A and mounted to piping - shows pressure connection orientation and piping dimensions), Dated 7/15/75.

Bovee and Crail Drawing #RFW-418-1.2, Rev. 11, "From Flow Meter to Reactor Vessel (Line 'A'), Date 7/15/75.

Jelco Drawing #757-D-622, Rev. C, "Tubing Arrangement IR-12", shows C34-N002A rack interconnections and rack connections.

Jelco Drawing #757-E-675, Rev. 0, "Electrical Wiring Diagram, Instrument Rack IR-12", shows wiring.

Jelco Drawing #757-E-538, Rev. D, "Instrument Assembly IR-12", shows rack placement of C34-N002A.

Jelco Drawing #757-E-535, Rev. D, "Instrument Assembly IR-12", shows rack side views.

Circle A.W. Products Drawing #757-E-532, Rev. D, "Instrument Assembly IR-64".

Bovee and Crail Drawing #RFW-415-8.10, Rev. 6, "Drain From 30" Reactor Feedwater Line to High Pressure Condenser HX-9", 3/25/80.

Bovee and crail Construction Drawing #RFW-418-3, "Reactor FW from Flow Meter to Reactor Vessel (Line "A"), Rev. 5.

Anchor Darling Valve Company Drawing #3084-3, Rev. B, "24 in. - 900# swing check valve, RFW-V-32A (B223-F032)".

Jelco Controls Inc. Drawing #757-E-703, Rev. B, "Electrical Wiring Diagram IR-62".

Circle A.W. Products Co. Drawing #757-E-544, Rev. C, "Instrument Assembly, IR-9".

Jelco Controls Drawing #757-C-619, Rev. C, "Tubing Arrangement; Instrument Rack IR-9".

Johnson Controls Drawing #D-220-072.0 - RFT-1B/IR-1B, Rev. 1, Line Identification List".

Johnson Controls Drawing #D-200-245-TG-441, Rev. 0, "Tubing Routing (As-Built)".

Jelco Controls Drawing #757-E-506, Rev. B, "Instrument Assembly, Rack 1B".

Jelco Controls Drawing #757-E-611, Rev. C, "Tubing Arrangement, Rack 1B".

Jelco Controls Drawing #757-E-664, Rev. B, "Electrical Wiring Diagram, Rack 1B".

Circle A.W. Products Drawing #757-A-506, Rev. C, "Material List, Rack 1B".

Control Components Inc. Drawing 9225, Rev. 2, "Self Drag Element 12" X 12" Angle Body", Shows technical data on RFW-FCV-10 (required output of RFW-E/P-10).

Jelco Controls Drawing 757-E-705, Rev. B, "Electrical Wiring Diagram IR-64".

Circle A.W. Products Co. Drawing 757-E-597, Rev. C, "Instrument Assembly IR-62".

D.3.6 Memoranda

Letter dated 4/12/82, no number, "RETRAN Initialization of WNP-2 Model (Draft)".

Letter dated 9/15/80 to G. L. Gelhaus from F. J. Markowski/S. F. Deng, "WNP-2 RETRAN Plant Model, Addition of Plant Control Systems".

WPPSS IOM to R. J. Barbee, Plant Technical from C. A. Fu, G.E. Std. and A WNP-2, "FW Flow Meter Calibration", Dated 1/26/83.

IOM EN-RLH-81-05, "Containment Isolation and Testability Evaluation", R. L. Heid, 10/12/81.

BRWP-RO-82-92, "Containment Isolation Review", 3/18/82.

BRWP-RO-82-153, "Same as G-3", 6/1/82.

BRAD-41B-82-002, "Contract 41B RFW-V-32A, B, "Valve Seat Modifications - Quotation Request", 1/21/82.

BRAD-41B-77-014, 6/11/77, "Revised Thermal Transient Data for RFW Valves RFW-V-10A, B and RFW-V-32A, B".

Rosemount Inc., "Material Report and Certification GE Purchase Order No. 282-F-9762", Dated 2/2/74.

Rosemount Inc., "Certificate of Compliance and System Calibration Data Sheet", Dated 8/22/74.

D.3.7 Contract Specifications (Technical)

Specification 2808-59, "Instrumentation and Control Boards".

Specification 2808-215, "Mechanical Equipment, Installation and Piping", Section 15B.

Specification 2808-220, "Instrumentation Installation" Division 50.

BRI Contract Bid Specification 2808-41, Attach. 1, "Nuclear Valve List - Nuclear Boiler, Reactor Feedwater", Page 15A-35, Rev. 3, 3/9/76, Pages 15A-157, 158, 166, 167, 140, Bid Issue 7/17/73.

Anchor Darling Contract Specification 2808-41, Part V, "Valve Specification".

Specification 2808-1, "NSSS Equipment Specifications".

Contract 2808-62, "Electrical Cable" Section 16A, Page 16A-6, (Guies Type L2 Cable for RFW-TF-41A).

Specification 2808-218, Section 50A, "Instrumentation and Control Board Installation".

Specification 2808-58, "Local Instrument Racks".

Specification 2808-218, "Electrical Installation", Section 50A, "Instrumentation and Control Boards Installation".

Johnson Controls Contract 220, Tubing Isometric Drawings.

WPPSS Document Change Control "FJN" #WNP2WBG-215-F-78-1401 (Contract Modification - Reactor Feedwater Calibration Standard).

D.3.8 Other

Instrument Society of America Reprint, "Survey of Information Concerning the Effects of Nonstandard Approach Conditions Upon Orifice and Venture Meters", P. S. Starrett, H. B. Voltage, P. F. Halfpermy, July 1980.

System Description No. 72, "Feedwater System", WPPSS Nuclear Project No. 2, Rev. 0, 9/25/75, pages 29, 30.

WPPSS Power Ascension Test 8.2.23.0, "Feedwater System Power Ascension Test Procedure", rough draft.

BWR Systems Analysis Course, Vol. II, Tab. 15, "Feedwater Level Control System" (6/6/81).

Instrument Society of America ISA-S26 (1968), "Dynamic Response Testing of Process Control Instrumentation".

WPPSS T/SU SPR-E-2156 (2/24/83), "RFW-FCV-10 Pressure Switch and Solenoid Valve".

WNP-2 FSAR, Para. 7.7.1.4, "Feedwater Control System"; 6.2.4, "Containment Isolation System"; 10.4.7.3,".

Code of Federal Regulations. 10CFR50, Appendix A, Criterion 55, Page 402.

NRC NUREG-0800, "Standard Review Plan", Para. 6.2.4, "Containment Isolation System", Rev. 2 (7/81).

D.5.4 Engineering Mechanics References

D.5.4.1 Design Requirement References

M400-3 Engineering Criteria Document Appendix 2 Pipe Support Design Guide.

Technical Memorandum 1271, QCII Equipment Nozzle Allowable Loads 6/14/82.

D.5.4.2 Calculations

8.42.8000 Revision 1 Pipe Stress Code.

8.16.2013 Hanger Design Calculation for RFW-24.

8.16.4983 Hanger Design Calculation for RFW-944N.

8.16.72.1 Hanger Design Calculation for RFW-943N, RFW-21, RFW-17.

SECTION E - SYSTEMS INTERACTIVE REVIEW REFERENCES

E.1 Fire Protection

E.1.1 Specifications

WNP-2, Final Safety Analysis Report, Appendix F, Ammendment 26
10CFR50, Appendix R.

APCSB 9.5-1, Appendix A, Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976.

E.1.2 Calculations

2.06.04, Rev. 1, Radwaste Bldg./Control Bldg. Feeder and Voltage Drop Calculations.

2.06.05, Rev. 3, Reactor Bldg. Feeder and Voltage Drop Calculations.

2.07.01, Rev. 2, High Voltage Cable Sizing - Ampocities and Conduits.

2.07.03, Rev. 1, A.C. Motor Control Center Bus and Cable Sizing.

E.1.2 Technical Memorandum

TM 1227, Rev. 3, Fire Protection Study, 4/22/82.

TM 1272, Rev. 2, Thermo-lag Fire Barriers for Electrical Cables, Cable Ampocity Derating, 10/22/82.

E.2 Pipe Break/Missile Evaluation/Jet Impingment/Falling Objects/Flooding

E.2.1 Specifications

22A2625, System Criteria and Application for Protection Against the Effects of Pipe Breaks, June 15, 1973.

22A3046, Rev. 1, Core Standby Cooling System Network Design Specifications, 7/14/77.

22A2802, Rev. 2, GE Reactor System Heat Balance 105% Rated Power.

BRI Engineering Criteria Document.

E.2.2 Calculations

5.49.050, Rev. 1, Pipe Break Analysis, Inside Containment.

5.49.051, Rev. 1, Target Determination, Pipe Breaks Inside Containment, 12/17/82.

5.49.052, Rev. 1, Shutdown Analysis for Pipe Breaks Inside Containment.

5.51.050, Rev. 1, Pipe Break Analysis, Outside Containment

5.51.051, Rev. 1, Target Resolution for Postulated Targets Outside Containment.

5.51.052, Safe Shutdown Analysis Outside Containment.

8.01.51, Rev. 0, WPPSS N.P. No. 2, LPCS Pipe Whip Analysis.

5.49.056, Rev. 3, Target Resolution for Postulated Targets Inside Containment, Draft.

SVIII, Vol. 81, Radwaste Missile Barriers, MG Sets 1 and 2

5.50.51, Target Determination for Credible Missiles Outside Containment, 6/25/82.

E.2.3 Technical Memorandum

TM 1020, Rev. 1, Regulatory Guide 1.46, Recommendation Concerning Implementation, 10/28/77.

TM 1085, Rev. 1, Pipe Break Outside of Containment - Structural Effects, 10/6/78.

TM 1151, Criteria for the Pipe Break and Missile Redundancy Evaluation Outside Primary Containment, 6/27/79.

E.2.4 Drawings

Electrical

E-550
E-551

Mechanical

M-519
M-520
M-521
M-523
M-529
M-530
M-543
M-557

Structural

- S-794
- S-918
- S-1001, Rev. 10
- S-1000, Rev. 21
- S-783, Rev. 12
- S-1024, Rev. 2

Isometric

- RWCU-895-8.12
- RWCU-894-14.21
- RWCU-277-1.3
- RWCU-895-1.7
- D220-X-106
- D220-X-108
- D-220-031.0-IR-68
- CEP-625-11.12
- M200 Sht. 129
- RCIC-664-1.7
- M200 Sheet 126
- M200 Sheet 128
- D220-7.1-X-78(e)
- EDR-571-4.5
- HPCS-630-31.33
- HPCS-630-29.30
- ED-A-9
- ED-A-16
- ED-A-6
- ED-A-5
- M-200, Sheet 2
- RHR-4434-1

Hanger

RWCU-181
RWCU-928N
RWCU-238
HPCS-64
HPCS-66

F20APKD500X4-C IR-RHR Pump Detail

238X178AD
239X527AD
239X241AD
238X201AD

E.2.5 Other

WNP-2, Final Safety Analysis Report.

NUREG 75/087, Standard Review Plan, Sections 3.5.1, 3.5.2, 3.6.1, 3.6.2.

Regulatory Guide 1.46.

Regulatory Guide 1.70.

BTP MEB 3-1 and APCSB 3-1, Section B.3, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment.

"Proposed ASME Non-Mandatory Appendix - Design Rules for Pipe Whip Restraints" Article L-1000, NF 54, N/D 77-66 N76-6 January 1980.

Crane Technical Paper #410, "Flow of Fluids Through Valves, Fittings, and Pipe".

ASME Boiler and Pressure Vessel Code, Section III, Appendix I.

AISC 7th Edition, "Manual of Steel Construction", June 1973.

American National Standard ANS-58.2, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture", ANSI-176.

Teledyne Engineering Services Technical Report TR-4536-1 Missile Impact Analysis, November 7, 1980.

Hexcel Manual TSB122 - Design Data for Preliminary Selection of Honeycomb Energy Absorption Systems

Gwaltney, R. C., "Missile Generation and Protection in Light Water Cooled Power Reactor Plant", Oak Ridge National Laboratory.

R. P. Kennedy, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects", Holmes & Narver, Inc., September 1975.

BC-TOP-9A, "Design of Structures for Missile Impact", Bechtel Power Corporation, September 1974.

ANSI 177-1974, Plant Design Against Missiles.

E.3 Qualification of Safety Related Equipment for Environmental Conditions and Dynamic Loads

E.3.1 Calculations

Supply System Calculations:

NE-02-81-06-0, August 13, 1982, "WNP-2 Subcompartment Temperature and Pressure Analysis for Postulated High Energy Pipe Breaks in the Reactor Building".

NE-02-81-07-0, September 10, 1982, "Postulated Pipe Break of 4" RCIC(13)-4 in RCIC Pump Room (R15) and Room (R112) Above RCIC Pump Room".

NE-02-81-08-0, September 8, 1982, "Postulated Pipe Break of 4" RCIC(13)-4 in Room (R113) Above RHR Pump 2C Room".

NE-02-81-09-0, September 10, 1982, "Postulated Pipe Break of 4" RCIC(13)-4 in TIP Room (R308)".

NE-02-81-13-0, September 10, 1982, "Postulated Pipe Break of 6" RWCU(2)-4 in the Valve Room (R313) Above TIP Room".

NE-02-81-14-0, September 16, 1982, "Postulated Pipe Break of 6" RWCU(2)-4 in Valve Room (R408) North of Containment EL 522'".

NE-02-81-15-0, December 16, 1982, "Postulated Pipe Break of 4" RWCU(1)-4 in RWCU Pump Room (R406 or R407)".

NE-02-81-16-0, September 14, 1982, "Postulated Pipe Break of 6" RWCU(1)-4 in Valve Room (R409) Above RWCU Pump Rooms".

NE-02-81-17-0, December 16, 1982, "Postulated Pipe Break of 6" RWCU(2)-4 in Valve Room (R509) North of Containment EL 548'".

NE-P2-81-18-0, November 5, 1982, "Postulated Pipe Break of 6" RWCU(1)-4 in Valve Room (R511) South of Containment EL 548'".

NE-02-81-19-0, December 16, 1982, "Postulated Pipe Break of 6" RWCU(1)-4 in the RWCU Heat Exchanger Room (R510)".

NE-02-81-20-0, December 16, 1982, "Postulated Pipe Break of Auxiliary Steam Line".

NE-02-82-41-0, September 10, 1982, "Cooldown of Reactor Building Rooms Followng a Pipe Break - Computer Model".

BRI Calculations:

5.07.14.1, October 29, 1976, "Blowdown From 4" AS(11)-2".

5.07.31, October 22, 1976, "Volume and Vent Area for Reactor Building".

5.07.32, October 25, 1976, "Pressurization of HPCS Rooms R11/R106 (El. 422'3")".

5.07.62, September 21, 1979, "Pressurization of Rooms 509/510 at El. 545'".

5.07.59.2, September 20, 1979, "Modification of Valve Room 408 at El. 522'".

E.3.2 Other

ANCR-NUREG-1335, September 1976, "RELAP4/MOD5 A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - User's Manual".

NUREG/CR-1185, Addendum 1, June 1980, "COMPARE-MOD1 Code Addendum" and LA-7199-MS, March 1978, "COMPARE-MOD1: A Code for the Transient Analysis of Volumes with Heat Sinks, Flowing Vents, and Doors".

NUREG-0800, July 1981, "US NRC Standard Review Plan".

NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment".

WNP-2 Environmental Qualification Report for Safety Related Equipment, September 1982.

WNP-2 Dynamic Qualification Report for Safety Related Equipment, September 1982.

E.4 Structural Members

E.4.1 Specifications

WNP-2 Final Safety Analysis Report

BRI Engineering Criteria Document

E.4.2 Calculations

SV-184, Pipe Break in Main Steam Tunnel.

SIII-18, Turbine Generator Building - Operating Floor.

SV-14, Reactor Building Elevation 441'-0" and 444'-0".

5.51.050, Rev. 1, Pipe Break Analysis Outside Containment.

E.5 Instrument Racks

E.5.1 Specifications

Contract 2808-58; Local Instrument Racks

E.5.2 Drawings

BRI:

M 621, Rev. 5, Instrument Rack List

M 621a, Rev. 2, Instrument Rack List

M 567, Rev. 7, Reactor Building General Arrangement

M 568, Rev. 23, Reactor Building General Arrangement

M 569, Rev. 23, Reactor Building General Arrangement

M 584, Rev. 6, Standby Service Water Pump House Arrangment

S 540, Rev. 8, Pump House Instrument Rack Supports

S 1083, Rev. 1, Reactor Building Instrument Rack Supports

E538-15VF-1, Rev. 0, Arrangement Drawing For IR-21

E538-15VF-2, Rev. 0, Arrangement Drawing For IR-22

E538-16VF-1, Rev. 0, Arrangmement Drawing For IR-24

E538-16VF-2, Rev. 0, Arrangmement Drawing For IR-25

E538-16VF-3, Rev. 0, Arrangement Drawing For IR-26

E538-18VF-1, Rev. 0, Arrangmement Drawing For IR-61

E538-18VF-2, Rev. 1, Arrangmement Drawing For IR-62

E538-18VF-3, Rev. 2, Arrangmement Drawing For IR-63

E538-19VF-1, Rev. 0, Arrangmement Drawing For IR-64

E538-19VF-2, Rev. 0, Arrangmement Drawing For IR-65

E538-19VF-3, Rev. 1, Arrangmement Drawing For IR-66

E538-20VF-1, Rev. 1, Arrangmement Drawing For IR-67

E538-20VF-2, Rev. 0, Arrangmement Drawing For IR-68

E538-21VF-1, Rev. 1, Arrangmement Drawing For IR-69

E538-21VF-2, Rev. 0, Arrangmement Drawing For IR-70

E538-22VF-1, Rev. 0, Arrangmement Drawing For IR-71

E538-22VF-2, Rev. 0, Arrangmement Drawing For IR-72

E538-23VF-1, Rev. 0, Arrangmement Drawing For IR-73

E538-23VF-2, Rev. 0, Arrangmement Drawing For IR-74

Vendor Drawings: Jelco (Circle AW), Rack Outline and Details

IR-21	CVI 02-58-00-32-1; -2; -3	Rev. 2-24-78
IR-22	CVI 02-58-00-33-1; -2; -3	Rev. 1-31-78
IR-24	CVI 02-58-00-40-1; -2; 03	Rev. 12-5-77
IR-25	CVI 02-58-00-41-1; -2; -3	Rev. 1-31-78
IR-26	CVI 02-58-00-4-1; -2; -3	Rev. 1-31-78
IR-61	CVI 02-58-00-9-1; -2; -3	Rev. 1-31-78
IR-62	CVI 02-58-00-25-1; -2; -3	Rev. 1-31-78
IR-63	CVI 02-58-00-20-1; -2; -3	Rev. 1-31-78
IR-64	CVI 02-58-00-22-1; -2; -3	Rev. 1-31-78
IR-65	CVI 02-58-00-10-1; -2; -3	Rev. 1-31-78
IR-66	CVI 02-58-00-13-1; -2; -3	Rev. 3-16-78
IR-67	CVI 02-58-00-24-1; -2; -3	Rev. 1-31-78
IR-68	CVI 02-58-00-11-1; -2; -3	Rev. 1-31-78
IR-69	CVI 02-58-00-19-1; -2; -3	Rev. 1-31-78
IR-70	CVI 02-58-00-21-1; -2; -3	Rev. 1-31-78
IR-71	CVI 02-58-00-12-1; -2; -3	Rev. 1-31-78

IR-72	CVI 02-58-00-18-1; -2; -3	Rev. 3-16-78
IR-73	CVI 02-58-00-23-1; -2; -3	Rev. 1-31-78
IR-74	CVI 02-58-00-50-1; -2; -3	Rev. 2-28-78

E.5.3 Other

Equipment Environmental and Seismic Qualification Documentation File; QID 185002, 10-13-82.

Circle AW Products Letter to Burns and Roe of 1-6-77; CAWBR-58-77-051.

Burns and Roe Letter to Circle AW Products of 11-16-77; BRCAW-58-77-083.

Pressure loads due to pipe break do not necessarily peak with pipe whip and jet impingement loads; however, in the analysis, they are considered to act simultaneously.

With regard to pipe break, when high energy pipes under pressure fail, a fluid jet is created. The associated jet impingement force on a target as well as the reaction force exerted on the piping by the fluid jet force have a time history qualitatively presented in Figure 3.6-118. This force is conservatively idealized as a step function load. For the fluid forces associated with these pipe failures, see Table 3.6-6.

To obtain a solution for the actual complex system, the structure is idealized by an equivalent single degree of freedom system (see Figure 3.6-119) following the procedures described by J. M. Biggs in Chapter 5 of "Introduction to Structural Dynamics" (Reference 3.6-1). The response of this mathematical idealization to a step function load (jet impingement) or to a step function load concurrently with an impact loading (due to whipping pipe) involves an energy transfer from the impacting object to the impacted structure. The following exposition on how this energy transfer is addressed makes use of procedures that have been presented by the Bechtel Corporation in its report on missile impact, Topical Report BC-TOP-9A, Revision 2 (Reference 3.6-13).

3.6.1.6.3.2 Structural Response to Whipping Pipe Missile Impact Load

a. Discussion

A method of energy-balance procedures is utilized in order to evaluate the structural response, when a missile impacts a target. The method utilizes the strain energy of the target at maximum response to counteract the residual kinetic energy of the target or target missile combination that results from the missile impact.

A missile of mass M_m is postulated to strike a spring-backed target mass, M_e, with a velocity, V_s. Since the actual coupled mass during impact varies, an estimated average effective target mass, M_e, is used to evaluate the inertia effects during impact. The impact of the missile is considered plastic. This assumes that the missile remains in contact with the target after impact.

WHERE PIPING IS RESTRAINED AND BREAK SEPARATION IS LIMITED TO ONE HALF PIPE DIAMETER OR LESS A FAN JET IS POSTULATED. A FAN JET IS PERPENDICULAR TO THE PIPE CENTERLINE AND EXTENDS 360° AROUND THE BREAK AT A 10° HALF ANGLE AS SHOWN IN FIGURE 3.6-148.

The values of μ_r should be less than the allowable ductility ratios, μ, given in Table 3.6-1.

3.6.1.6.3.3 Jet Impingement

Jet impingement loads are loads that emanate from a break in a high energy line. It is postulated that the characteristics of the jet are such that the jet exits from a break opening in the pipe equal in area to the cross sectional area of the pipe itself (see Figure 3.6-117). The jet is postulated to travel conforming to the configuration of the cross sectional area of the pipe for a distance of five pipe diameters and then to diverge at an angle of divergence of 10°. For the jet thrust forces at the postulated breaks, see Table 3.6-6. Jet loads impacting structures are treated as equivalent static loads. A dynamic load factor is applied to the jet force emanating from the pipe and the resulting load is modified by an appropriate load factor according to its use in combination with other loads. The structure impacted is then evaluated for structural capability.

3.6.1.6.4 Allowable Design Stresses and Strains

For allowable design stresses and strains for reinforced concrete and structural steel, see 3.8.4.5 and Tables 3.8-12 and 3.8-17, except as modified in 3.6.1.6.4.1 and 3.6.1.4.2.

3.6.1.6.4.1 Pipe Whip Loading With or Without Other Loads

The acceptability of pipe whip loading with or without other loads is considered from two aspects:

a. The overall structural response of the impacted structural element

b. The local damage sustained by the impacted structural element.

The overall structural response is considered acceptable if the ductility ratio resulting from the loading does not exceed the maximum allowable ductility ratios as given in Table 3.6-1. The determination of ductility ratios utilizes the procedures set forth in 3.6.1.6.3 and the loading combinations in 3.6.1.6.6. In using these procedures, the allowable limit on section strength, M , used in the determination of yield displacement X_e, (3.6.1.6.3.2e, Tables 3.6-9, 3.6-10 and Figure 3.6-120) is computed in accordance electrical division to which the component belongs; what the function of the component is; the various references, such as the drawings, in which the component is found; devices interconnecting the component and another system; and additional information of this type. This coding facilitates storage of the input for retrieval at any time.

Table 3.6-6 lists the high energy design basis break locations outside containment, the piping subsystems involved, the pipe diameter, the plan figure showing the piping subsystem, and the maximum blowdown thrust or the thrust versus time figure. ~~and the room or area containing the postulated pipe break. Figures 3.6-41a through 3.6-41h illustrate the high energy break locations outside containment.~~

Figures 3.6-12 through 3.6-36 illustrate and list the high energy break locations inside containment.

Moderate energy crack locations are postulated in accordance with Standard Review Plans 3.6.1 and 3.6.2.

3.6.1.11.2 Method of Analysis for Postulated High Energy Fluid System Ruptures

3.6.1.11.2.1 Effects of Postulated Passive Component Failures

Postulated pipe breaks in high energy fluid systems are investigated to determine their effects on the ability to bring the plant to a safe shutdown and to limit the offsite radiological consequences to an acceptable level as stated in 10CFR50.

On a case-by-case basis, the effects of pipe whip, jet impingement, and the resulting environmental conditions on safety-related equipment are evaluated. The effects of the postulated pipe break are dependent on the fluid properties of the system, the location and orientation of the pipe break, the proximity to safety-related systems, components, and structures, and the individual design limits of the safety-related systems, components, and structures.

3.6.1.11.3 Method of Analysis for Postulated Moderate Energy Fluid System Ruptures

3.6.1.11.3.1 Approach

Postulated ruptures in moderate energy fluid systems do not generate pipe whip. The analysis investigates the effects of the environment which results from such a postulated rupture on safety-related equipment, including the effects of water spray.

The effects of the postulated moderate energy pipe cracks are dependent on the fluid properties, available fluid reservoir, drain systems, location of the safety-related equipment, components, and structures, and the individual design limits of the safety-related equipment, components, and structures.

Where moderate energy pipe cracks are postulated in close proximity to high energy systems, the environmental analysis compares the effects of both high and moderate energy pipe ruptures. The most limiting case is evaluated for safe cold shutdown.

Moderate energy pipe cracks are postulated according to the criteria in 3.6.2.1.

3.6.1.11.3.2 Method of Analysis

The locations of all postulated ruptures, resulting in through wall leakage cracks, are identified for later retrieval. The analysis assumes that the spray resulting from a postulated moderate energy rupture causes the malfunction of all equipment not enclosed by watertight compartments.

Additionally, the most damaging single random active component failure in a system not effected by the postulated passive component failure is postulated. If the direct consequences of the pasive component failure results in a turbine or reactor trip, then offsite power is assumed unavailable.

3.6.1.11.4 Summary of Analysis

~~whereas the~~ In those cases where the
~~The~~ analyses discussed in 3.6.1.11.2 and 3.6.1.11.3 ~~do not~~ identify ed a location where a postulated passive component

Impacted pipes of smaller nominal diameter than the impacting pipe are assumed to fail, regardless of wall thickness of impacted pipe. Impacted pipes of both larger nominal diameter and thinner wall thickness than the impacting pipe are assumed to develop through wall leakage cracks.

c. Additionally, a single random active component not affected by a) and b) is assumed to malfunction. Should a) or b) result in a turbine generator or reactor trip, then offsite power is assumed unavailable.

d. After a), b), and c) above have been evaluated, possible shutdown modes are analyzed. If shutdown is possible, the postulated passive component failure is not significant from a safety standpoint.

e. Should alternate shutdown modes not be available then:

   1. Reroute or relocate cable, pipe, or equipment to prevent loss of function.

   2. If (1) is not feasible, shield the adversely affected component(s) to prevent loss of function.

f. The flooding and environmental effects of moderate energy failure are evaluated to determine whether they are more severe than the high energy breaks and are addressed in 3.6.1.15.

The area temperature is evaluated by determining the limiting postulated pipe break and using RELAP4/MOD5 (Reference 3.6-21). The limiting pipe break for temperature analysis is that pipe break giving the highest energy release rate over the longest blowdown period.

The effects of flooding are evaluated by determining the limiting pipe break and calculating the effects of the fluid release. The limiting pipe break for flooding analysis is that pipe break with the highest mass flow rate over the longest blowdown period.

Peak differential pressure analysis results are provided in Table 3.6-12 and discussed in 3.6.1.20.

NO REVISION THIS PAGE

THE COMPONENT WAS RELOCATED OR PROTECTED

IMPACTED A SAFETY RELATED COMPONENT WHICH

failure in a high or moderate energy system precluded the safe shutdown and cooling of the reactor. ~~Therefore, the ruptures in fluid piping systems, which are postulated, have no effect on the ability to bring the reactor to a cold shutdown condition.~~

This analysis by actual examination of the plant is undertaken to provide results based on as-built conditions.

Design drawings are used to supplement the study in cases where piping or equipment have not been installed. Prior to fuel load, a walkdown of the plant is performed to verify the results of the analysis and confirm that all design modifications have been implemented.

Piping layouts for areas containing high and moderate energy lines, whose failure can affect the performance of safety-related equipment, are presented as Figures 3.6-43 through 3.6-62, inclusive.

(8) fix

Section 3.6.1.11 discusses in detail the methods used to demonstrate that no single postulated passive component failure, in conjunction with a single active component failure, precludes safe shutdown of the plant.

The following should serve to further clarify the method of analysis:

a. The forces developed at each postulated high energy pipe break are determined by the methods of 3.6.2.2. The effects of the resultant pipe whip and jet impingement are evaluated. Credit is taken for automatic isolation and operator action to mitigate the consequences of the postulated pipe break, if the equipment required for this function is not affected by the break or included in 3.6.1.11.4(c) below.

b. As a first step, all equipment impacted by the whipping pipe or jet is assumed to fail. If the equipment is required for safe cold shutdown or accident mitigation, a detailed analysis is performed to determine if the equipment will actually fail. Structures contacted by the whipping pipe or jet are evaluated for structural adequacy by the methods of 3.6.2.2.

3.6.1.13 Electrical Equipment Environmental Qualifications

All electrical systems, necessary for safe shutdown and necessary to maintain the plant in a safe shutdown condition, are designed to remain functional in the general area environment resulting from a high energy line break or from leakage cracks in moderate energy piping. Specific equipment is either:

a. Designed to remain functional as long as necessary in the general area environment.

b. Isolated from the general area environment in compartments capable of maintaining normal equipment operating conditions.

Certain rotating equipment cannot be designed to function in the more severe, local steam environment. However, due to physical separation, rotating equipment, of not more than one subsystem, is exposed to the local conditions which exceed the general area accident environment. Required redundancy is thus maintained for safety equipment.

Refer to 3.11 for a more complete description of environmental design of electrical equipment.

3.6.1.13.1 Identification of Equipment

Safety equipment required to mitigate the consequences of an accident and place the reactor in a cold shutdown condition is listed in Table 3.11-2. The table also indicates the required duration, following an accident, which equipment is required to operate.

3.6.1.13.2 Environmental Design

Refer to 3.11 for a discussion of environmental design and an analysis of safety-related electrical components. The section identifies the safety-related equipment that must operate in a hostile environment, and Table 3.11-2 indicates the postulated environmental envelop conditions for both the general and local accident areas.

3.6.1.13.2 Jet Impingement Barriers

JET IMPINGEMENT BARRIERS HAVE BEEN PROVIDED WHERE

For results of the steam system study, see 3.6.1.11.4. ~~An-~~ ~~alysis indicates~~ ~~jet impingement barriers~~ THEY are ~~not~~ required ~~at WNP-2 since no postulated pipe break precludes~~ reactor safe shutdown. Some room walls, floors, and ceilings act as jet impingement barriers, ~~however~~.

IN ADDITION,

TO PROTECT COMPONENTS REQUIRED FOR

3.6.2.3.2 Jet Impingement Effect

3.6.2.3.2.1 Physical Separation

The physical separation of different essential systems and components is used to ensure that the plant retains function of sufficient essential systems to assure safe shutdown in the event of a postulated LOCA, and subsequent generation of a jet stream together with an additional single random active component failure and the loss of offsite power.

Where physical separation cannot be used to protect systems, a detailed analysis is performed to determine the effects of jet impingement on their operability. If necessary, barriers are provided to protect structures, systems, and components required for a safe shutdown, to prevent offsite radiological consequences, and to mitigate the effects of a LOCA.

3.6.2.3.2.2 Jet Impingement Evaluation

The evaluation of the adequacy of physical separation included the inspection of all essential systems and their components that are necessary to start, operate, and control the essential systems required for safe shutdown. The evaluation included the following:

a. Review pipe break locations ~~inside primary containment,~~ to provide conservative jet stream orientation and geometry.

b. Review effected equipment by both design drawing examination and plant walkdown.

c. Review signals that result in the actuation of essential systems.

d. Review signals that are necessary to be returned to inside primary containment, to activate the shutdown systems.

e. Review availability of power that is required inside primary containment to operate the essential systems.

f. Review mechanical engineered safety systems required for safe shutdown.

TABLE 3.5-12

SUMMARY OF SUBCOMPARTMENT PRESSURE ANALYSIS[a]

Compartment Where Break Occurs			Piping System	Differential Pressure			
Elevation (ft.)	Room Number	Description	Line Designation	Maximum Differential (psi)	Differential Between the Rooms	Time of the Peak (sec)	Design Pressure (psi)
442	R14/R113	RHR Pump Rooms	4" RCIC (13)-4	0.33	R14, R113/R206	0.33	0.50
				0.33	R14, R113/R12, R114	0.33	0.50
				0.33	R14, R113/R15, R112	0.33	0.50
422	R15/R112	RCIC Pump Room	4" RCIC (13)-14	0.51	R15, R112/R206	0.53	0.76
				0.51	R15, R112/R14, R113	0.53	0.76
				0.51	R15, R112/R6, R116	0.53	0.76
471	R206	El. 471' Open Floor Area	4" AS (11)-2	0.05	R206/R103, R105, R106, R305, R308 R310, R306, R315	0.35	0.08
				0.05	R206/R114, R113, R112	0.35	0.08
				0.05	R206/R116, R115		
501	R308	TIP Room	4" RCIC (13)-4	0.32	R308/R305, R206, R313	0.03	0.50
501	R308	TIP Room	6" RWCU (2)-4	0.48	R308/R305, R206, R313	0.35	0.60
501	R313	El. 510' Valve Room	6" RWCU (2)-4	0.41	R313/R308, R408	0.35	0.60
522	R404	El. 522' Open Floor Area	8" CRD (12)-3	0.03	R404/R305, R504, R508	0.04	0.05

(a) Table applies to reactor building secondary containment, exclusive of the main steam tunnel, tunnel ventway, and tunnel extension.

No revision This page
3.5-94

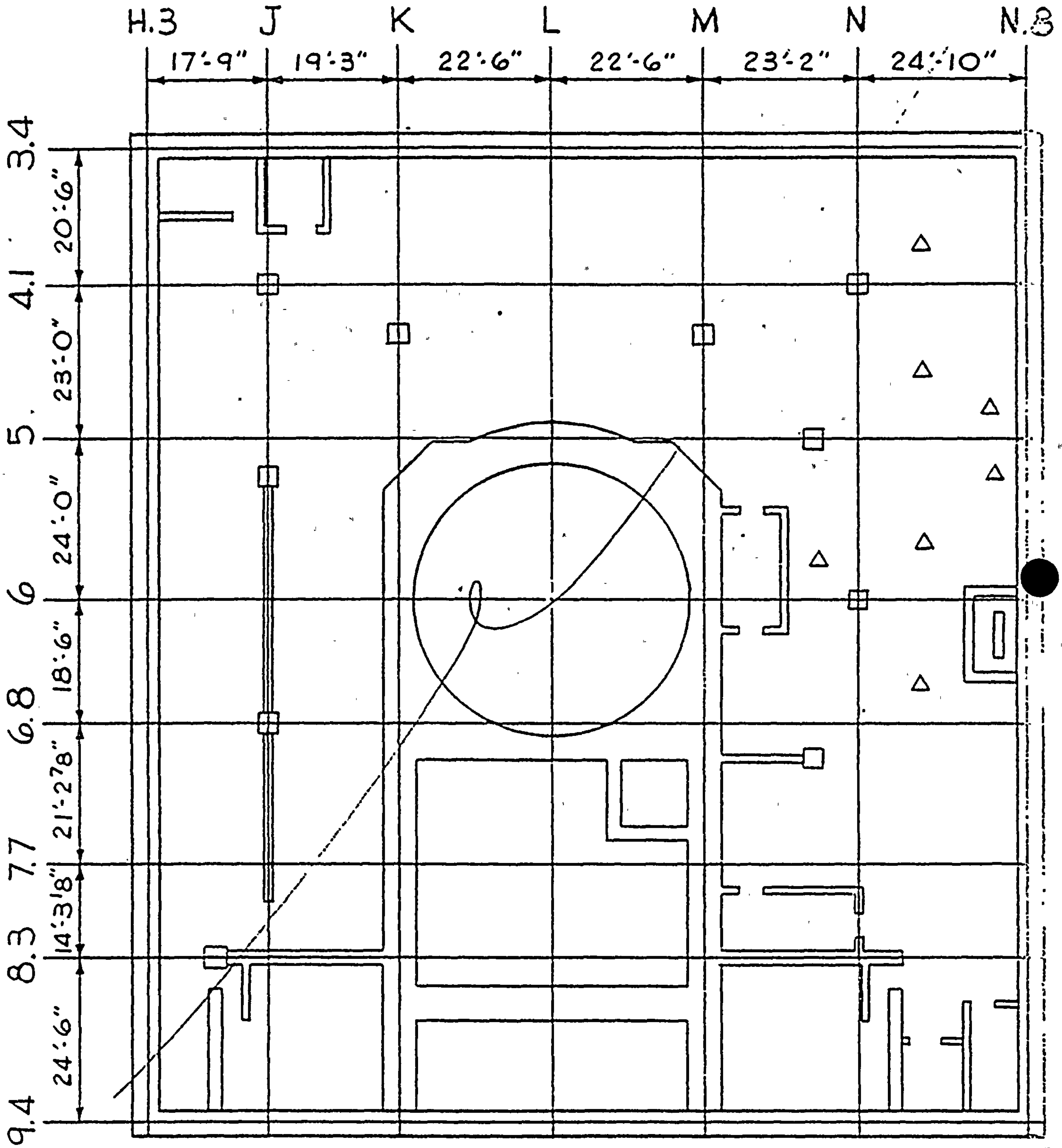
TABLE 3.6-11

DESIGN LOAD IN AREAS WHERE PIPING FAILURES OCCUR

	Pipe Break Nos.	Room	Elev. (ft.)	Differential Pressure (psi)	Differential Temperature °F Int. to Int.	Differential Temperature °F Int. to Ext.	Live Load (psf)	Hung Loads (psf) From Floor	Hung Loads (psf) From Ceiling	Equip. Loads (Kips)
120-8	~~8-10~~	R 15	422	0.51	0°	40°	-	-	59	1.4k Pump
120-4	~~4~~	R 113	441	0.33	0°	40°	250	59	68	None
120-5,6,7	~~5-7~~	R 112	441	0.51	0°	40°	250	59	68	None
139-3,4	~~41-45~~	R 206	471	0.05	0°	40°	250	32	34	None
120-1,2	~~1, 2,~~	R 313	510'-6"	0.48	0°	40°	250	40	30	None
128-11	~~21, 22~~									
128-10	~~20~~	R 408	522	1.0	0°	-	250	41	88	None
126-3,5	~~13-15,~~	R 406	522	15.0	0°	-	250	126	0	1.5k Pump
129-47,48	~~30-34~~	& 407								
126-1,2	~~11, 12,~~	R 409	535	11.0	0°	-	250	40	80	None
129-39,41,42,43,44,45	~~23-29,~~									
129-48	~~32~~									
144-27,28	~~60, 61,~~	R 511	548	4.4	20°	-	400	80	55	None
144-32	~~65~~									
126-6, 128-7,8	~~16-18,~~	R 510	548	1.8	20°	-	400	65	51	Heat Exchs.
142-20,21,22,23	~~53-55~~									16.2 & 29.5
144-29,31	~~62-64~~									
128-9	~~19~~	R 509	548	2.1	20°	-	400	88	50	None
139-1	~~46, 47,~~	R 604	572	0.03	0°	40°	250	15	36	Heat & Vent Unit 51K
148-1,2,3,5,6,7,	~~49, 70~~									
8,9,10,11,12	~~52~~									
	~~90-94~~	~~R 504~~	~~548~~	~~0.90~~	~~0°~~	~~40°~~	~~400~~	~~59~~	~~15~~	~~None~~
120-6	~~5~~	R 308	501	0.41	0°	40°	1000	63	55	None
	Steam Tunnel	R 310	501	20.0	20°	-	1000	277	41	None

NOTES: 1. For location of pipe break nos., see ~~Figures 3.6-43 through 3.6-66.~~ TABLE 3.6-6

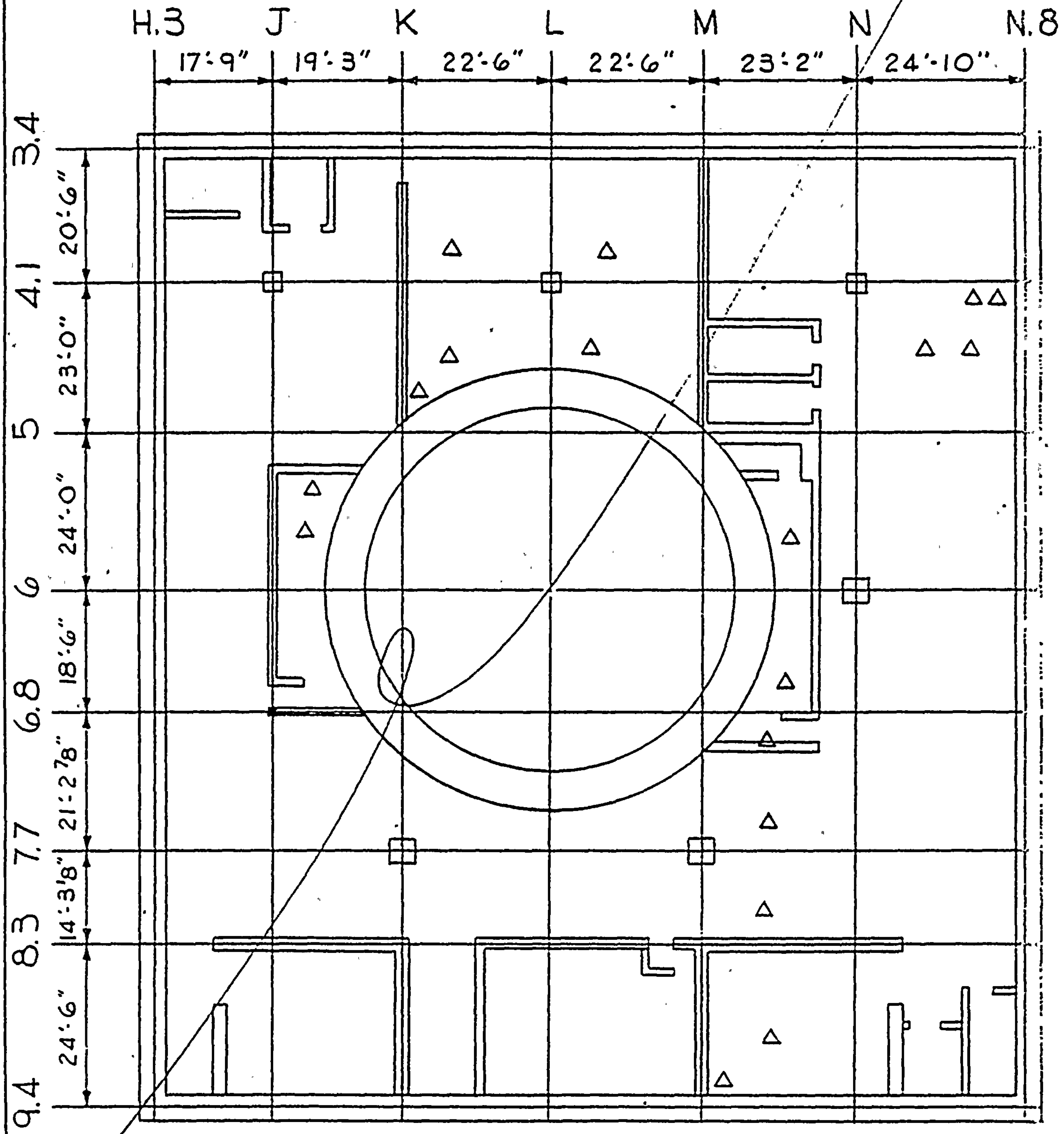
2. For vertical and horizontal seismic factors, see 3.7.



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC PLAN @ EL. 572'	FIGURE 3.6-41g

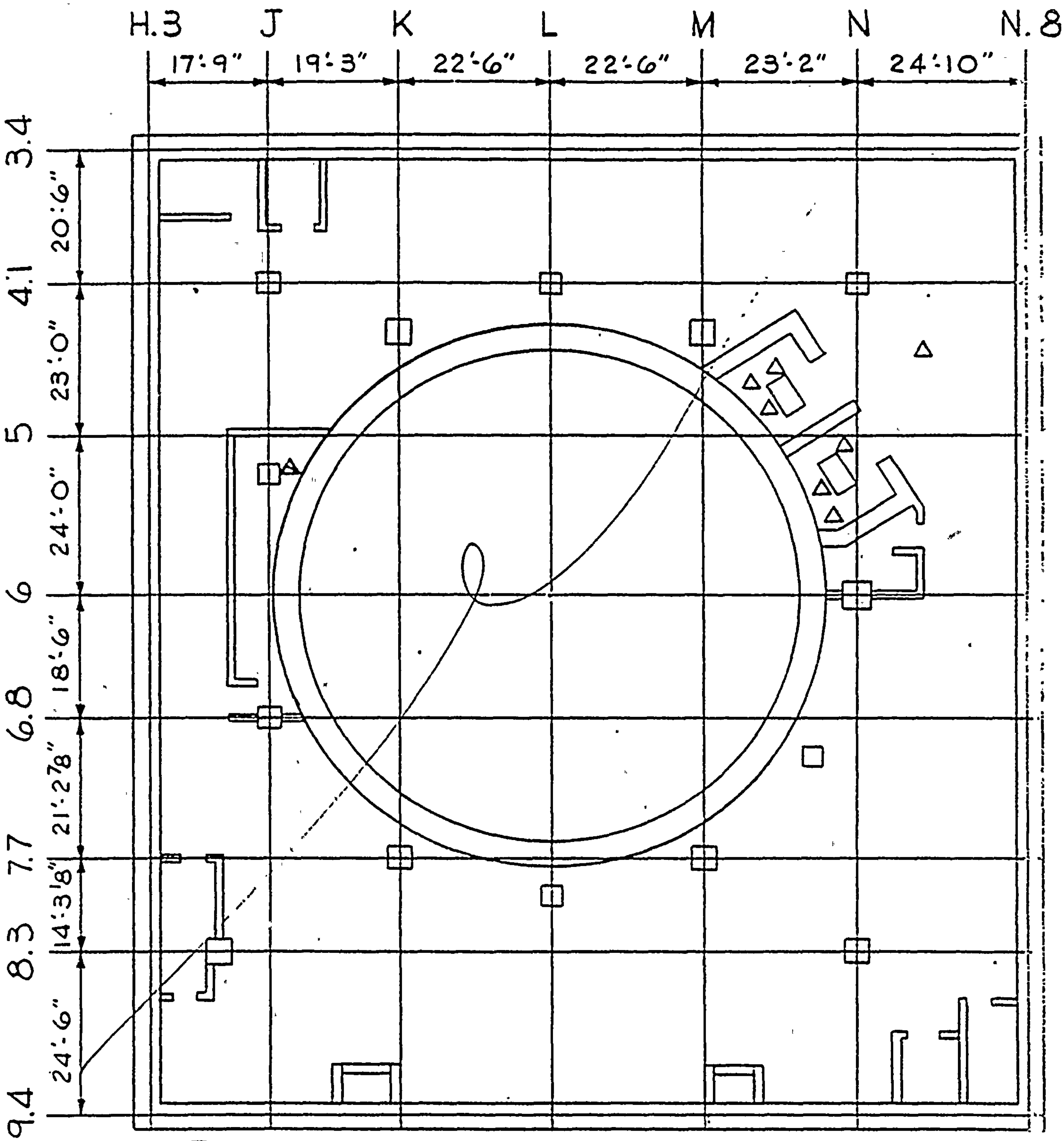
AMENDMENT NO. 25
June 1982



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 548'	FIGURE 3.6-41

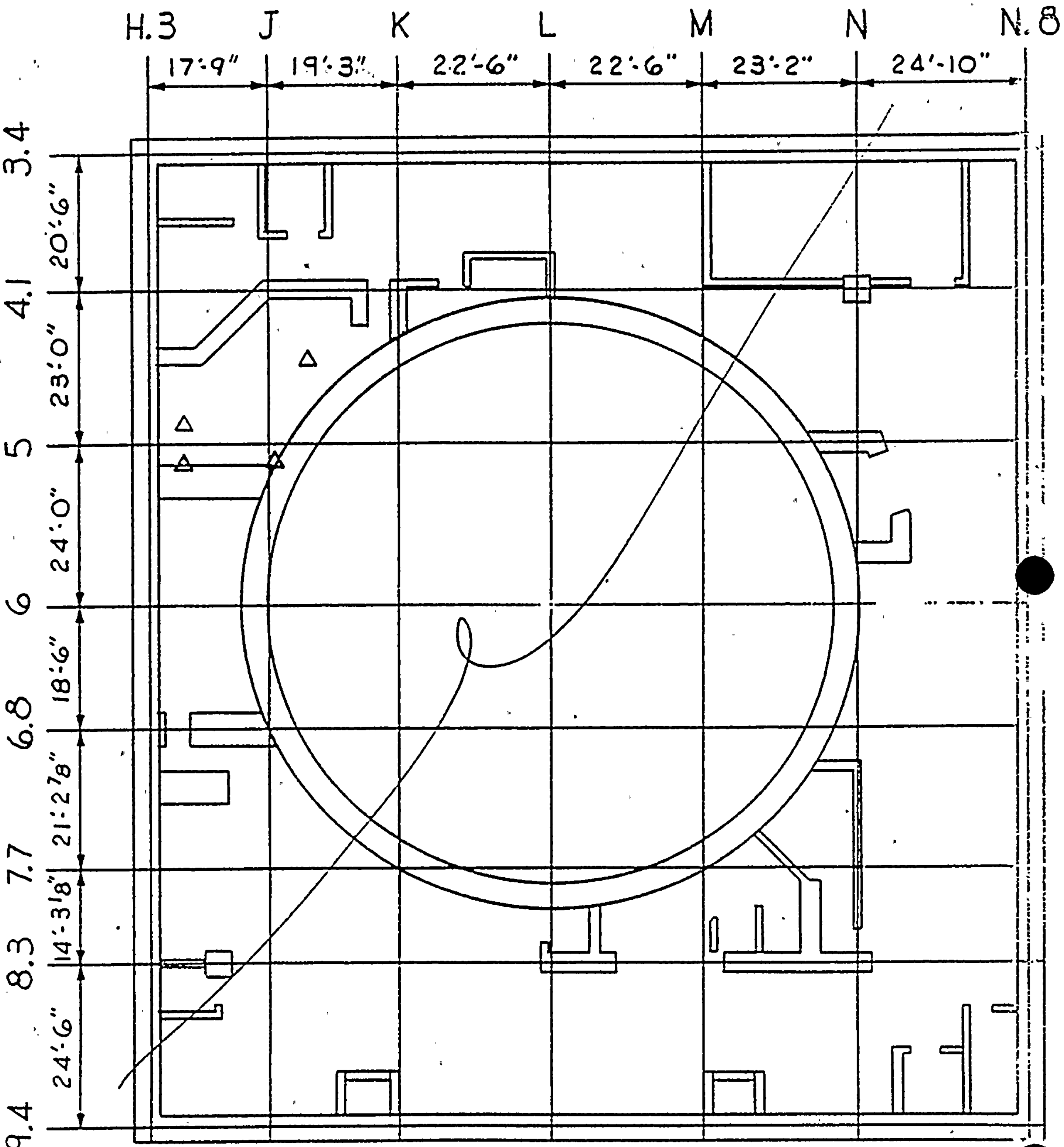
AMENDMENT NO. 25
June 1982



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 522'	FIGURE 3.6-4

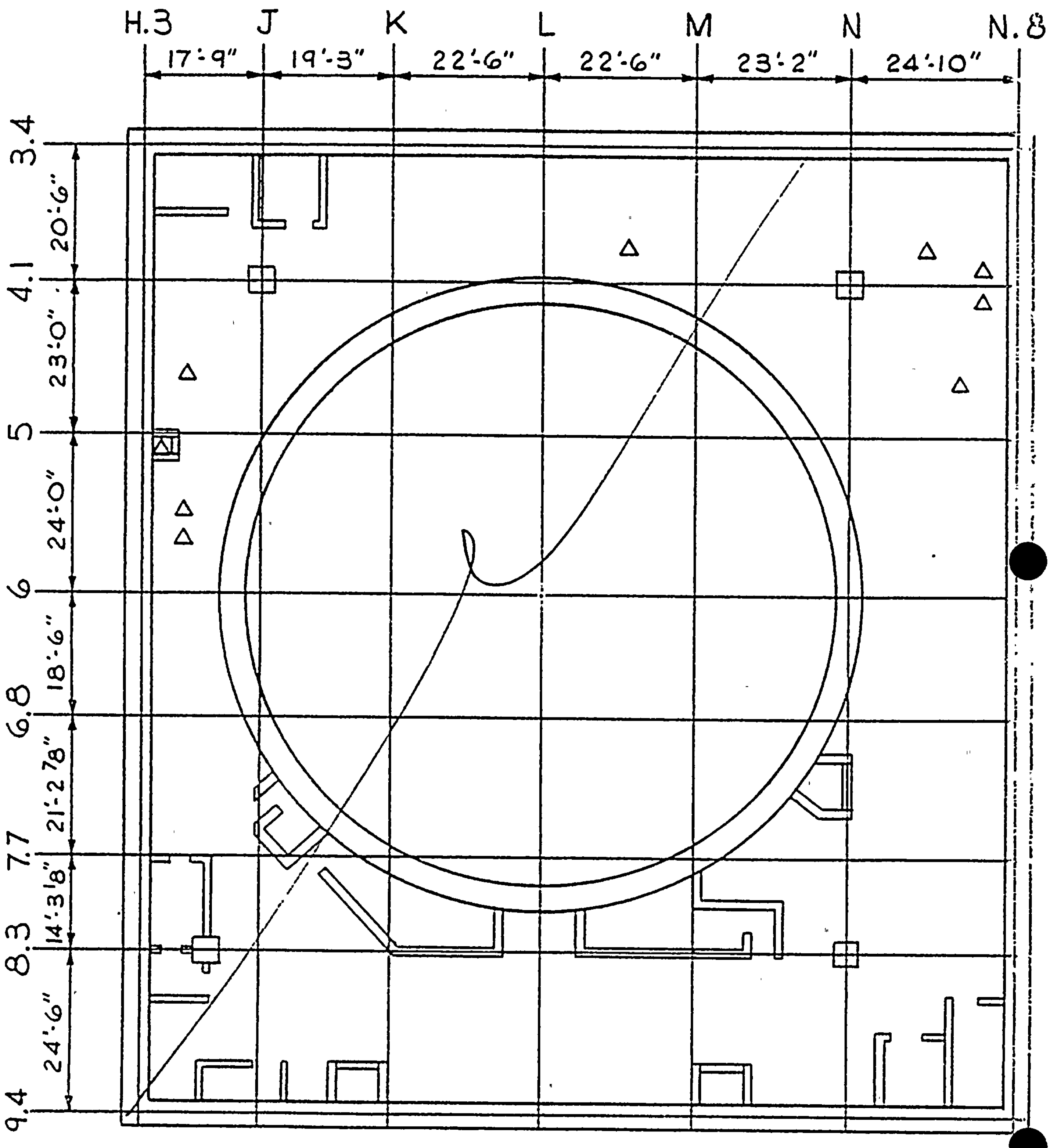
AMENDMENT NO. 25
June 1982



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 501'	FIGURE 3.6-11d

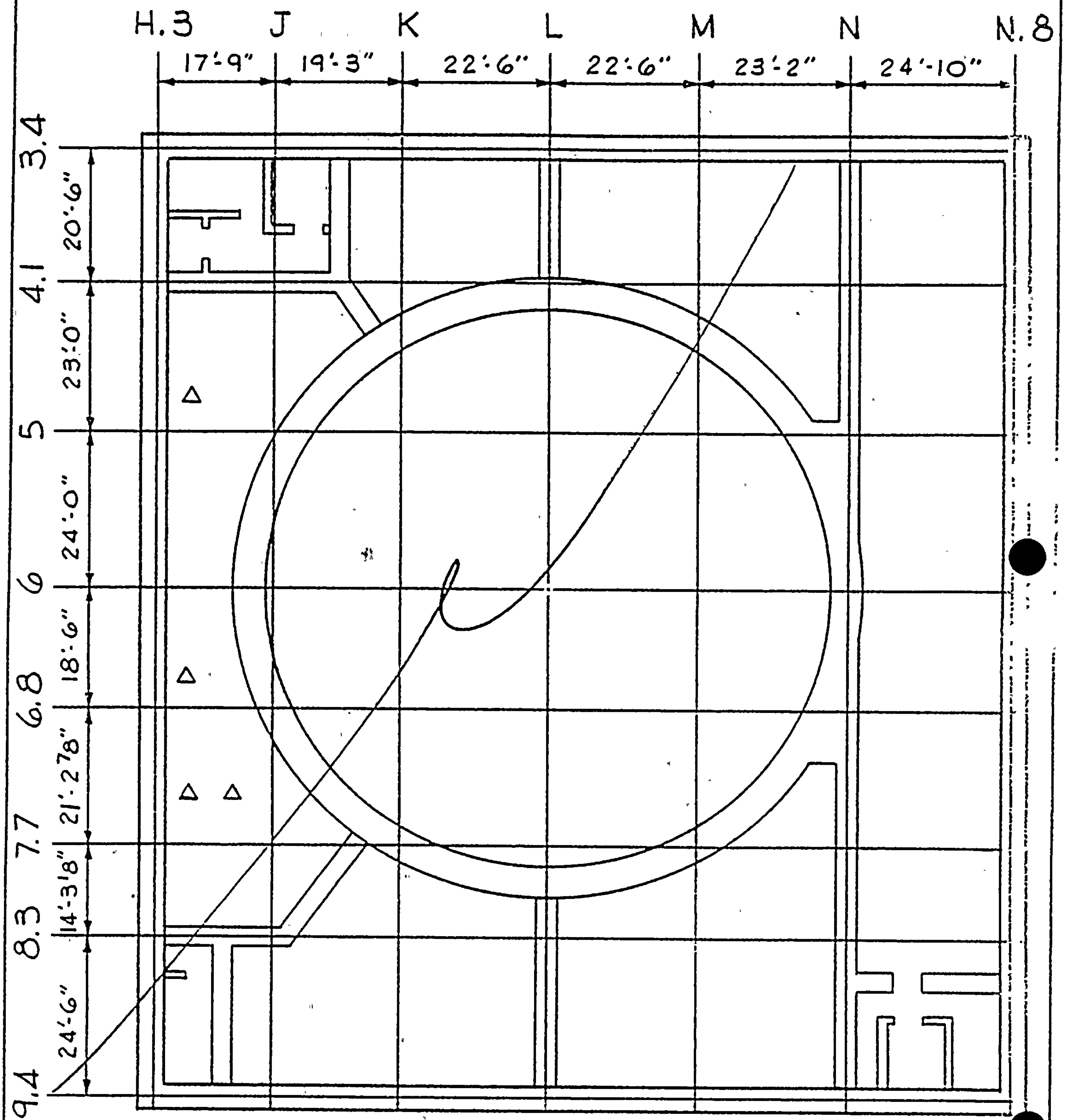
AMENDMENT NO. 25
June 1982



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 471'	FIGURE 3.6-41c

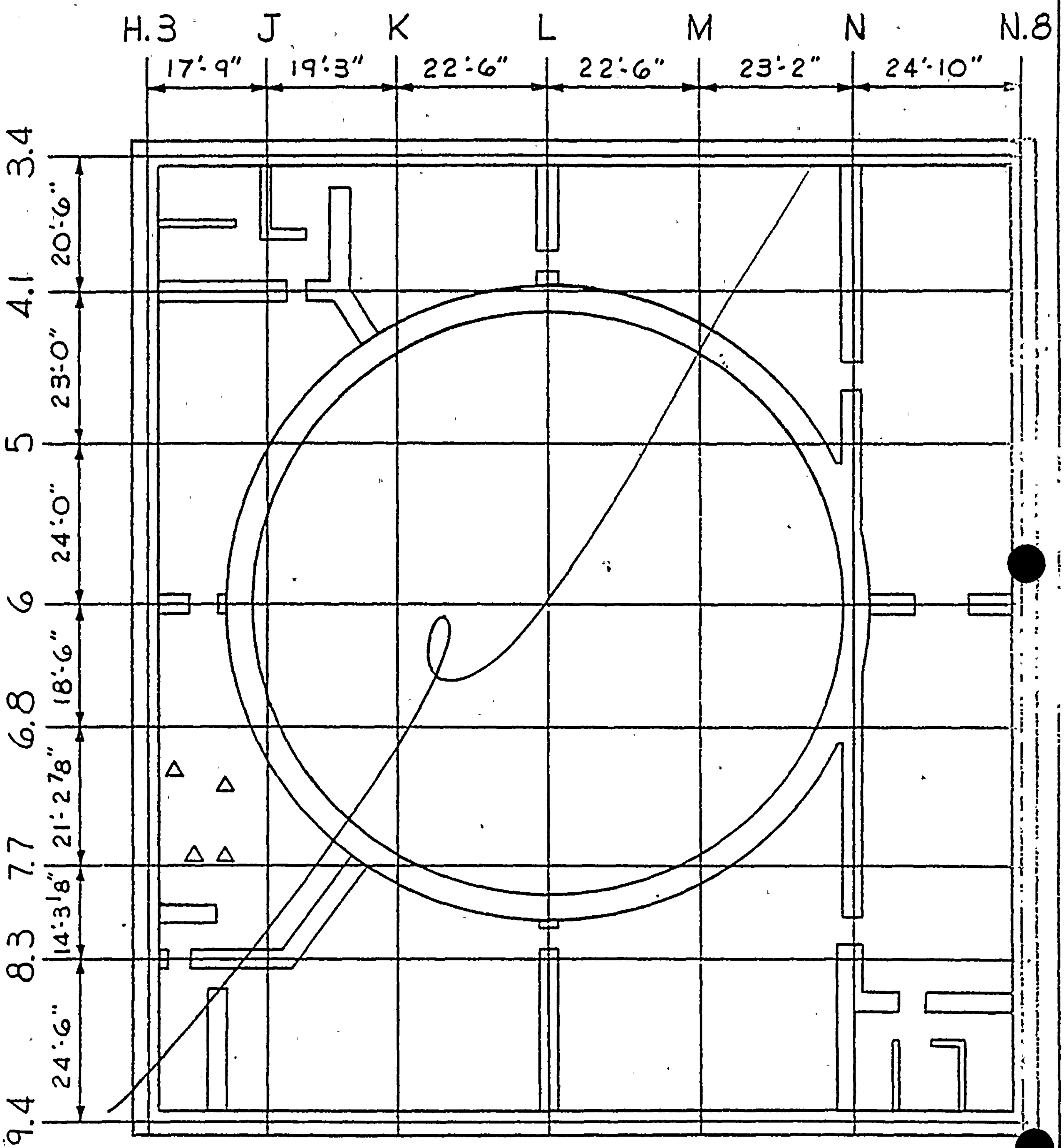
AMENDMENT NO. 25
June 1982



DELETE FIGURE

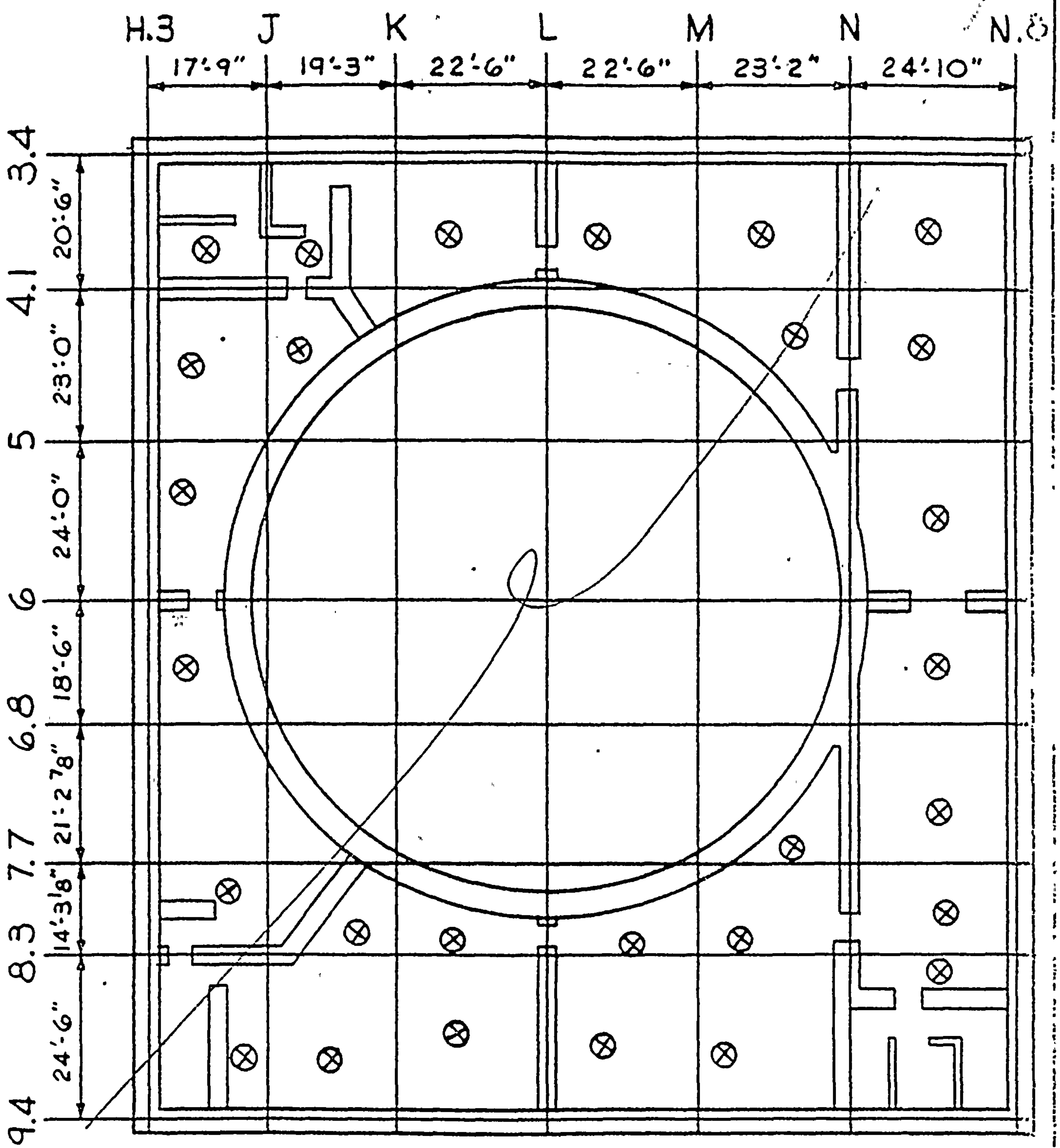
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 441'	FIGURE 3.6-4:b

AMENDMENT NO. 25
June 1982



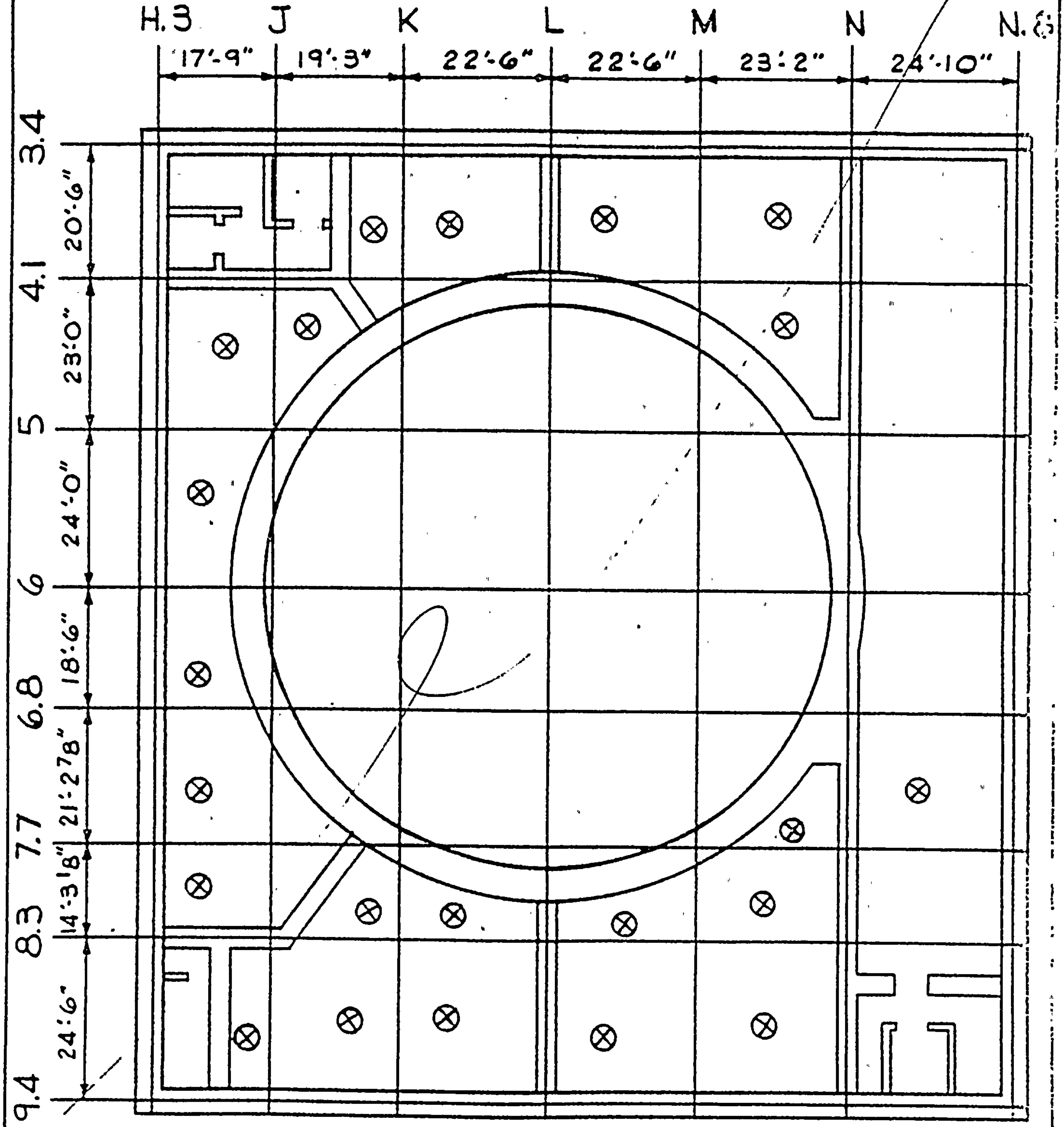
DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH ENERGY FLUID PIPING SYS. RUPTURE LOC. PLAN @ EL. 422'-3"	FIGURE 3.6-41



DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 422'-3"	FIGURE 3.6-42



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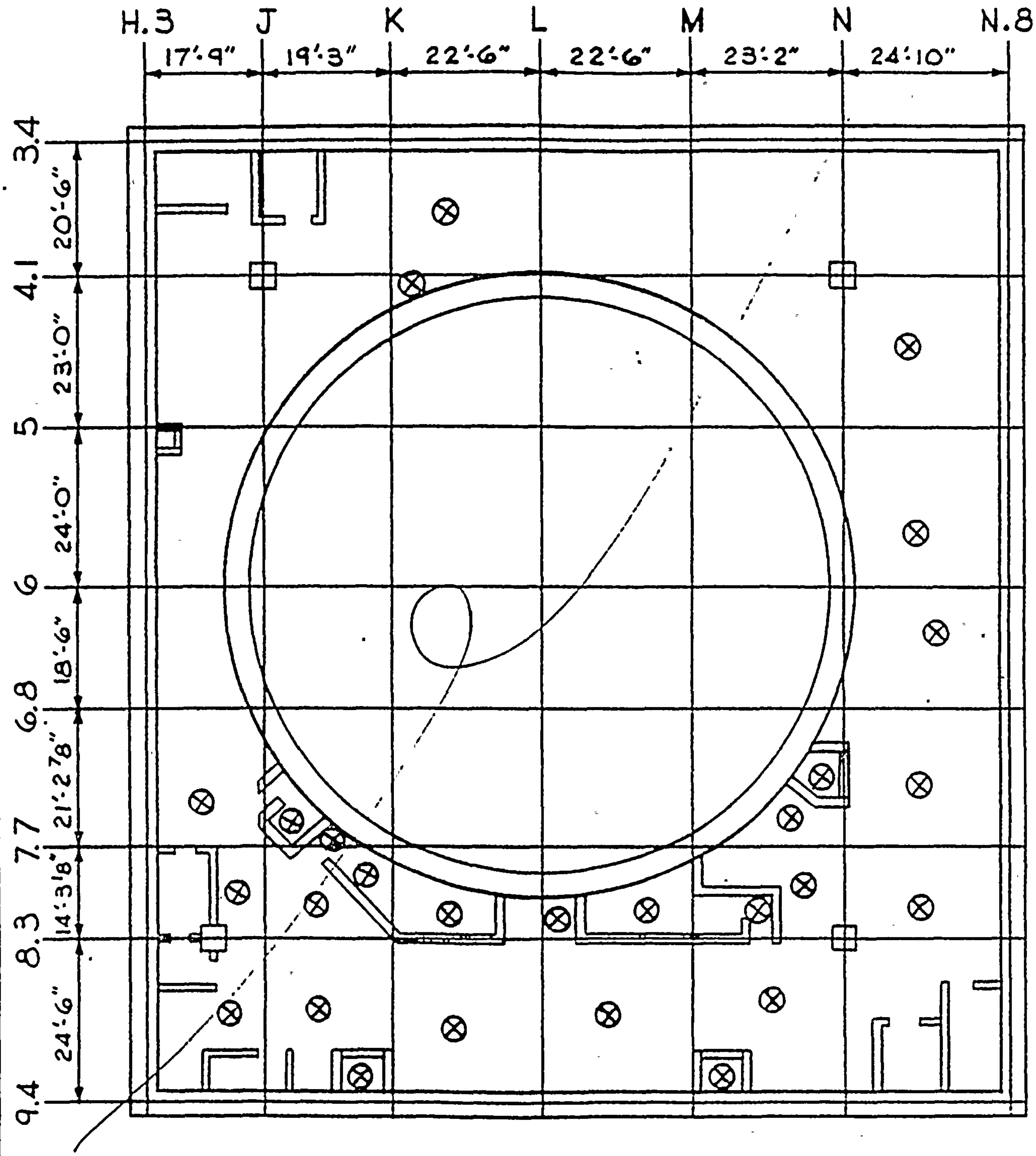
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 441'	FIGUR 3.6-4



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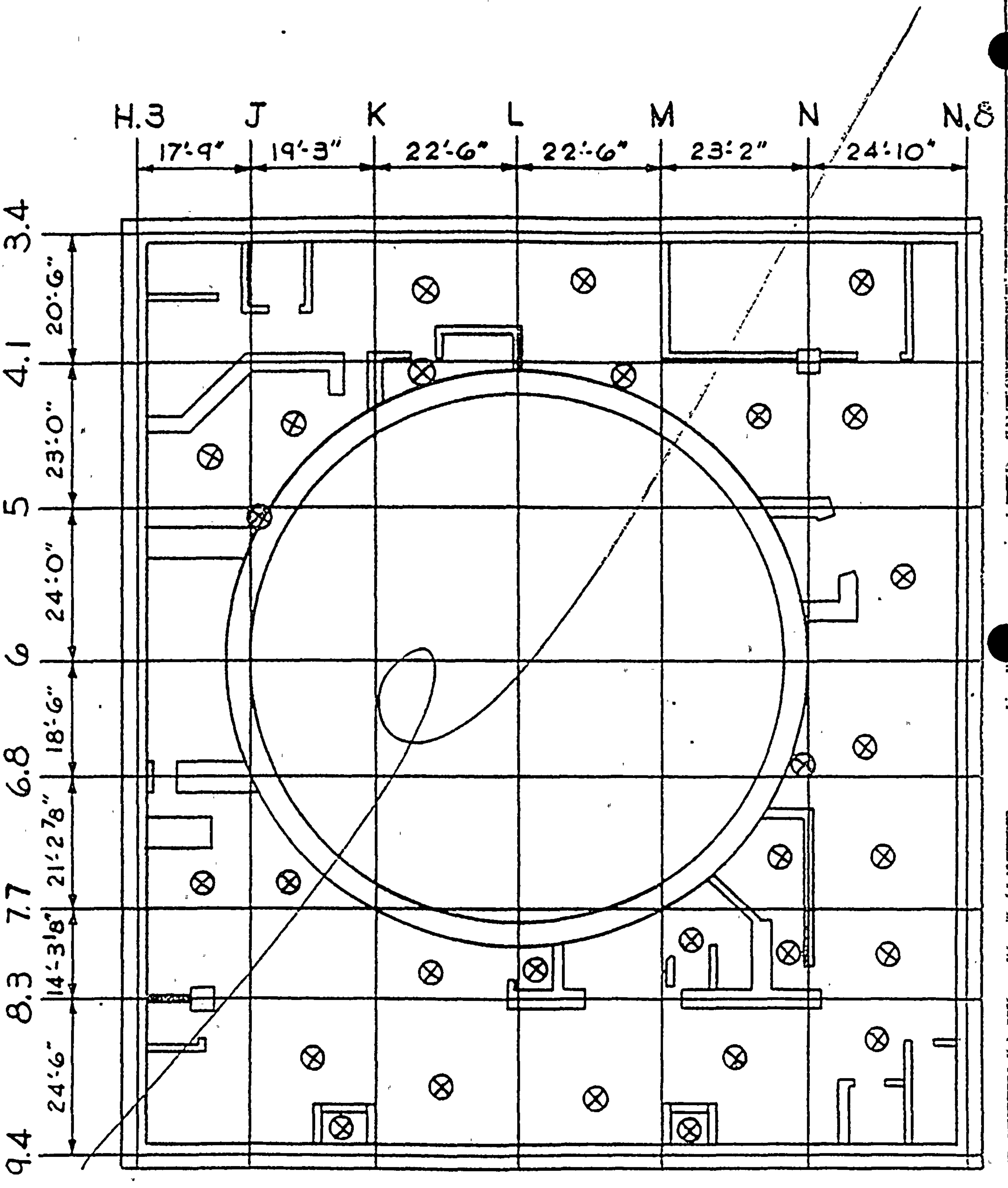
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 471'	FIGURE 3.6-42c



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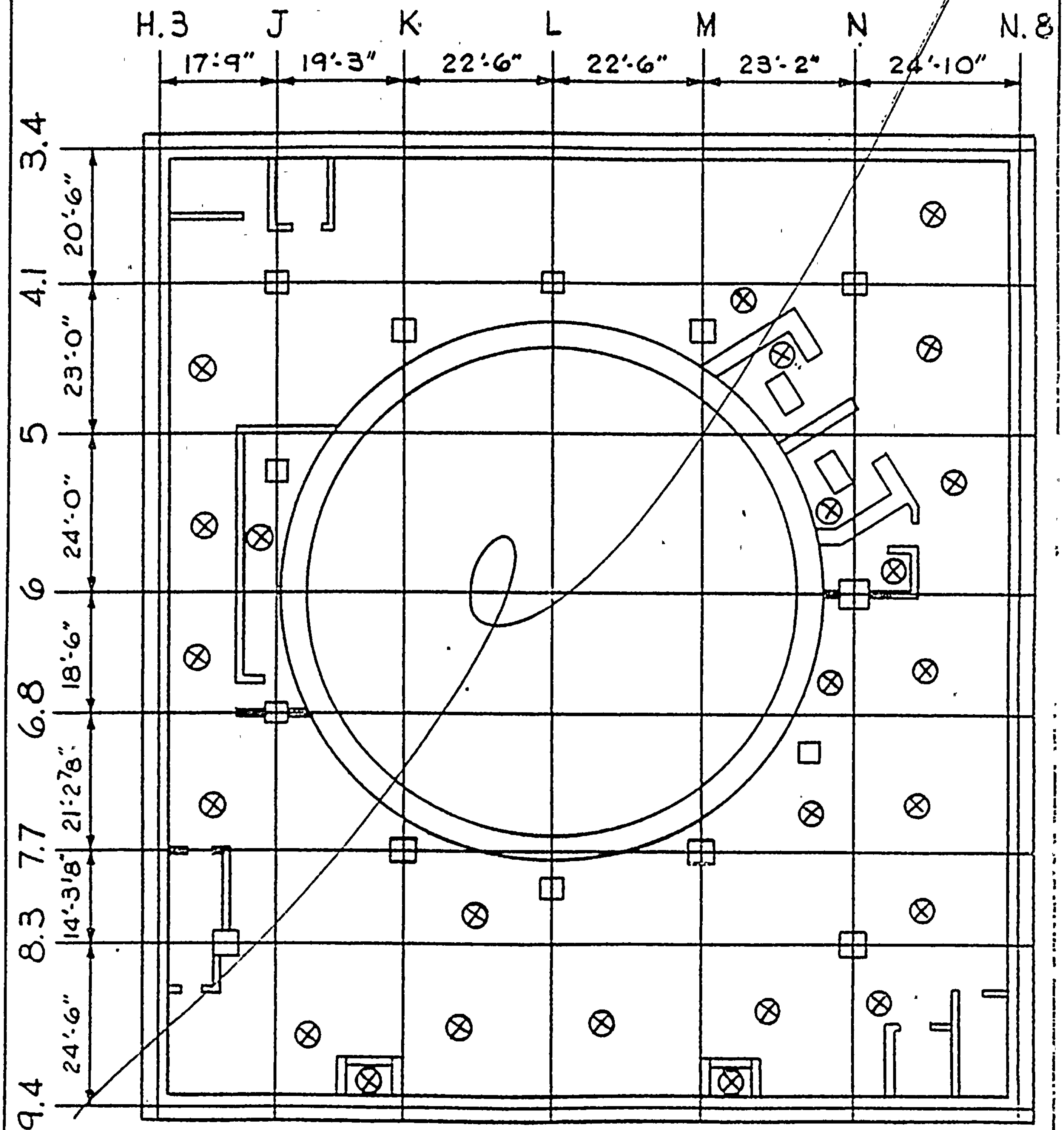
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 501'	FIGURE 3.6-42



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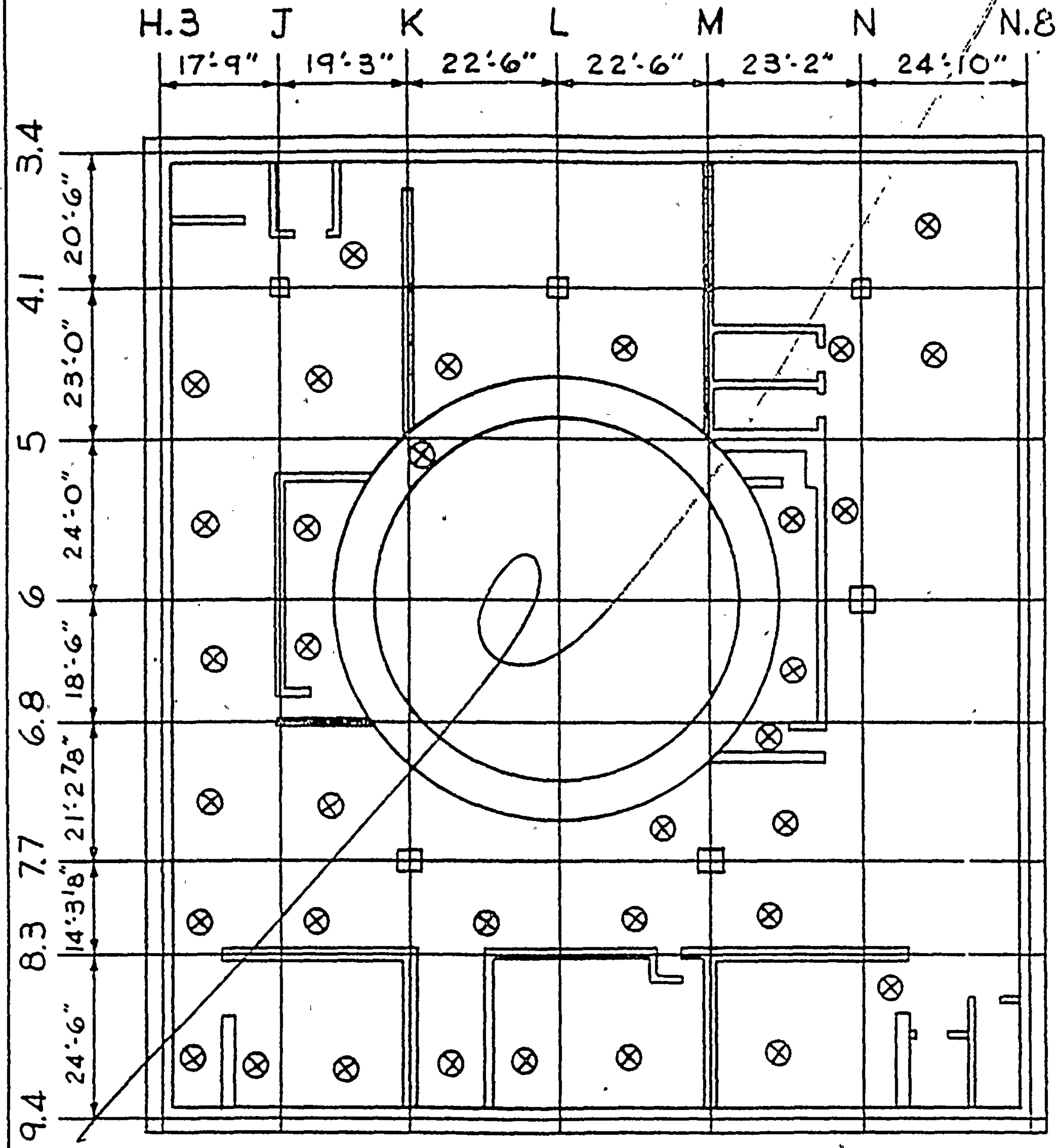
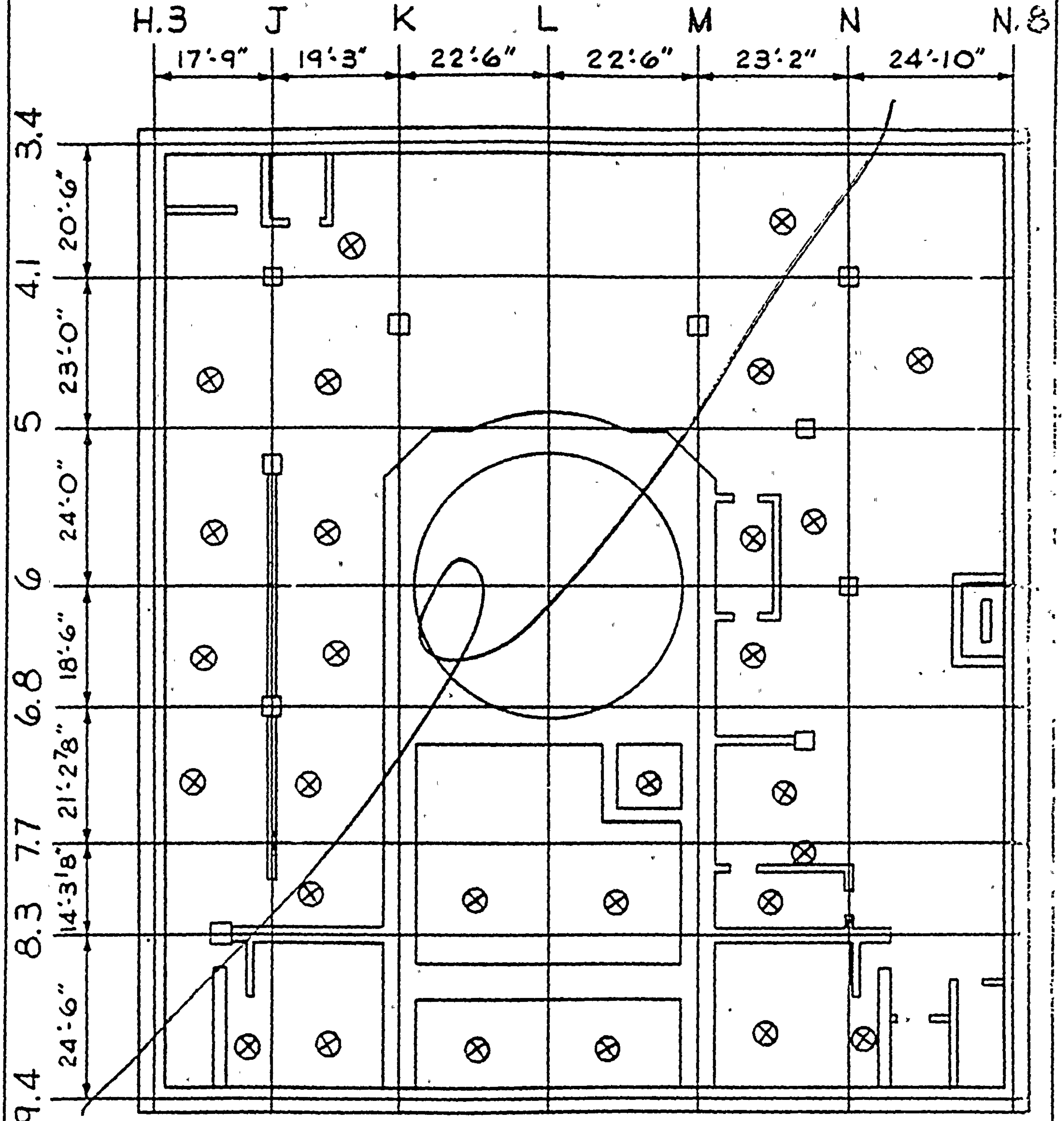
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 522'	FIGURE 3.6-42



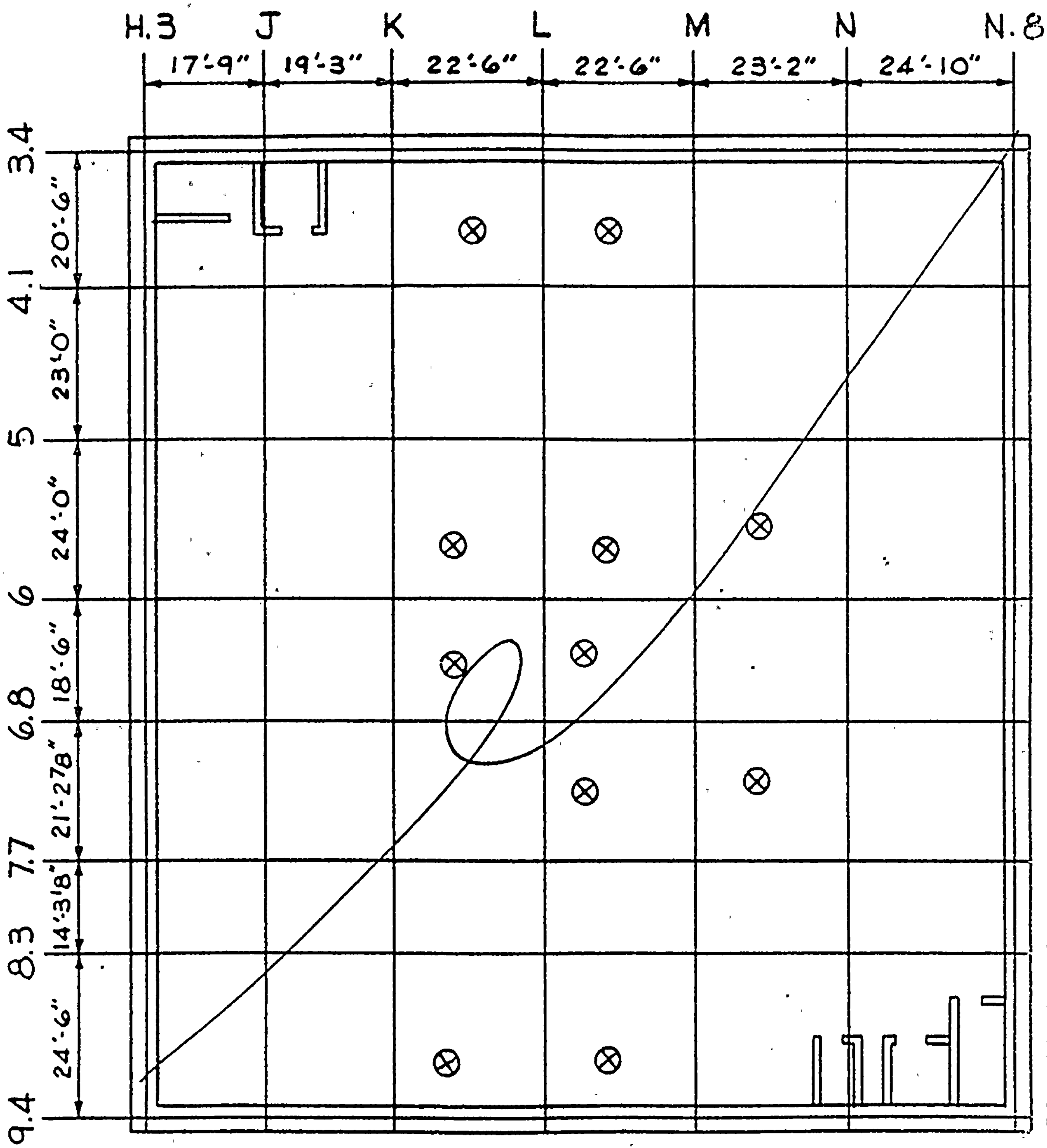
DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 548'	FIGURE 3.6-421



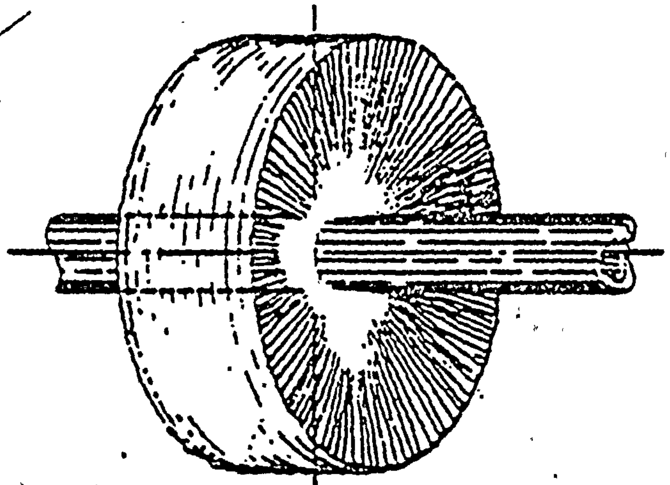
DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 572'	FIGU... 3.6-4...



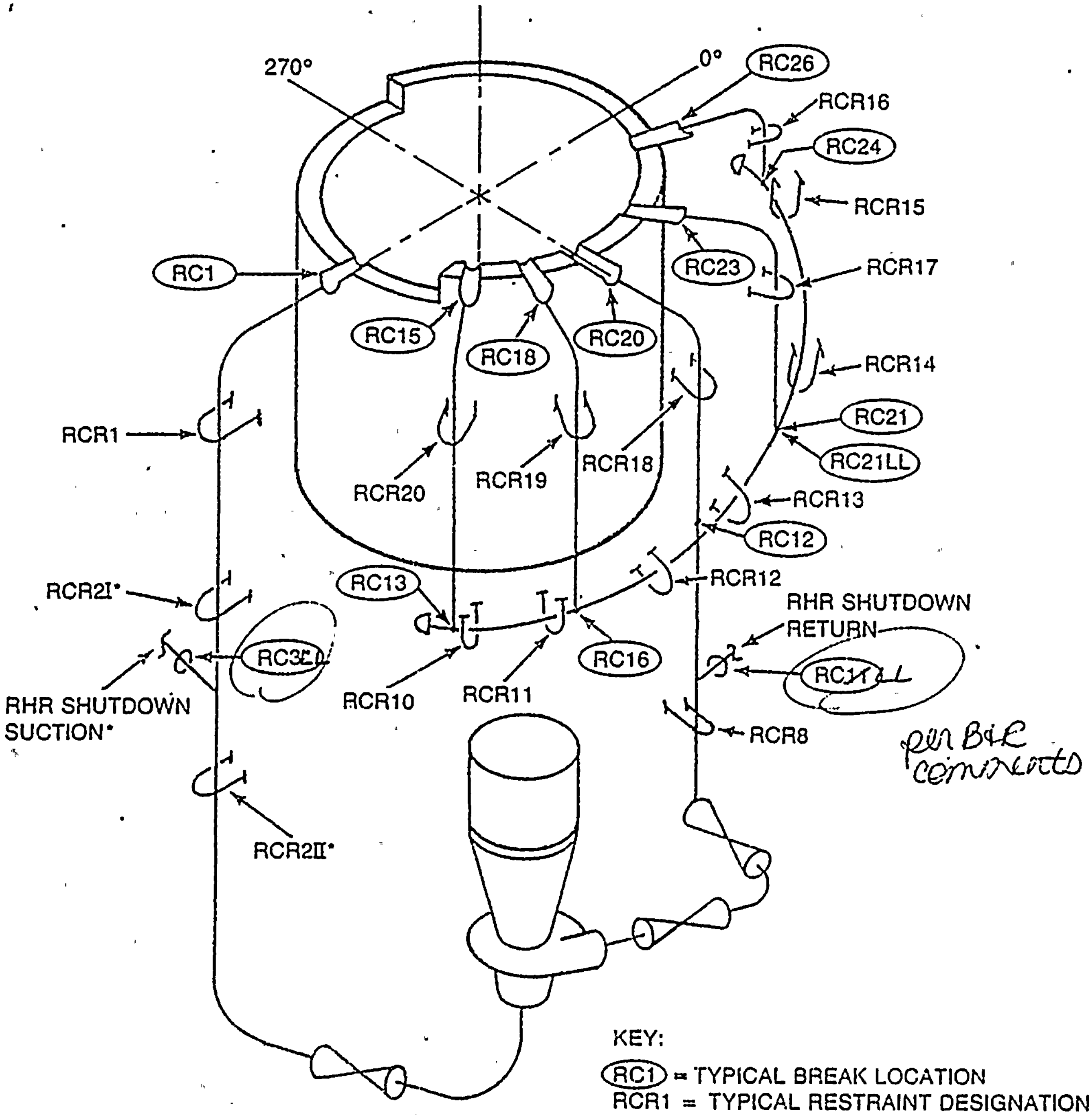
DELETE FIGURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MODERATE ENERGY FLUID PIPING SYSTEM RUPTURE LOC. PLAN @ EL. 606'-10 1/2"	FIGURE 3.6-42



(B) Jet From Circumferential Break with Ends Restrained
(FAN JET)

FIGURE 3.6-148



KEY:

RC1 = TYPICAL BREAK LOCATION
RCR1 = TYPICAL RESTRAINT DESIGNATION

SUFFIX "LL" INDICATES LONGITUDINAL BREAK
* INDICATES LOOP A ONLY

NOTES: (1) THIS FIGURE REPRESENTS LOOP A. LOOP B IS SIMILAR EXCEPT AS NOTED.
(2) SEE FIGURE 3.6-35b FOR RESTRAINT-BREAK LOCATION CORRELATION AND BREAK TYPES.
(3) ONLY THOSE RESTRAINTS THAT MAY ACT DURING THE POSTULATED BREAKS ARE SHOWN.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION PIPING SYSTEM	FIGURE 3.6-35a

BRSCN No. 83-46

Due 6/6/83

BURNS AND ROE S A R CHANGE NOTICE

(BRSCN)

Part I SAR section(s) affected: 3.6

Ref: WPBP-RO-83-11.3

SS-SCN 82-175

Part II Description of Need for Amendment: Review & concur with proposed changes due to New Jacobs Pipe Break Analysis made by Supply System and G.E.

Part III Are there any new commitments in change: YES ☐ NO ☒

Identify: __________

See attached pages for proposed revisions. Attach supporting documentation for information.

Part IV Approvals

Approvals indicate authorization to submit the proposed change to the client. Differing viewpoints should be resolved as much as possible before sign-off. Resolution of conflict should be explained in remarks.

	Signature	Date	Remarks
Licensing Lead	J. L. Brown	6/8/83	~~See comments on page 22~~ SEE FIGURE 3.5A ? TELECON W/ D. BOSI: B&R & SS CONCUR WITH WRITE-UP.
~~Group Supervisor~~			
Other	[signature]	6/10/83	
Appropriate Licensing Eng. or Sup.			
Project Licensing Supervisor	M.A. Conn	6/15/83	

*WNP-2 Approved Deviation

For INFO

Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

May 26, 2983
WPBR-RO-83-163
NS-L-02-JCA-83-050

Mr. J. A. Forrest
Project Manager
Burns and Roe, Inc.
601 Williams Blvd.
Richland, WA 99352

Dear Mr. Forrest:

Subject: CHANGES DUE TO NEW LOADS PIPE BREAK ANALYSIS (SCN 82-175 ATTACHED)

Please review and concur with the attached SCN 82-175 for incorporation into the Supply System's final amendment into the WNP-2 FSAR. The subject SCN also includes revisions to the FSAR by General Electric.

Please respond by June 8, 1983.

Very truly yours,

L T Harrold

L.T. Harrold
Assistant Director,
WNP-2 Engineering

JCA/mt
Attachment

cc: WS Chin - BPA
WG Conn - B&R RO
AI Cygelman - B&R Site
AN Kugler - B&R Site
TA Mangelsdorf - BECH
N Powell - BECH
JJ Verderber - B&R NY
WNP-2 Files
M Duer - B&R HAPO

WNP-2 SCN 82-125

SAR CHANGE NOTICE

SAR Section(s) Affected: 3.6 3.12

Description of Change: Changes due to New Loads Pipe Break Analysis

Reasons for Change: ______

This SCN satisfies OCI Log Commitment No.: N/A

This SCN commits to the following: ______

This SCN will be incorporated into Amendment No.: 31

See attached pages for original and/or revised SAR Section(s).

Approvals: Signature indicates authorization to file the subject change into an amendment.

	Signature	Date	Remarks
Lead Technical Reviewer(s) LTRs	DMBox *	3/29/83	Includes revisions provided by LAN's 51 & 70
W. K. Stockdale	W.K. Stockdale **	3/30/83	Fig 3.6-52, JUST ABOVE PAR. REF TO 3.12.13 SHOULD BE TO 3.12.33
Project ~~Licensing~~ Engrg. Manager	[illegible]	4/6/83	
Plant Operations Manager	[illegible]	4/15/83	
Project ~~Engineering~~ Licensing Manager	[illegible]	[illegible]	
Project QA Manager*	N/A		

*Applicable only for changes affecting Quality Assurance.

35 29 32 ok 51

* Forward to BRI for concurrence and incorporation into final write up

** Note GE revisions on page 3.6-23

SCO 12-175
Rossi/Lindblad

GENERAL ELECTRIC CO.
NUCLEAR POWER SYSTEMS DIVISION

LICENSING ACTION NOTICE
WPPSS NUCLEAR PROJECT NO. 2

Notice # 70 Rev. 0

Transmittal Date: February 18, 1983

Responds to: Letter from David Rossi 12-2-82

SUBJECT: New Loads Pipe Break Analysis

FSAR: 3.6 and 3.12 , Reference LAN #51.

NRC Question #:

Cindy
RECEIVED
MAR 4 - 1983
R. M. NELSON
MANAGER
WNP-2 PROJECT LICENSING

ACTION REQUIRED:

Attached are marked up copies of pages from the FSAR. These changes are in answer to questions from Mr. David Rossi.

In response to your question on Section 3.6.2.5.3.5; "Aren't the cases of Table 6.3-7 bounding? Therefore, this statement is moot. Suggest this be revised to reflect that Table 6.3-7 lists the bounding cases."; Table 6.3-7 cases are not bounding. The writeup in Section 3.6.2.5.3.5 should be left as transmitted in LAN #51.

Submitted by P.B. Kingston Date 2-18-83
P. B. Kingston (Licensing Engineer)

Reviewed by A.F. DeVault Date 2/23/83
A. F. DeVault (Project Engineer)

Approved by F.A. MacLean Date 2/23/83
F. A. MacLean (Project Manager)

Distribution:
1. Licensing Eng. 682
2. Projects 394
3. WPPSS (Original)
4. Burns & Roe (R.O.)

PK:cal/K11189
11/19/82

GENERAL ELECTRIC CO.
NUCLEAR POWER SYSTEMS DIVISION

LICENSING ACTION NOTICE
WPPSS NUCLEAR PROJECT NO. 2

Notice # 51 Rev. ________

Transmittal Date: August 12, 1982

Responds to: N/A

SUBJECT: New Loads Pipe Break Analysis

FSAR: Sections 3.6, 3.12

NRC Question #: N/A

ACTION REQUIRED:
Attached are the recommended FSAR changes of Section 3.6 and 3.12 to reflect the New Loads Pipe Break Analysis.

Please Note: Ge recommends that Burns & Roe review Section 3.6.2.5.3.6, items a,b,c for consistency with Section 6.2.4. It may be preferrable to replace the write-up here with a reference to Section 6.2.4.

Submitted by [signature] Date [illegible]
L. E. Santos (Licensing Engineer)

Reviewed by [signature] Date [illegible]
A. F. DeVault (Project Engineer)

Approved by [signature] Date [illegible]
F. A. MacLean (Project Manager)

Distribution:
1. Licensing Eng. 52.
2. Projects 39-
3. WPPSS (Original)
4. Burns & Roe (R.O.)

LS:hmc/1615
8/13/81

RECEIVED
AUG 17 1982
R. M. NELSON
MANAGER
WNP-2 PROJECT LICENSING

3.6.2 DETERMINATION OF BREAK LOCATIONS FOR DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Information concerning postulated break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high and moderate energy fluid system piping inside and outside of primary containment is presented in this section. The information presented in this section, and in 3.6.1, confirms that the requirements for the protection of structures, systems, and components relied upon for safe reactor shutdown, or to mitigate the consequences of a postulated pipe break, have been met.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following section establishes the criteria for the location and configuration of postulated breaks and cracks in high energy and moderate energy piping systems both inside and outside of primary containment.

High-energy fluid systems are defined as those systems, or portions of systems, that during normal plant conditions[a] are maintained pressurized under conditions where either one or both of the following are met:

a. Maximum temperature exceeds 200°F

b. Maximum pressure exceeds 275 psig

Moderate energy fluid systems are defined as those systems, or portions of systems, that during normal plant conditions are pressurized under both of the following conditions:

a. Maximum temperature is 200°F or less.

b. Maximum pressure is 275 psig or less.

(a) Normal plant conditions is defined as the plant operating conditions during reactor startup, operation at power, HOT STANDBY or COOL-DOWN TO reactor cold shutdown, but excluding test modes. ~~Since hot standby occurs less than one percent of the total operating time, this condition is excluded from consideration of high energy piping.~~

BRI! → Please review for compatibility with your pipe analysis

~~supply systems~~

~~We prefer to leave the definition of normal plant conditions as the NRC has accepted it (i.e. no changes). However, if the words "hot standby" and "cooldown" are inserted, the last sentence should remain incorporated to make clarity to the hot standby mode.~~ 8-17-75

Piping systems are classified as moderate-energy systems, when they operate as high energy piping for only short periods in performing their system function. For the major operational period they qualify as moderate-energy fluid systems. An operational period is considered "short" if the total fraction of time that the system operates, within the pressure-temperature conditions specified for high-energy fluid system, is less than approximately two percent of the time period that the system operates as a moderate energy fluid system, or less than one percent of the normal operating life span of the plant.

A postulated pipe break is defined as a sudden, gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break), or as development of a sudden longitudinal ~~uncontrolled~~ crack (longitudinal split). These are postulated for high energy fluid systems only. For moderate energy fluid systems, pipe rupture is confined to postulation of ~~[illegible]~~ LEAKAGE cracks in piping and branch runs. These cracks affect the surrounding environmental conditions only, and do not cause jet impingement or uncontrolled whipping of the pipe.

A moderate energy piping system crack is not postulated simultaneously with a high energy piping system break, nor is any pipe break or crack outside containment postulated concurrently with a pipe break or crack inside containment.

Postulated pipe break locations are selected as described herein; and are based on the guidelines provided in Regulatory Guide 1.46, Rev. 0; the U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position APCSB 3-1, Appendix B; and as expanded in NRC Branch Technical Position MEB 3-1 for piping inside and outside primary containment.

3.6.2.1.1 Postulated Pipe Break Locations in High Energy Fluid System Piping Not in the Containment Penetration Area

Pipe breaks (not including leakage cracks) are postulated at locations as indicated below:

82-175

3.6.2.1.1.1 Postulated Pipe Break Locations in ASME Section III Class 1 Piping Runs

a. The terminal ends[a] of the pressurized portions of the run.

b. Intermediate locations of postulated pipe breaks are selected by application of one of the following sets of rules:

(1) Pipe break is postulated at each location of significant change in flexibility, such as pipe fittings (elbows, tees and reducers), and circumferential connections to valves and flanges.

(2) Based on stress and fatigue analysis, as calculated according to ASME Code Section III Sub-article NB-3600, no break is postulated if any of the following applies:

(a) S_n[b] does not exceed $2.4S_m$[c]

(b) S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U)[d] does not exceed 0.1

(a) Terminal ends are extremities of piping runs that connect to structures, equipment, or pipe anchors that ~~are assumed to~~ act as rigid constraints to free thermal expansion of piping. A branch connection to a main piping run is a terminal end for a branch run, except when the nominal size of the branch is at least one half that of the main piping run, and the branch and main runs are modeled as a common piping system during the piping stress analysis.

(b) S_n is the primary plus secondary stress intensity range, as calculated by use of Equation (10) of ASME Code Section III Subsection NB, Paragraph NB 3653.1 between any two load sets (including the zero load set) for normal and upset plant conditions, including an OBE event transient.

(c) S_m is the design stress intensity, as described in ASME Code Section III Subsection NB Paragraph NB 3229.

(d) U is the Cumulative Usage Factor that indicates the total fatigue damage as calculated by the procedure in ASME Code Section III Subsection NB, Paragraph NB 3653.

82-175

(c) S_n exceeds $3S_m$ but S_e[(e)] and S_r[(f)] are each less than $2.4S_m$, and U does not exceed 0.1

c. When two or more intermediate locations cannot be determined by stress or usage factor limits as described above, then intermediate locations of significant change in flexibility are chosen as postulated pipe rupture locations on a reasonable basis for each piping run[(a)] or branch run[(b)] as necessary to provide protection. A reasonable basis as used herein considers the locations of highest computed value of stress, S_n. Cumulative usage factor is also considered. As a minimum, two intermediate locations are chosen for each piping run or branch run, except for a piping run having only one change in direction in which case only one intermediate break is postulated. Intermediate breaks are not postulated in sections of straight pipe, where there are no pipe fittings, valves, or flanges.

(e) S_e is the nominal value of expansion stress as calculated by use of Equation (12) of ASME Code Section III Subsection NB, Paragraph NB 3653.6(a).

(f) S_r is the range of primary plus secondary membrane plus bending stress intensity, exlcuding thermal bending and thermal expansion stresses as calculated by use of Equation (13) of ASME Code Section III Subsection NB.

(a) A piping run is defined as piping which interconnects equipment such as pressure vessels, pumps, and other equipment that act as rigid constraints to free thermal expansion of piping.

(b) A branch run is defined as differing from a pipe run only in that it originates at a piping intersection as a branch of the main pipe run, except that branch lines which are included with main run piping in the stress analysis computer mathematical model and are shown to have significant effect on the main run behavior are considered part of the main run.

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Piping and electrical penetration details are discussed and shown in 3.8.6.

The stress criteria for postulating breaks in containment penetration piping between isolation valves is given in 3.6.2.1.2.1 and 3.6.2.1.2.2.

Welded attachments, for pipe supports or other purposes, to these portions of piping are avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of 3.6.2.1.2. In addition, the number of circumferential and longitudinal piping welds and branch connections are minimized.

Any pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) are designed such that they are not welded directly to the outer surface of the piping except where such welds are 100 percent volumetrically examinable while in service, and a detailed stress analysis is performed to demonstrate compliance with the limits of 3.6.2.1.2.

Tunnel structures surrounding the primary containment penetration piping are designed for the thermal and pressure loads of a through-wall leakage crack regardless of crack postulation requirements. Refer to 3.6.1.20 for further discussion.

Access for inservice inspection of welds in high energy (hot type) containment penetration assemblies is described in 3.8.6.1.1. All required inservice inspection locations are accessible.

3.6.2.1.3 Postulated Through Wall Leakage Crack Locations in High and Moderate Energy Fluid Systems

In high energy piping systems consisting of ASME Code Section III Class 1 piping, (including fluid system piping between primary containment isolation valves) cracks are not postulated, ~~provided the primary plus secondary stress intensity range, S_n, does not exceed 1.2 S_m, for transients resulting from normal plant conditions. There are no moderate energy piping systems consisting of ASME Code Section III Class 1 piping.~~ for example, no cracks are postulated in the recirculation piping system.

In ~~high energy and~~ moderate energy piping systems consisting of ASME Code Section III Class 2 and 3 piping and moderate energy non-nuclear piping, including fluid system piping between primary containment isolation valves, cracks are not

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postulated provided the stress range of 0.4 ($1.2S_h$[a] + S_A[b]) is not exceeded for the load combination which includes the effects of pressure, weight, other sustained loads and occasional loads such as the operating basis earthquake, and thermal expansion loads. Since all piping in structures housing safety-related systems are supported and controlled as Seismic Category I systems regardless of service, the criteria for postulated cracks is the same as above for all systems.

3.6.2.1.4 Types of Breaks and Cracks Postulated in High Energy and Moderate Energy Fluid System Piping

3.6.2.1.4.1 Breaks in High Energy Fluid System Piping

The following types of breaks are postulated in high energy fluid system piping:

a. No breaks need be postulated in piping having a nominal diameter less than, or equal to one inch.

b. Circumferential breaks are postulated only in piping exceeding a one inch nominal pipe diameter.

c. Longitudinal splits are postulated only in piping having a nominal pipe diameter equal to or greater than 4 inches.

d. Longitudinal splits are not postulated at terminal ends.

e. At each of the postulated break locations, consideration is given to the occurrence of either a longitudinal split or circumferential break. Both types of breaks are considered, if the maximum stress ranges in the circumferential and axial directions are not significantly different. Only one type break is considered as follows:

(a) S_h is the allowable stress at maximum (hot) temperatures defined in ASME Code Section III, Article NC 3611.2

(b) S_A is the allowable stress range for thermal expansion, as defined in ASME Code Section III, Article NC 3611.2.

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(2) If this type of analysis indicates that the maximum stress range, in the circumferential direction, is at least 1.5 times that in the axial direction, only a longitudinal split is postulated.

f. Where break locations are selected without the benefit of stress calculations, circumferential breaks are postulated at the piping welds to each fitting, valve or welded attachment. Postulated longitudinal splits are described in FSAR 3.6.2.1.4.1.i.

g. For a longitudinal split, the break area is assumed to be equal to cross-sectional flow area of the pipe.

h. For circumferential breaks, pipe whipping is assumed to occur in the plane defined by the piping configuration, and is assumed to cause pipe movement in the direction of the jet reaction.

i. A longitudinal break is assumed to result in an axial split without severence and to be oriented at any point about the circumference of the pipe, or alternately, at the point(s) of highest stress as indicated by a detailed stress analysis. If a postulated break location is at a non-axisymmetric fitting, such as a tee or elbow, the split is assumed to be oriented (but not concurrently) on each side of the fitting at its center, perpendicular to the plane of the fitting and is assumed to cause pipe movement in the direction of the jet reaction.

j. For a circumferential break, the dynamic force of the jet discharge at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure, as modified by an analytically or experimentally determined thrust coefficient. A circumferential break is assumed to result in pipe severence with full separation, except as limited by structural design features. The break is assumed to be oriented perpendicular to the

AMOUNTING TO AT LEAST A ONE PIPE DIAMETER LATERAL DISPLACEMENT OF THE RUPTURED PIPING SECTIONS.

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longitudinal axis of the pipe. Line restrictions, flow limiters, and the absence of energy reservoirs are accounted for, in the calculation of the design jet discharge.

3.6.2.1.4.2 Cracks in High Energy and Moderate Energy Fluid System Piping

The following controlled, through-wall leakage cracks, are postulated in high energy and moderate energy fluid systems (or portion of systems):

a. Cracks are postulated in fluid systems or portions of systems whose size exceeds a nominal pipe diameter of one inch.

b. Fluid flow, from the postulated crack, is based on a circular opening of area equal to that of a rectangle one-half pipe-diameter in length and one-half pipe wall thickness in width.

c. The flow from the postulated crack is assumed to result in an environment that wets all unprotected components within the compartment, with subsequent flooding in the compartment and communicative compartments. Flooding effects are determined on the basis of a conservatively estimated time period required to affect corrective action.

3.6.2.1.5 Protection Criteria for the Effects of Pipe Break

Protection from the effects of a whipping pipe due to a pipe break is provided where necessary. Protection from pipe whip need not be provided if any one of the following conditions exists:

a. The piping is classified as moderate energy piping.

b. Following a single postulated pipe break, piping for which the unrestrained movement of either end of the ruptured pipe, in the direction of the jet reaction about a plastic hinge, formed within the piping, cannot impact any structure, system or component important to safety.

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(1) The transient forcing functions OCCUR at points along the pipe ~~results~~ from the propagation of waves (wave thrust) along the pipe, and AT THE BROKEN END from the reaction force due to the momentum of the fluid leaving the end of the pipe (blowdown thrust).

(2) The waves cause various sections of the pipe to be loaded with time-dependent forces. It is assumed that the pipe is one-dimensional in that there is no attenuation or reflection of the pressure waves at bends, elbows, and the like. Following the rupture, a decompression wave is assumed to travel from the break at a speed equal to the local speed of sound within the fluid. Wave reflections ~~could~~ occur at the break end, ~~changes in direction of piping~~, and the pressure vessel END until a steady flow condition is established. ~~Vessel and~~ free-space AND VESSEL conditions are used as boundary conditions. The blowdown thrust causes a reaction force perpendicular to the plane of the pipe break, REACHING A FINAL STEADY STATE VALUE.

(3) The initial blowdown force on the pipe is taken as the sum of the wave and blowdown thrusts and is equal to the vessel pressure (P_o) times the break area (A). After the initial decompression period (i.e., the time it takes for a wave to reach the first change in direction), the force is assumed to drop off to the value of the blowdown thrust (i.e., $0.7\ P_oA$).

(4) Time histories of transient pressure, flow rate, and other thermodynamic properties of the fluid can be used to calculate the blowdown force on the pipe using the following equation:

$$F = \left\{ (P - P_a) + \frac{\rho u^2}{g} \right\} A$$

where:

F = Blowdown Force

P = Pressure at exit plane

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no change

P_a = Ambient pressure

u = Velocity at exit plant

ρ = Density at exit plane

A = Area of break

g = Gravitational constant

(5) Following the transient period, a steady-state period is assumed to exist. Steady-state blowdown forces are calculated, considering frictional effects. For ~~saturated steam~~ THE RECIRCULATION SYS., these effects reduce the blowdown forces from the theoretical maximum of 1.26 P_oA. The method of accounting for these effects is presented in Reference 3.6-3. For sub-cooled water, a reduction from the theoretical maximum of 2.0 P_oA is found through the use of Bernoulli's and other standard equations, such as Darcy's equation, which account for friction.

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b. The following is an alternate method for calculating blowdown forcing functions.

The computer code RELAP3 (Reference 3.6-9) is used to obtain exit plane thermodynamic states for postulated ruptures (see 3.12.11 for further discussion of RELAP3). Specifically, RELAP3 calculates exit pressure, specific volume and mass rate. From these data the blowdown reaction load is calculated using the following relation:

$$\frac{T}{A_E} = P_E - P_\infty + \frac{G_E^{\,2}\,\bar{V}_E}{g_c}$$

$$R = -\frac{T}{A_E} \times A_E$$

where:

$\frac{T}{A_E}$ - thrust per unit break area

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P_E - exit pressure

P_∞ - receiver pressure

G_E - exit mass flux

$\bar{v}_E$ - exit specific volume

g_c - gravitational constant

R - Reaction force on the pipe

3.6.2.2.2 Analytical Methods to Define Response Models

3.6.2.2.2.1 General Description of Analytical Methods

The prediction of time-dependent and steady-thrust reaction loads caused by blowdown of sub-cooled, saturated, and two-phase fluid from a ruptured pipe, is used in the design of piping systems and in the evaluation of dynamic effects of pipe breaks. A detailed discussion of the analytical methods employed to compute these blowdown loads are given in 3.6.2.2.1. The analytical methods used to account for this loading are discussed below.

3.6.2.2.2.2 Dynamic Analysis of the Effects of Pipe Rupture

a. Criteria

(1) Analysis is performed for each postulated pipe break.

(2) The analysis includes the dynamic response of all components of the system including the pipe, pipe whip restraints and all structures required to transmit loading to foundation. The structures are analyzed for a suddenly applied force in conjunction with impact and rebound effects due to gaps between piping and pipe whip restraints.

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(3) The analytical model adequately represents the mass/inertia and stiffness properties of the system.

(4) Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration, and to cause pipe movement in the direction of the jet reaction.

(5) Piping contained within the broken loop, is no longer considered part of the reactor coolant pressure boundary (RCPB). Plastic deformation in the pipe is considered as a potential energy absorber. Limits of strain are imposed ~~which are~~ similar to ~~strain~~ pipe whip ~~levels allowed in~~ restraint ~~plastic mem-bers~~ material (see 3.6.2.2.3.2). Piping systems are designed so that plastic instability ~~does~~ would not occur in the pipe at the design dynamic and static loads, unless damage studies are performed which show that the consequences ~~do~~ could not result in the direct damage of any essential system or component.

(6) Components, such as vessel safe ends and valves, which are attached to the broken piping system and do not serve a safety function or whose failure would not further excalate the consequences of the accident, are not designed to meet ASME Code require-ments for essential components under faulted loading. However, if these components are required for safe shutdown, or if they serve a safety function to protect the structural integrity of an essential component, then these components are designed to Code limits for faulted conditions and to the limits necessary to ensure operability.

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b. Analytical Models

(1) Lumped-Parameter Analysis Model: Lumped mass points are interconnected by springs to take into account for the effects of inertia and stiffness inherent in the system, and time histories of the responses are computed by numerical integration to account for gaps and inelastic effects. This analytical method is discussed in detail in Reference 3.6-4.

(2) Energy-Balance Analysis Model: Kinetic energy, generated during the first quarter cycle movement of the ruptured pipe as imparted to the piping/restraint system through impact, is converted into equivalent strain energy. Deformations of the pipe and the restraint are compatible with the level of absorbed energy.

(3) Pipe whip restraints, for the reactor recirculation system, are designed by the NSSS supplier. The analytical method utilized for this design is the computer program PDA which is described in Reference 3.6-4 and further discussed in 3.12.33. Pipe whip restraints for all other piping systems, requiring such protection, are designed by the architect/engineer; The method described in c., (below) is utilized for this pipe whip restraint design. ¶ t

c. Simplified Dynamic Analysis

(1) In order to simplify dynamic analysis the following conservative assumptions are utilized:

(a) The entire structure including pipe, restraint linkage, support beams and major structure to foundation connections absorb energy by elastic, elasto-plastic, or plastic deformation. In order to provide a simplified dynamic mathematical model, one member is generally considered to absorb all the energy. This member is classified as an energy

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Reference 3.6-6 provides the ductility ratio that corresponds to collapse (μ_c). For structural steel members, these values vary, with upper limits in the order of 20 to 30 and up (for very ductile structures). For WNP-2, the maximum permissible ductility ratio is limited to 50% of (μ_c), except that energy absorbing members in direct contact with primary containment are limited to 5% of (μ_c). For WNP-2, only steel members are utilized as energy absorbing members, as defined in 3.6.2.3.3.2.d. The maximum values of (μ_c), for various structural components, are given in Table 3.6-1.

(i) The equation derived in Figure 3.6-2 accounts for a suddenly applied, constantly maintained force, in conjunction with a kinetic energy of impact on the resisting member. Total transfer of energy is implied. This is combined with the constantly maintained force (from ruptured piping blowdown) on the restraint structure. This assumption is consistent with a zero coefficient of restitution (full plasticity), and is a conservative assumption.

With regard to rebound, it should be noted that if a coefficient of restitution of unity is assumed (full rebound), there is zero kinetic energy transfer to the restraint structure.

If a coefficient of restitution less than unity is assumed (partial rebound), there is a partial amount of kinetic energy transfer to the restraint structure.

A coefficient of restitution of zero, conservatively assumed in the application of the equation mentioned above,

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gives zero rebound with 100% kinetic energy transfer to the restraint structure.

It should also be noted, that the assumption of a suddenly applied, constantly maintained force, as used in the equation mentioned above is conservative with respect to rebound. Rebound implies a finite time of short duration contact with the restraint structure, in contrast to the infinite time assumed.

(3) Actual structural resistance, for the above structures, is determined by methods of limit analysis using a dynamic yield strength, as defined in 3.6.2.2.3.1.

3.6.2.2.3 Material Properties Under Dynamic Loads

3.6.2.2.3.1 Dynamic Yield Strength

To account for the rapid strain rate effects, dynamic yield strength is utilized. This phenomenon is documented in References 3.6-6 and 3.6-7. Material tests have shown a consistent increase in yield strength under rapid loading. Under rapid strain rate, carbon steel yield strength consistently improves by more than 40%. High strength alloy steel displays a somewhat smaller improvement. For WNP-2, a conservative dynamic yield strength of 110% of minimum static yield strength, at the specified operating temperature, is utilized.

3.6.2.2.3.2 Maximum Strain of Tension Members

~~Pure tension members, such as U-Bars shown on Fig. 3.6-4 which constitute pipe whip limit stops, are permitted to deform a maximum of 50% of the minimum uniform strain, during energy absorption.~~ See Insert

3.6.2.2.3.3 Maximum Deformation of Flexural Members

Deformations of energy absorbing flexural support members are generally limited to 50% of that deformation which corresponds to structural collapse, except that deformation of energy absorbing members in direct contact with the primary containment vessel is limited to 5% of that deformation which corresponds to structural collapse.

Insert p. 3.6-42

3.6.2.2.3.2 Maximum Strain of Tension Members

Pure tension members, such as U-Bars shown on Figure 3.6-4 which act to limit pipe whip are permitted to deform during energy absorption, (a) a maximum of 50% of the minimum uniform strain (at the maximum stress on an engineering stress-strain curve) based on actual restraint material tests, or (b) one-half of minimum percent elongation as specified in the applicable ASME Code Section II or ASTM Specifications, if demonstrated to be ~~as or more conservative than (a).~~ The dynamic tensile and impact properties are specified to be not less than: (a) 70% of the static percent elongation, or (b) 80% of the statically determined minimum total energy absorption.

less than 50% of the minimum uniform strain based on representative test results.

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c. Jet impingement loading on primary containment penetrations is discussed in 3.8.6.

3.6.2.3.3 Pipe Whip Restraints

3.6.2.3.3.1 Definition of Function

Pipe whip restraints, as differentiated from piping supports, are designed to function and carry load for an extremely low probability gross failure in a piping system carrying high energy fluid. The piping integrity does not depend on the pipe whip restraints for any loading combination. If the piping integrity is compromised by a pipe break, the pipe whip restraint acts to limit the movement of the broken pipe to an acceptable distance. The pipe whip restraints (i.e., those devices which serve only to control the movement of a ruptured pipe following gross failure) will be subjected to a once in a lifetime loading. ~~For design purposes,~~ the pipe break event is considered to be a faulted

for the condition piping system ~~and the pipe,~~ its restraints, and structure to which the restraint is attached. ~~are analyzed accordingly.~~ INSERT Plastic deformation of the pipe is considered as a potential energy absorber. Piping systems are designed so that

would plastic instability ~~does~~ not occur in the pipe under design dynamic and static loads, if the consequences of such instability ~~will~~ could result in the loss of the primary containment integrity or loss of required plant shutdown capability.

3.6.2.3.3.2 Pipe Whip Restraint Features

a. The restraints are close to the pipe to minimize the kinetic energy of impact and yet are sufficiently removed from the pipe to permit unrestricted thermal pipe movement.

b. To facilitate in-service inspection of piping, the restraints are generally located a suitable distance away from all circumferential welds and are of bolted construction so as to be removable.

c. Pipe whip restraint structures fall into one of the following two categories:

(1) Energy absorbing members - These are modelled as elastic, elasto-plastic or plastic springs in a dynamic analysis.

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INSERT FSAR p. 3.6-51

Section 3.6.2.3.3.1

The design and analysis of these components for this event are described later in this Section, and in Section 3.6.2.2. Piping is no longer considered to be a part of the RCPB following the break.

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The required resistance (strength) of these structures is derived by application of the principles of structural dynamics.

(2) Load transmitting members - These are relatively stiff components and are modelled as rigid members in the dynamic analysis Their function is to transmit loading from the source to foundation. The load due to the postulated pipe rupture is in the form of an equivalent static load and is derived as a result of the dynamic analysis performed for the energy absorbing members.

d. Energy absorbing members are ductile structures such as simple beams, frames and ring girders, (including the piping system itself), having the capability to deflect significantly in absorbing the energy imparted to them by a postulated broken pipe. For loading conditions, including the effects of postulated pipe rupture, these members are designed within the limits for inelastic systems as stated in Table F1322.2-1 of ASME Boiler and Pressure Vessel Code Section III Appendix F "Rules for Evaluation of Faulted Conditions", adjusted to account for rapid strain rate effects, as discussed in 3.6.2.2.3. These members are constructed to meet the requirements of Quality Class I structures. ~~They can also take the shape of~~

¶ the U-Bar straps, as shown in Figure 3.6-4 and described in 3.6.2.2.3.2, ~~which~~ act as ENERGY ABSORBING, non-linear, non-rebounding, plastic springs. The U-Bar straps are justified by empirical data, AS DESCRIBED IN 3.6.2.2.2.2.b(3) AND 3.12.~~2~~.33

e. Load transmitting members are rigid components such as clevises, brackets or pins, rigid pipe whip restraint weldments as shown in Figures 3.6-4 AND 3.6-5a through 3.6-5e, or similar components; as well as major structures such as the drywell diaphragm floor, primary containment vessel, reactor pedestal, reactor building and foundation. For loading conditions, including the effects of postulated pipe rupture, the~~se~~ members IN FIGURES 3.6-5a THROUGH 3.6-5e are designed within the limits stated in Table F1322.2-1 of ASME Code Section III Appendix F "Rules for Evaluating Faulted Condition" for

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components and component supports; except that the members beyond those included in the dynamic analytical model (i.e. reactor pedestal, reactor building, as well as certain steel members assumed to be infinitely rigid) are designed to AISC, ACI and other appropriate structural component criteria. All these members are constructed to the requirements of Quality Class I structures.

✓ INSERT PARAGRAPH

f. The recirculation pump discharge and suction piping utilizes the U-Bar strap pipe whip ~~sup-~~ ~~ports~~, RESTRAINTS (FIGURE 3.6-4) while all other systems listed in Table 3.6-2 utilize rigid types as shown in Figures 3.6-5a through 3.6-5e or similar configurations.

g. Typical installations of pipe whip restraints are shown in Figures 3.6-6 through 3.6-10.

3.6.2.3.3.3 Pipe Whip Restraint Loading

a. For the purpose of predicting the pipe rupture forces associated with the reactor blowdown, the local line pressures are assumed to be those normally associated with the reactor operating at 105 percent of rated power and with a vessel dome pressure of 1025 psig.

b. In calculating pipe reaction, full credit is taken for any line restriction and line friction between the break and the pressure reservoir. The following represent typical restrictions to flow which are specifically considered:

(1) Jet pump nozzles

(2) Core spray nozzles (inside internals shroud)

(3) Feedwater sparger

(4) Steamline flow limiter

The hydraulic bases and calculational techniques for predicting unbalanced forces on a pipe associated with a postulated instantaneous pipe rupture are as discussed in 3.6.2.2.1.

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Insert Page 3.6-53

The design limits for connecting members such as clevises, brackets, and pins per Figure 3.6-4 are based on the following stress limits:

(1) Primary stresses (in accordance with definitions in ASME Section III) are limited to the higher of:

?(a) 70% of Su, where Su = minimum ultimate strength by tests or ASTM specification;

(b) Sy + 1/3 (Su - Sy), where Sy = minimum yield strength by test or ASTM specification; or

(2) Recommended stress limits in accordance with ASME Code Section III, Subsection NF for faulted conditions, if applicable. The design limits for welds of connecting members to steel structures are based on the following stress limits: the maximum primary weld stress intensity (two times shear stress) is limited to three times AWS or AISC building allowable weld shear stress.

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c. The dynamic loading on the pipe whip restraint commences at the effective time of impact of the pipe with the restraint. It includes the following:

(1) Unbalanced force on the pipe associated with a postulated instantaneous pipe rupture in the form of a suddenly applied force.

(2) Dynamic inertia load of the moving section of pipe which is accelerated by the unbalanced force associated with the pipe rupture and collides with the restraint. This load is in the form of kinetic energy of impact.

3.6.2.3.4 Pipe Whip Effects on Safety Related Components

Pipe whip (displacement) effects on safety related structures, systems and components can be placed in two categories: (a) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run in which the break occurred; and (b) controlled pipe whip displacements as they apply to external components such as building structure, other piping systems, cable trays and conduits.

3.6.2.3.4.1 Pipe Displacement Effects on Components in Same Piping Run

a. The criteria which is used for determining the effects of pipe displacements on in-line components are as follows:

(1) Components such as vessel safe ends, and valves which are attached to the broken piping system and do not serve a safety function or whose failure would not further escalate the consequences of the accident, need not be designed to meet ASME Code Section III imposed requirements for essential components under faulted loading.

(2) If these components are required for safe shutdown, or serve a safety function to protect the structural integrity of an essential component, the Code requirements for faulted conditions and limits to ensure operability, if required, are met.

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a. Assurance of primary containment leak tightness.

b. Assurance that potential for damage is such that the maximum pipe break areas and/or combinations of pipe break areas do not exceed the values described in 3.6.2.5.3.2 so that emergency core cooling system capability is not impaired.

c. Assurance that the control rod drive system maintains sufficient function to assure reactor shutdown.

d. Assurance that there is sufficient capability to maintain the reactor in a safe shutdown condition.

The criteria used to define pipe rupture locations for piping systems discussed in 3.6.2.5.4 follows 3.6.2.1.1.1b(1) except for the following which follow 3.6.2.1.1.1b(2) and 3.6.2.1.1.1.c in case of item d:

a. One elbow only, in each of the two redundant reactor feedwater systems inside primary containment, in 3.6.2.5.4.2 and in Figures 3.6-16a and 3.6-17a.

b. The entire standby liquid control (SLC) system in 3.6.2.5.4.4 and in Figure 3.6-19a.

c. The entire RPV drain system in 3.6.2.5.4.13 and in Figure 3.6-32a.

d. The entire reactor recirculation cooling system in 3.6.2.5.4.14 and in Figures 3.6-35a and 3.6-35b.

Figures 3.6-12a through 3.6-35 show the piping configurations for each high energy system inside primary containment and include numerical identification of all significant points of interest in the piping system, locations of pipe whip supports and postulated pipe break locations. The pipe whip supports are identified by the acronym PWS followed by an identification number on Figures 3.6-12a through 3.6-34~~a~~ and as noted on Figure 3.6-35a.

3.6.2.5.3 System Requirements Subsequent to Postulated Pipe Rupture

3.6.2.5.3.1 Control Rod Insertion Capability

To maintain the ability to insert the control rods in the event of a pipe break, no more than one in any array of nine control rod drive (CRD) withdrawal lines may be completely

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3.6.2.5.3.2 Core Cooling Requirements

The designed ECCS capability can be maintained provided that dynamic effects consequences do not exceed the following break area, break combination, and maintenance of minimum core cooling requirements.

3.6.2.5.3.3 Maximum Allowable Break Areas

a. For breaks involving recirculation piping, the total effective area of all broken pipes, including the effective area of the recirculation line break, does not exceed the total effective area of the design basis double-ended recirculation line break. By limiting the total area of all broken pipes involving recirculation loops, to an area less than, or equal to that of the design basis accident (DBA) (circumferential break of recirculation loop), no accident can be more severe than the DBA.

b. ~~For breaks not involving recirculation piping, the effects are much less severe than recirculation line breaks. Hence, the total break area can be allowed to be larger than the recirculation breaks. Therefore, the total break area does not exceed the sum of one feedwater header pipe area, one steam line pipe area (upstream of flow limiter), and one core spray pipe area.~~

SEE INSERT

3.6.2.5.3.4 Break Combinations

In addition to the pipe break area restrictions, breaks involving one recirculation loop do not result in loss of function or damage to the other recirculation loop, or loss of coolant from the other loop in excess of that which can result from a break of the attached cleanup connection on the suction side of the loop.

3.6.2.5.3.5 Required Cooling Systems

~~To ensure compliance with Appendix A of 10 CFR Part 50, General Design Criteria for Nuclear Power Plants, the following cooling system requirements are met after an additional single active safety system failure.~~

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Sect. 3.6.2.5.3.3

(b) For breaks not involving recirculation piping, the total effective area of all broken pipes for a given system shall not exceed the total effective area of the double-ended break of the maximum area pipe connected to the reactor boubdary for that system.

fix

Sect. 3.6.2.5.3.5

To ensure compliance with Appendix A of 10 CFR Part 50, General Design Criteria for Nuclear Power Plants, the cooling system requirements after an additional single active safety system failure are defined in Table 6.3-7. Cases which do not meet the requirements in Table 6.3-7 must be assessed ~~in~~ an individual basis to determine compliance with core cooling requirements.

on

Aren't the cases of Table 6.3.7 bounding? Therefore this statement is moot. Suggest this be revised to reflect that Table 6.3.7 list the bounding cases.

RB

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a. ~~For breaks not involving recirculation piping, at least two LPCI pumps or one core spray system is available for core cooling.~~

b. ~~For breaks involving recirculation piping, at least one core spray line and 2 LPCI pumps, or 2 core spary lines, are available for core cooling.~~

c. ~~For a LOCA with a total effective break area less than 0.7 ft^2, either the HPCS or ADS is available for reactor depressurization.~~

d. ~~For liquid breaks, such as cleanup suction or the combination of liqdid and steam breaks whose total break area is less than 0.7 ft^2 in which the ADS system is required for depressurization, at least 6 ADS valves are available.~~

e. ~~For breaks less than the equivalent flow area of one open ADS valve, at least 6 ADS valves are available. However, the required number of ADS valves is one less for each additional steam break area equivalent to the area of one open ADS valve.~~

ok

3.6.2.5.3.6 Containment System Integrity

The following were considered in addressing the LOCA dynamic effects with respect to containment system integrity:

a. Leak tightness of the containment fission product barrier is assured throughout any LOCA.

b. For those lines which penetrate the containment and are closed during normal operation, the inboard isolation valves are as close as practicable to the reactor pressure vessel. This arrangement reduces the length of pipe subject to a pipe break.

c. Pipe whip supports are provided in the vicinity of normally open isolation valves inside and outside primary containment for high energy systems, to assure that operability of these valves remains unimpaired during a postulated pipe rupture event.

83-175

support is also utilized as a rigid three-way
support.

3.6.2.5.4.14 Reactor Recirculation Cooling System

a. System Arrangement

TWO LOOPS "A" AND "B" OF THE SYSTEM
The recirculation piping consist~~s~~ of the pump
discharge and suction piping ~~systems.~~ The recir-
culation pump "A" and "B" discharge lines are
~~arranged with mirror image symmetry~~, in the
northern and southern segments of primary con-
tainment. The lines exit the reactor pressure
vessel in five, equally spaced, 12-inch diameter
lines commencing at azimuth 30° and ending at
azimuth 150° (for ~~the mirror image~~ azimuth 210°
to 330°). These five lines drop vertically
alongside the sacrificial shield wall, from ele-
vation 536'~~-0 1/4"~~ to a 16-inch diameter header
at centerline elevation of 528'~~-1 1/4"~~. A single
24-inch diameter line then drops vertically from
the center of the header to elevation 506'~~-5 1/8"~~
where it is routed into the discharge nozzle~~s~~ of
the recirculation pump~~s~~.

AS SHOWN IN FIGURE 3.6-35a.

ARRANGED IN A DIAMETRICALLY OPPOSED MANNER,

RESPECTIVELY

"A"

"B" LINES

0"

0"

3 7/8"

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The recirculation pump ~~"A" and "B"~~ "B" AND "A" suction lines ARE ~~consits of two mirror image systems~~ oriented RESPECTIVELY along the 0° and 180° azimuths with respect to the reactor pressure vessel. Each ~~loop~~ SUCTION LINE consists of a single 24-inch diameter line which exits the reactor pressure vessel at elevation 535'-3/4" and drops vertically alongside the sacrificial shield wall to elevation 502'-6 1/8" where it is routed to the suction nozzle~~s~~ of the recirculation pump~~s~~.

b. Pipe Whip Protection

For the recirculation pump suction and discharge systems, the location of postulated pipe breaks and pipe whip restraints are shown on Figure 3.6-35a ~~3.6-35~~ which is representative of both recirculation loops. CONFORMANCE OF THE POSTULATED BREAK LOCATIONS WITH THE CRITERIA OF SECTION 3.6.2.1.1.1 IS DEMONSTRATED IN FIGURE 3.6-35b. Where pipe breaks are postulated inside primary containment, the recirculation system piping is restrained to prevent unacceptable motion. These restraints are generally mounted on the side of the sacrificial shield wall structure or the reactor pressure vessel (RPV) pedestal, immediately below. Four restraints, which are located near the diaphragm floor and are not near the sacrificial shield wall or the RPV pedestal, consist of saddle type structures mounted on the diaphragm floor.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the reactor recirculation cooling system piping to assure safety as defined in 3.6.2.5.2. Pipe whip ~~supports~~ RESTRAINTS are provided to prevent impact with the diaphragm floor as well as to mitigate the consequences of a pipe rupture with respect to surrounding piping systems, structures and components required for safe shutdown.

The physical separation of the recirculation system from the containment vessel precludes any damage that could result as a result of postulated pipe break.

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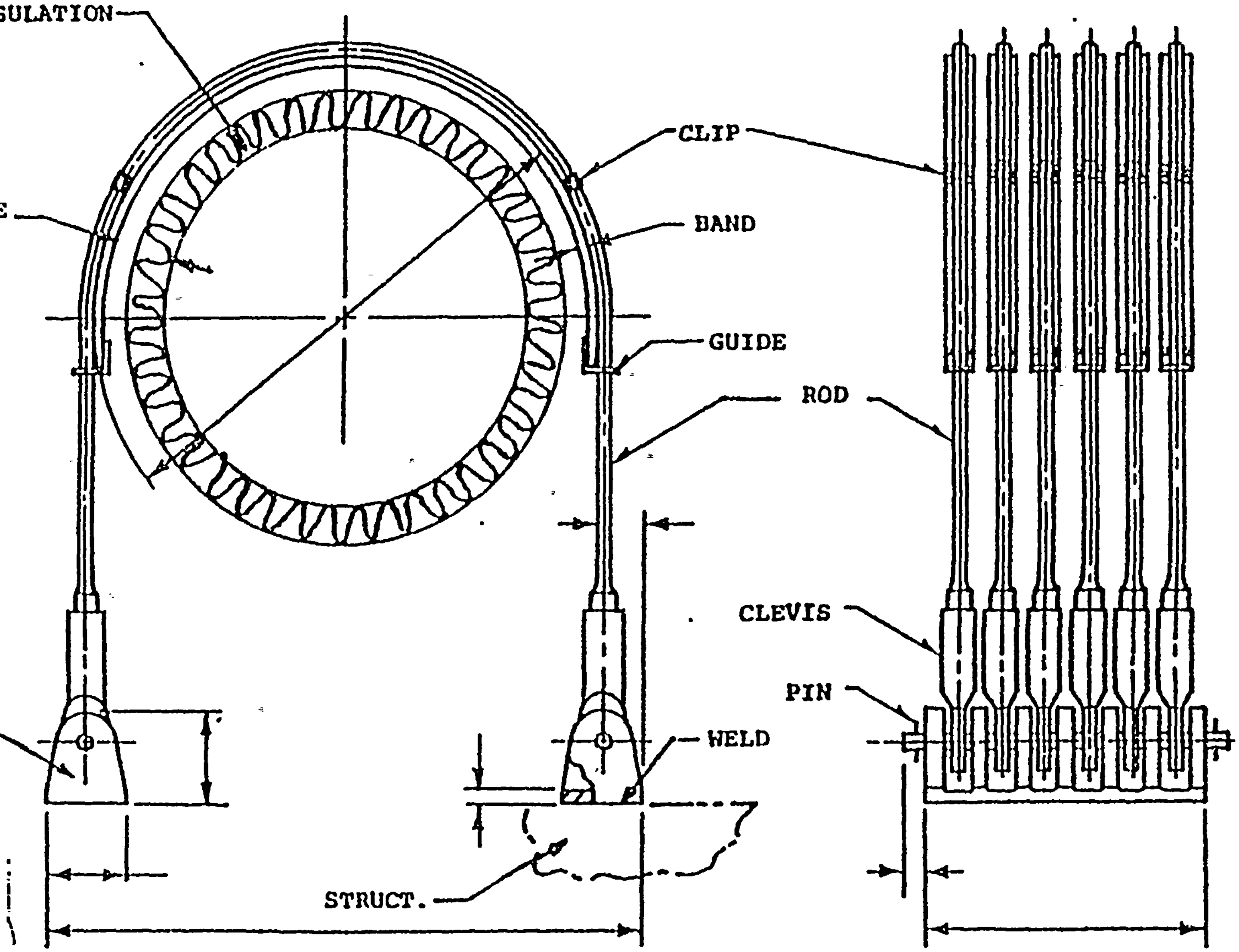

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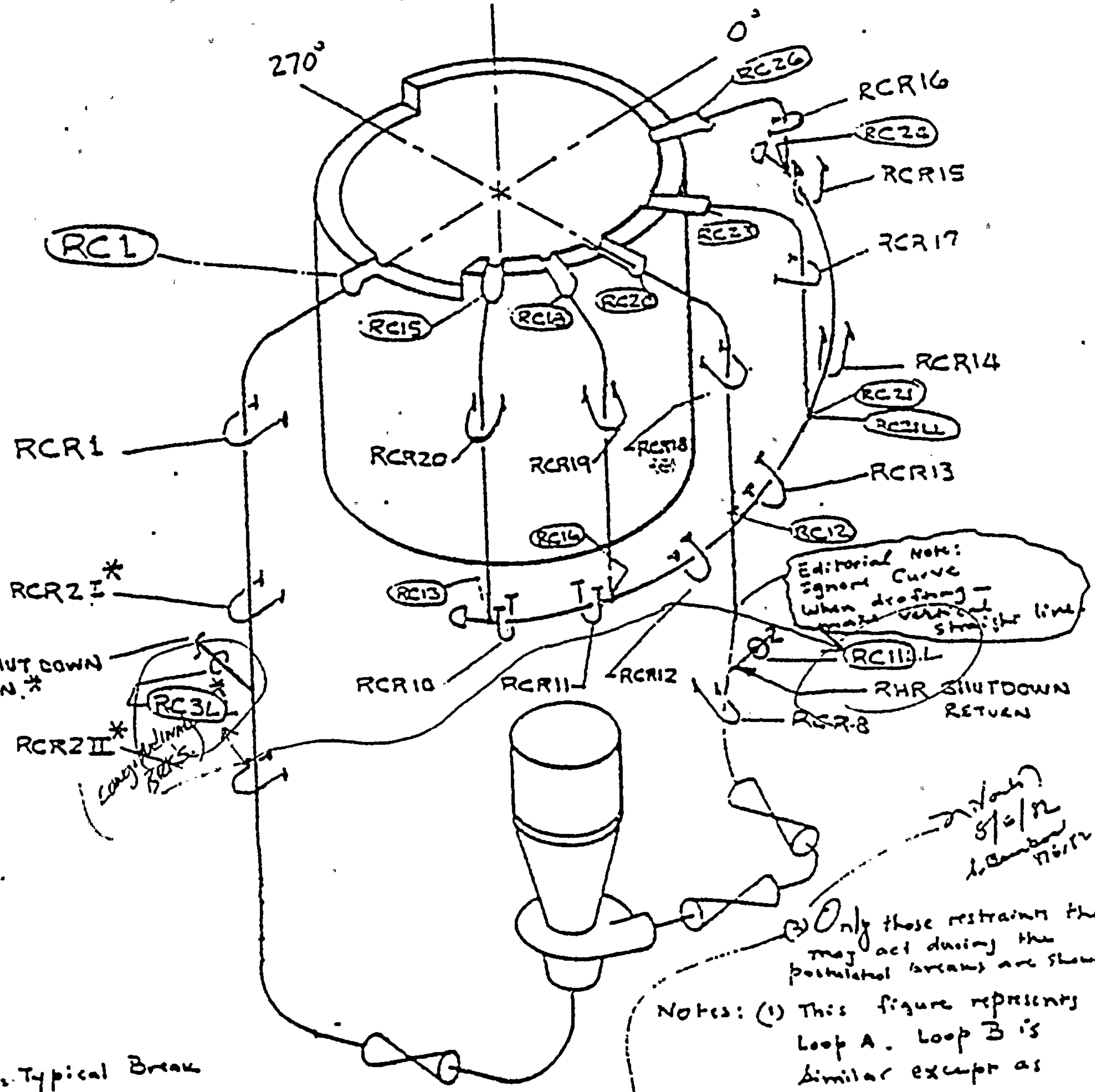
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

U-BAR ~~FLEXIBLE~~ TYPE PIPE WHIP RESTRAINT CONFIGURATION

FIGURE 3.6-4

HANFORD

~~LOOP A~~



(3) Only those restraints that may act during the postulated breaks are shown.

KEY:

RC1 = Typical Break Location

RCR1 = Typical Restraint Designation

Suffix "LL" indicates longitudinal break

* indicates Loop A only

Notes: (1) This figure represents Loop A. Loop B is similar except as noted.

(2) See figure 3.6-35b for restraint-break location correlation and break types.

(3)

REACTOR RECIRCULATION PIPING SYSTEM

FIGURE 3.6-35a

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FIGURE 3.6-35b

RECIRCULATION PIPING SYSTEM OPERATING STRESSES AND RESTRAINTS AT BREAK LOCATIONS (1)

BREAK IDENT (1)	Acting Restraint No. (1)	STRESS RATIO PER ASME EQNS. EQ(10) $\frac{Sn}{3Sm}$	EQ(12) $\frac{So}{3Sm}$	EQ(13) $\frac{S}{3Sm}$	USAGE FACTOR	BREAK TYPE	BREAK BASES SECTION NO.
RC1	RCR1	0.62	0.13	0.52	0.00	CIRCMF	3.6.2.1.1.1.a
RC15	RCR20	0.56	0.12	0.33	0.00	"	"
RC18	RCR19	0.70	0.26	0.48	0.00	"	"
RC20	RCR18	0.63	0.16	0.49	0.00	"	"
RC23	RCR17	0.68	0.22	0.49	0.00	"	"
RC26	RCR16	0.67	0.23	0.49	0.00	"	"
RC13	RCR10	0.94	0.24	0.65	0.01	" (3)	3.6.2.1.1.1.c
RC16	RCR11	1.10	0.17	0.69	0.03	" (3)	"
RC12	RCR12 RCR13	1.17	0.60	0.55	0.096	" (5)	"
RC21	RCR14	1.52	0.65	0.68	0.60	" (3)	3.6.2.1.1.1.b(2)(b)
RC21LL	RCR14	1.52	0.65	0.68	0.60	LONG (2)	"
RC24	RCR15	1.14	0.44	0.66	0.05	CIRCMF (3)	3.6.2.1.1.1.c

{See Footnotes on next page.

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FIGURE 3.6.35b

RECIRCULATION PIPING SYSTEM OPERATING STRESSES AND RESTRAINTS AT BREAK LOCATIONS (1)

BREAK IDENT (1)	Acting Restraint No. (1)	STRESS RATIO PER ASME EQNS. EQ(10) $\frac{Sn}{3Sm}$	EQ(12) $\frac{Se}{3Sm}$	EQ(13) $\frac{S}{3Sm}$	USAGE FACTOR	BREAK TYPE	BREAK BASES SECTION NO.
RC11LL	RCR8	1.02	0.19	0.83	0.08	LONG (3)	3.6.2-1-1.1b(2)(c)
RC3LL	RCR2I RCR2II	0.97	0.34	0.70	0.01	LONG (3)	3.6.2.1.1.1.c

Notes: (1) This information is for Loop A and is typical for Loop B. Break RC3LL applies to Loop A only. See Figure 3.6-35a for Break identity and Restraint No.

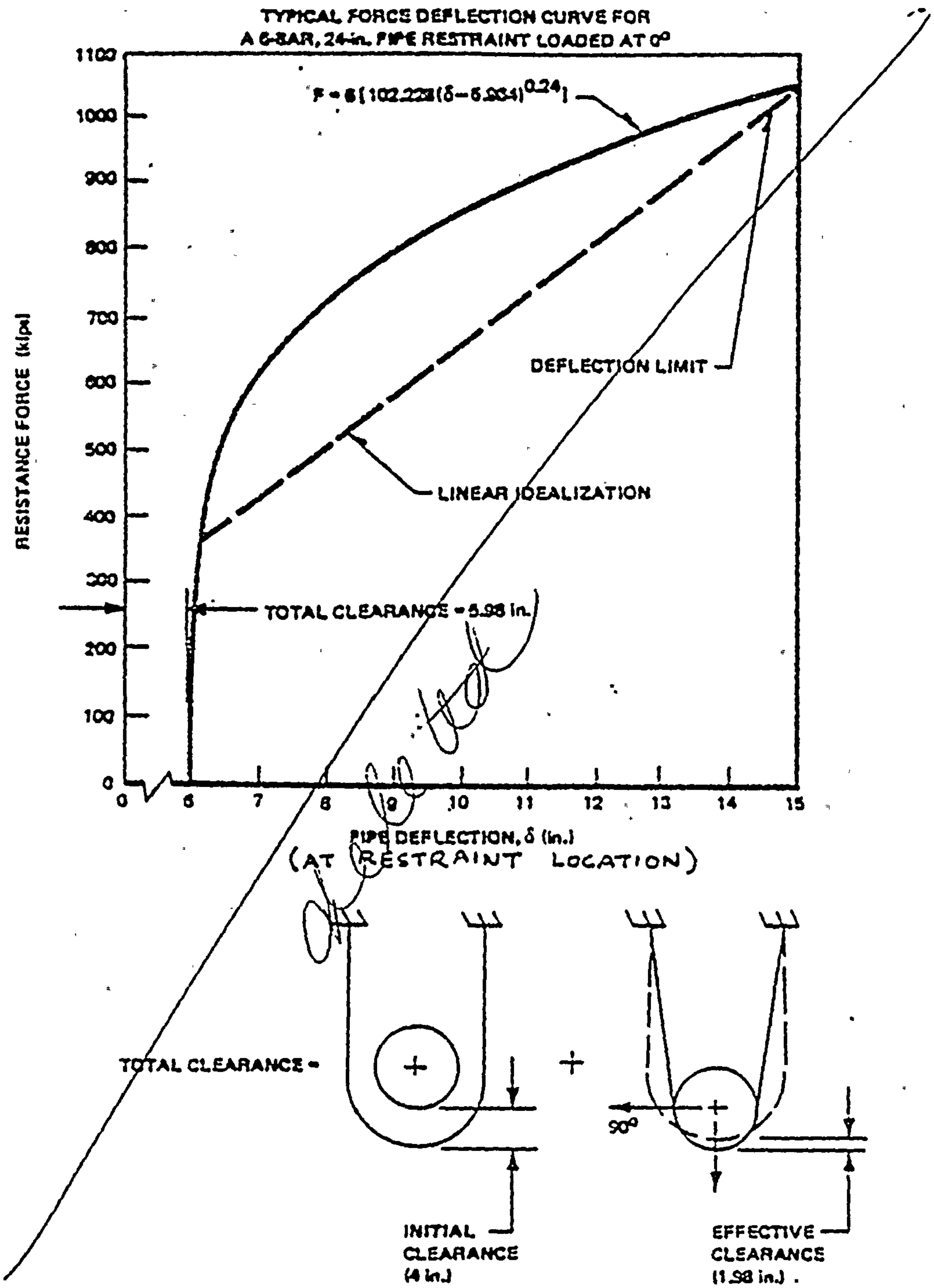
(2) Out-of-plane longitudinal

(3) Break at connection of contour nozzle to the header

(4) Circumferential break at branch weld, but longitudinal type for the riser.

(5) Break at branch weld.

was deleted in A-39, why inserted again?



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TYPICAL RESTRAINT FORCE-DEFLECTION CURVE	FIGURE 3.12-5

82-175

THIS FIGURE HAS BEEN
INTENTIONALLY DELETED

Refered to Figure 3.6-35a

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	BREAK LOCATIONS AND RESTRAINTS ANALYZED, PDA VERIFICATION PROGRAM	FIGURE 3.12-6

TABLE 3.12-3

RESTRAINT PROPERTIES USED IN ~~ANALYSES~~ a

~~BY BOTH NSC AND PDA~~[1] Comparison

General Restraint ~~Data~~ Property for 1 Bar of a Restraint

$F = C_2\ (\Delta\ \text{restraint})^n$

Where Δ restraint = δ pipe - Total clearance (See Figure 3.12-5)

Pipe Size (In)	Rest Load Direction	C_2	n	Limit Δ Restraint	Initial Clearance	Effective Clearance	Total Clearance
12	0°	27,733	0.24	6.129	4	1.941	5.941
12	90°	14,795	0.401	9.063	4	12.247	16.247
16	0°	109,265	0.24	6.278	4	1.934	5.934
16	90°	62,599	~~0.377~~	8.978	4	12.187	16.187
24	0°	102,228	0.24	8.222	4	1.984	5.~~0~~984 9
24	90°	55,531	0.375	11.972	4	13.685	17.685
24	38° ~~(2)~~ a	109,888	~~0.24~~	~~5.588~~	4	5.698	9.698
24	52° ~~(2)~~ a	109,835	0.24	5.473	4	8.462	12.462

See Comment No. 23

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[1] NSC denotes Nuclear Services Corporation, and PDA denotes "Pipe Dynamic Analysis Program for Pipe Break Movement" by General Electric Company.

~~[2] Applies to Restraint RCR 3 only.~~ a

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TABLE 3.12-3 (Continued)

COMPARISON OF PDA AND NSC CODE[1]

Break Indent (Figure 3.12-6)	Restraint Indent (Figure 3.12-6)	No. of Bars PDA	No. of Bars NSC	Load (kips) PDA	Load (kips) NSC	Restraint Deflection (in.) PDA	Restraint Deflection (in.) NSC	% Of Design Restraint Deflection PDA	% Of Design Restraint Deflection NSC	Pipe Deflection (in.) PDA	Pipe Deflection (in.) NSC
$RC1_J$	RCR1	5	5	003.2	708.3	6.57	7.926	79.93%	96.4 %	17.72	15.58
$RC2_{LL}$	RCR1	5	5	766.4	458.4	14.99	7.495	125 %	62.6 %	35.83	24.52
$RC3_{LL}$	RCR2	6	6	747.0	639.7	2.27	3.73	27.65%	45.35%	17.16	20.11
$RC3_{LL}$	RCR2	6	6	796.6	780.3	10.22	10.54	57.8 %	59.6 %	41.48	43.0
$RC4_{LL}$	RCR3	5	5	836.0	838.4	7.64	8.05	92.95%	97.98%	18.87	16.43
$RC4_{LL}$	RCR3	8	8	1319.0	1073.9	5.43	4.62	99.23%	76.85%	23.38	17.25
$RC4C_V$	RCR3	8	8	1260.7	1275.0	4.49	5.58	80.37%	99.89%	22.56	18.73
$RC6A_V$	RCR3	8	8	928.5	722.5	1.22	1.77	22.46%	31.7 %	23.68	95.39
$RC7_J$	RCR7	6	6	953.3	80.61 80.6	6.28	5.76	76.4 %	70.12%	16.46	21.63
$RC8_{LL}$	RCR6	4	4	599.0	0	[illegible]	0	112.46%	0	26.76	8.39
	RCR7	6	6	895.0	0	8.16	0	110.76%	0	29.316	
$RC9C_V$	RCR6	4	4	575.8	520.16	4.16	5.53	50.63%	67.33%	13.2	14.56
$RC9_{LL}$	RCR8	6	6	830.2	546.78	11.408	6.815	95.29%	56.9 %	36.612	26.24
RC11A	RCR8	6	6	818.3	493.6	10.98	5.99	91.72%	50.07%	31.404	23.71
RC13	RCR10	4	4	668.4	478.0	5.87	3.66	93.5 %	58.39%	13.37	10.44
RC16	RCR11	4	4	687.4	518.4	6.59	4.38	105 %	69.86%	15.37	10.22
$RC14C_V$	RCR20	8	8	285.0	309.6	2.83	5.88	46.3 %	95.92%	15.45	13.96
$RC14_{LL}$	RCR20	8	8	116.3	129.9	0.96	3.36	10.5 %	37.1 %	22.13	23.56

A

[1] NSC denotes Nuclear Services Corporation, and PDA denotes "Pipe Dynamic Analysis Program for Pipe Rupture Movement" by General Electric Company.

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TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
1	RCIC(13)-4	120-1	4	Later	3.6-49
2	RCIC(13)-4	120-2	4	3.6-69, 70	3.6-49
3	RCIC(13)-4	120-3	4	3.6-65, 66	3.6-49
4	RCIC(13)-4	120-4	4	Later*	3.6-48
5	RCIC(13)-4	120-5	4	Later*	3.6-48
6	RCIC(13)-4	120-6	4	Later*	3.6-48
7	RCIC(13)-4	120-7	4	Later*	3.6-48
8	RCIC(13)-4	120-8	4	Later*	3.6-47
~~9~~	~~RCIC(13)-4~~	~~120-9~~	~~4~~	~~Later*~~	~~3.6-47~~
10	RCIC(13)-4	120-10	4	3.6-63, 64	3.6-47
11	RWCU(1)-4	126-1	6	3.6-79, 80	3.6-51
12	RWCU(1)-4	126-2	6	3.6-75, 76	3.6-50
13	RWCU(1)-4	126-3	4 2	Later*	3.6-50
~~14~~	~~RWCU(1)-4~~	~~126-4~~	~~4~~	~~Later*~~	~~3.6-50~~
15	RWCU(1)-4	126-5	2	Later*	3.6-50
16	RWCU(1)-4	126-6	2 4	3.6-81, 82	3.6-51
17	RWCU(2)-4	128-7	6	3.6-67, 68	3.6-51
18	RWCU(2)-4	128-8	6	Later*	3.6-51
19	RWCU(2)-4	128-9	6	Later*	3.6-51
20	RWCU(2)-4	128-10	6	Later*	3.6-50
21	RWCU(2)-4	128-11	6	Later*	3.6-49
22	~~RWCU(2)-4~~	~~128-12~~ 9	~~6~~	~~Later*~~	~~3.6-49~~
23	~~RWCU(3)-4~~	~~128-39~~ 9	~~6~~	~~Later*~~	3.6-50
24	~~RWCU(3)-4~~	~~128-40~~ 9	~~6~~	~~Later*~~	3.6-50
25	~~RWCU(3)-4~~	~~128-41~~ 9	~~6~~	~~Later*~~	3.6-50

REPLACE PAGES WITH ATTACHED

AMENDMENT NO. 25
June 1982

TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
26	RWCU(3)-4	129-42	6	Later*	3.6-50
27	RWCU(3)-4	129-43	4	Later*	3.6-50
28	RWCU(3)-4	129-44	4	Later*	3.6-50
29	RWCU(3)-4	129-45	4	Later*	3.6-50
30	~~RWCU(3)-4~~	~~129-46~~	~~4~~	~~Later*~~	3.6-50
31	RWCU(3)-4	129-47	3	Later*	3.6-50
32	RWCU(3)-4	129-48	4	Later*	3.6-50
33	~~RWCU(3)-4~~	~~129-49~~	~~4~~	~~Later*~~	3.6-50
34	RWCU(3)-4	129-50	3	Later*	3.6-50
35	MS(20)-4	134-1	2	Later*	3.6-44
36	MS(20)-4	134-2	2	Later*	3.6-44
37	MS(20)-4	134-3	2	Later*	3.6-44
38	MS(20)-4	134-4	2	Later*	3.6-44
~~39~~	~~MS(20)-4~~	~~134-5~~	~~3~~	~~Later*~~	~~3.6-44~~
40	AS(11)-2	139-1	3	3.6-97, 98	3.6-43
~~41~~	~~AS(11)-2~~	~~139-2~~	~~2~~	~~Later*~~	~~3.6-43~~
42	AS(11)-2	139-3	3	3.6-93, 94	3.6-43
43	AS(11)-2	139-4	4	Later*	3.6-43
~~44~~	~~AS(11)-2~~	~~139-5~~	~~4~~	~~Later*~~	~~3.6-43~~
~~45~~	~~AS(11)-2~~	~~139-6~~	~~4~~	~~Later*~~	~~3.6-43~~
46	AS(11)-2	139-7	4	Later*	3.6-43
~~47~~	~~AS(11)-2~~	~~139-8~~	~~3~~	~~Later*~~	~~3.6-43~~
~~48~~	~~AS(11)-2~~	~~139-9~~	~~4~~	~~Later*~~	~~3.6-43~~
~~49~~	~~AS(11)-2~~	~~139-17~~	~~2~~	~~3.6-95, 96~~	~~3.6-43~~
50	~~AS(11)-2~~ AS(10)-2	141-10	6	Later*	3.6-43

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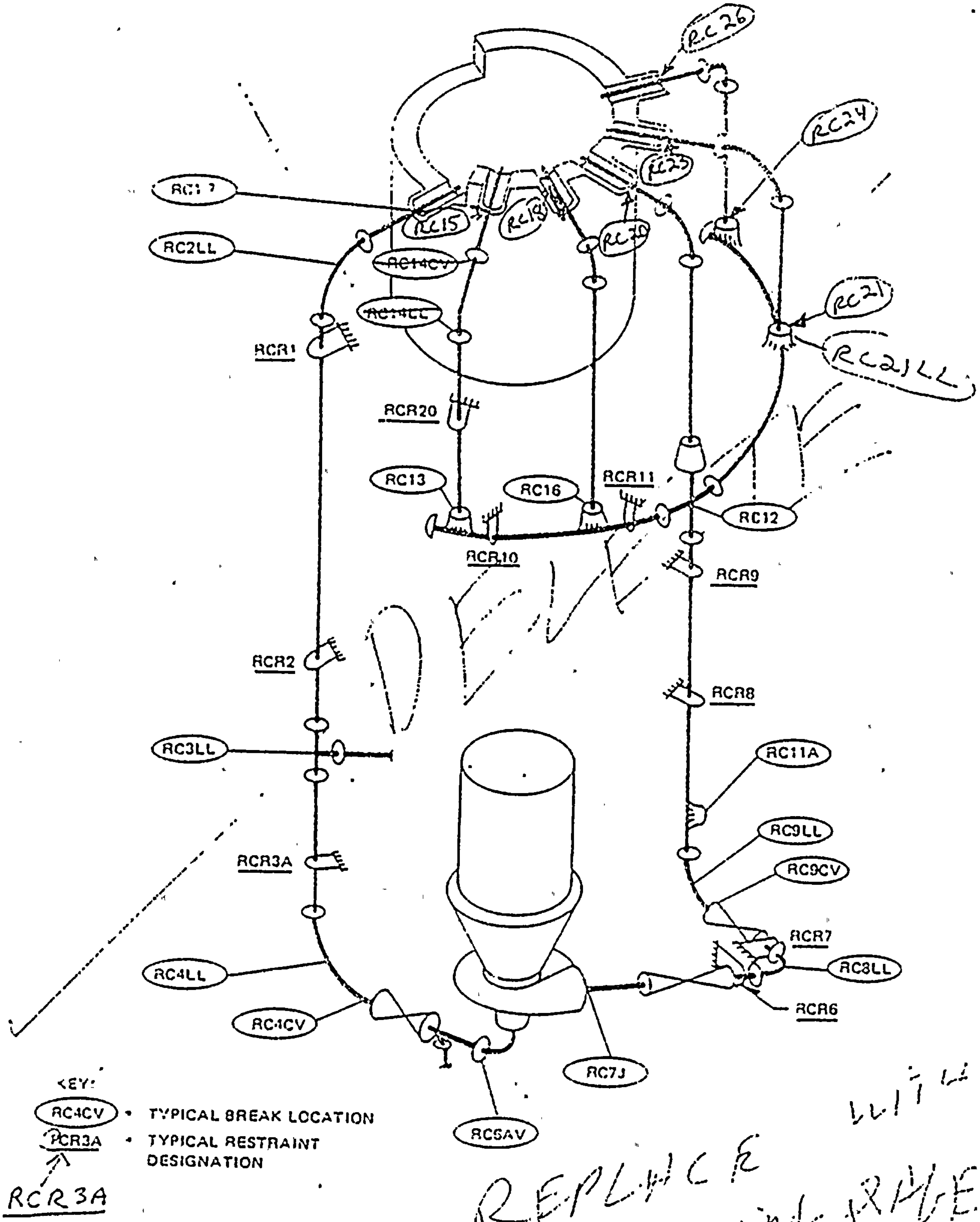
June 1982

Amendment No. 5
August 1979



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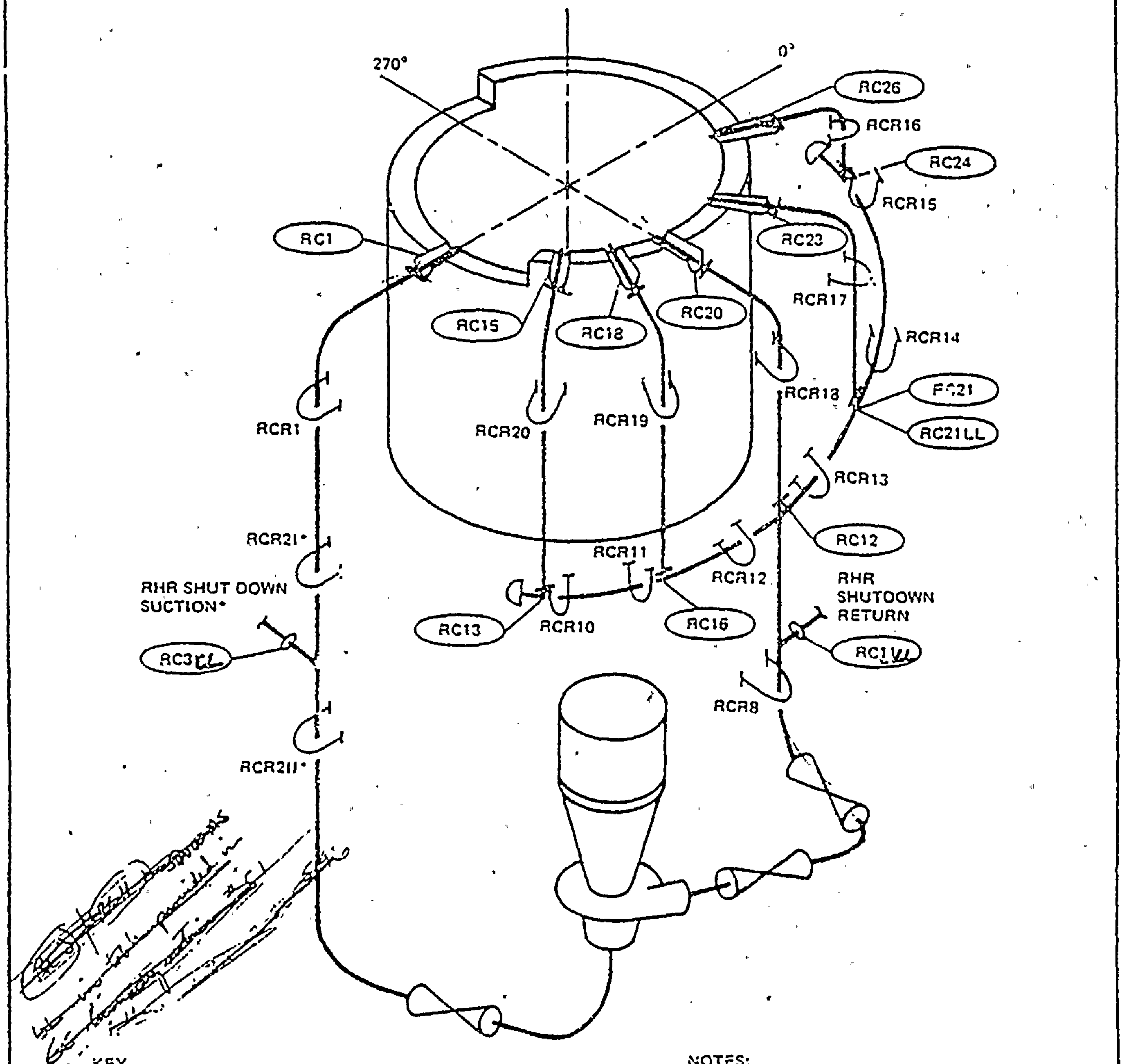
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION COOLING SYSTEM	FIGURE 3.6-35

Updated in Amendment 31



KEY

RC1 • TYPICAL BREAK LOCATION

RCR1 • TYPICAL RESTRAINT DESIGNATION

SUFFIX "LL" INDICATES LONGITUDINAL BREAK

* INDICATES LOOP A ONLY

NOTES:

1. THIS FIGURE REPRESENTS LOOP A. LOOP B IS SIMILAR EXCEPT AS NOTED.
2. SEE FIGURE 3.6-35b FOR RESTRAINT-BREAK LOCATION CORRELATION AND BREAK TYPES.
3. ONLY THOSE RESTRAINTS THAT MAY ACT DURING THE POSTULATED BREAKS ARE SHOWN.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	BREAK LOCATIONS AND RESTRAINTS ANALYZED, PDA VERIFICATION PROGRAM	FIGURE 3.6-35a

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AMENDMENT NO. 32
—— 1983

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
NODE 11	NODE 11 A
NODE 12	NODE 41 A
NODE 41	
NODE 42	
NODE 9	
NODE 37	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RFW INSIDE MAIN STEAM TUNNEL	FIG 3.6-24b

Form BR8002A (9/82) 200M

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RFW INSIDE MAIN STEAM TUNNEL

FIGURE 3.6-34a

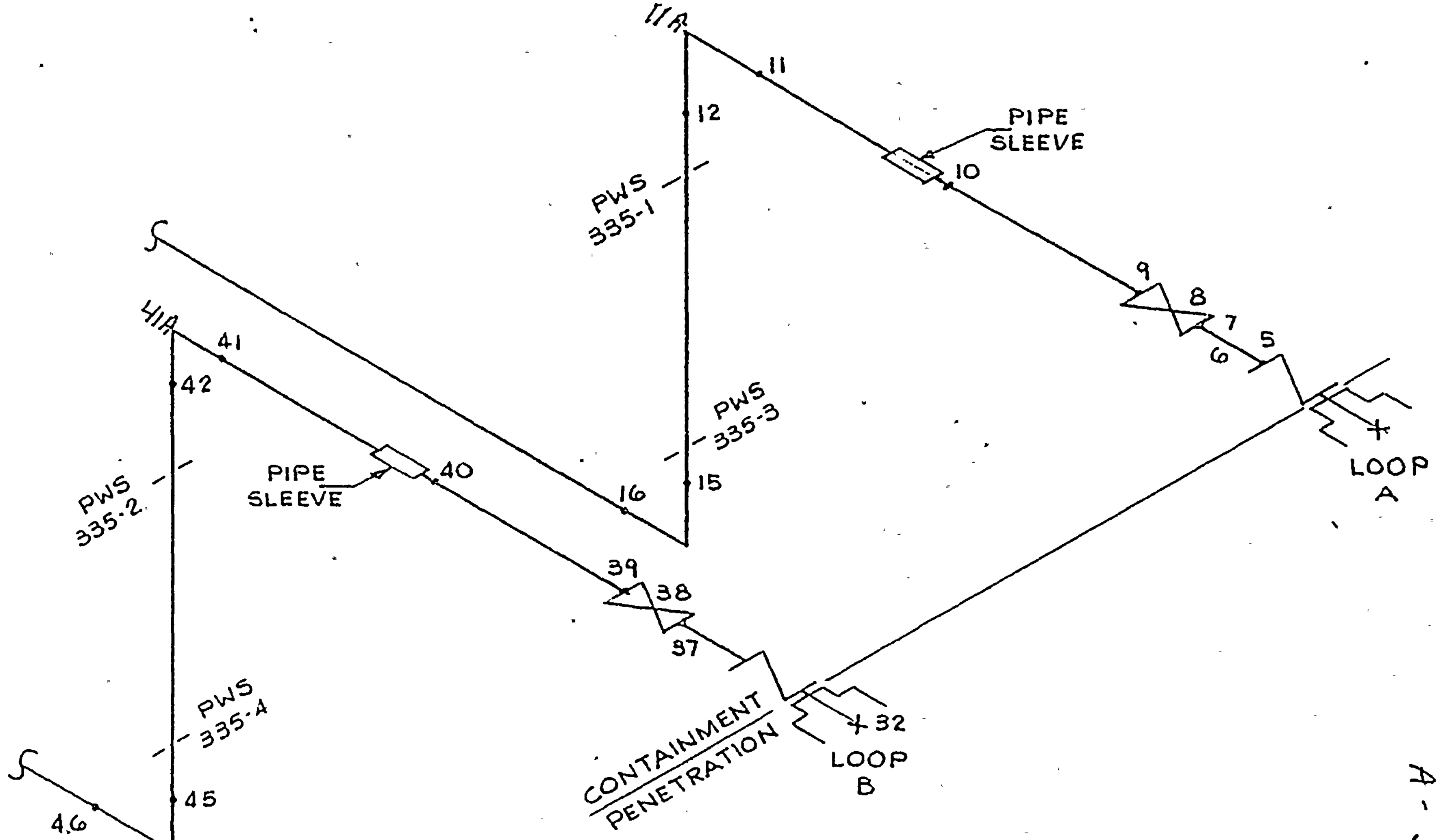

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TABLE 3.6-5

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
51	AS(10)-2	141-11	6	Later* 7.28	3.6-43
52	AS(10)-2	141-12	6	Later* 7.28	3.6-43
53	RWCU(1)-4	142-20	4	Later* FIG 3.6-85	3.6-51
54	RWCU(1)-4	142-21	4	Later* "	3.6-51
55	RWCU(1)-4	142-22	4	Later* "	3.6-51
56	RWCU(1)-4	142-23	4	Later* "	3.6-51
57	RWCU(1)-3	144-24	4	Later*	3.6-53
~~58~~	~~RWCU(1)-3~~	~~144-25~~	~~4~~	~~Later*~~	~~3.6-51~~
59	RWCU(1)-3	144-26	4	Later*	3.6-51
60	RWCU(1)-3	144-27	4	Later*	3.6-51
61	RWCU(1)-3	144-28	4	Later*	3.6-51
62	RWCU(1)-3	144-29	6	Later*	3.6-51
~~63~~	~~RWCU(2)-3~~	~~144-30~~	~~6~~	~~Later*~~	~~3.6-51~~
64	RWCU(2)-3	144-31	4	Later*	3.6-51
65	RWCU(2)-3	144-32	6	Later*	3.6-51
~~66~~	~~RWCU(2)-3~~	~~144-33~~	~~6~~	~~Later*~~	~~3.6-51~~
67	RWCU(2)-3	144-34	6	Later*	3.6-51
~~68~~	~~RWCU(2)-3~~	~~144-35~~	~~6~~	~~Later*~~	~~3.6-51~~
69	RWCU(2)-3	144-36	6	Later*	3.6-53
70	HS(9)-2	148-1	3	3.6-112, 113	3.6-43
71	HS(1)-2	148-2	4	Later*	3.6-43
~~72~~	~~HS(5)-2~~	~~148-4~~	~~4~~	~~3.6-105, 106~~	~~3.6-43~~
73	HS(5)-2	148-5	2	Later*	3.6-43
74	HS(5)-2	148-6	2	Later*	3.6-43
75	HS(5)-2	148-7	2	Later*	3.6-43

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TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
76	HS(5)-2	148-8	2	Later*	3.6-43
77	HS(5)-2	148-9	2	Later*	3.6-43
78	HS(5)-2	148-10	2	Later*	3.6-43
79	HS(5)-2	148-11	2	Later*	3.5-43
80	HS(5)-2	148-12	2	Later*	3.6-43
~~81~~	~~HS(9)-2~~	~~148-13~~	~~2~~	~~Later*~~	~~3.6-43~~
~~82~~	~~HS(1)-3~~	~~148-14~~	~~4~~	~~3.6-87, 88, 107~~	~~3.6-43~~
~~83~~	~~HCO(11)-1~~	~~149-1~~	~~6~~	~~Later*~~	~~3.6-62~~
84	HCO(11)-1	149-2	4	Later*	3.6-62
~~85~~	~~HCO(11)-2~~	~~149-3~~	~~4~~	~~Later*~~	~~3.6-58~~
~~86~~	~~HCO(11)-2~~	~~149-4~~	~~3~~	~~Later*~~	~~3.6-58~~
87	HCO(11)-2	149-5	3	3.6-99	3.6-58
~~88~~	~~HCO (11)-2~~	~~149-6~~	~~3~~	~~3.6-100, 101~~	~~3.6-58~~
~~89~~	~~HCO (11)-2~~	~~149-7~~	~~3~~	~~Later*~~	~~3.6-62~~
~~90~~	~~HCO (11)-2~~	~~149-8~~	~~3~~	~~3.6-102~~	~~3.6-59~~
~~91~~	~~HCO-(5)-2~~	~~149-9~~	~~2.5~~	~~Later*~~	~~3.6-59~~
~~92~~	~~HCO-(5)-2~~	~~149-10~~	~~2.5~~	~~Later*~~	~~3.6-59~~
~~93~~	~~HCO (5)-2~~	~~149-11~~	~~2.5~~	~~Later*~~	~~3.6-59~~
~~94~~	~~HCO (5)-2~~	~~149-12~~	~~2.5~~	~~Later*~~	~~3.6-59~~
95	RFW(1)-4	335-1	24	Later*	3.6-49
96	RFW(1)-4	335-2	24	Later*	3.6-49
97	RFW(1)-4	335-3	24	Later*	3.6-49
98	RFW(1)-4	335-4	24	Later*	3.6-49
99	AS(9)-2	342-13	6	Later*	3.6-43
100	AS(9)-2	342-14	4	Later*	3.6-43

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DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
~~101~~	~~AS(1)-2~~	~~342-15~~	~~8~~	~~Later*~~	~~3.6-43~~
~~102~~	~~AS(3)-2~~	~~341-16~~	~~2~~	~~Later*~~	~~3.6-43~~
103	MS(1)-4	400-8	25	Later*	3.6-44
~~104~~	~~MS(1)-4~~	~~400-9~~	~~26~~	~~Later*~~	~~3.6-44~~
~~105~~	~~MS(1)-4~~	~~400-10~~	~~30~~	~~Later*~~	~~3.6-44~~
106	MS(1)-4	400-11	26	Later*	3.6-44
~~107~~	~~MS(1)-4~~	~~400-12~~	~~26~~	~~Later*~~	~~3.6-44~~
~~108~~	~~MS(1)-4~~	~~400-13~~	~~30~~	~~Later*~~	~~3.6-44~~
109	MS(1)-4	400-14	26	Later*	3.6-44
~~110~~	~~MS(1)-4~~	~~400-15~~	~~26~~	~~Later*~~	~~3.6-44~~
~~111~~	~~MS(1)-4~~	~~400-16~~	~~30~~	~~Later*~~	~~3.6-44~~
~~112~~	~~MS(1)-4~~	~~400-17~~	~~30~~	~~Later*~~	~~3.6-44~~
113	MS(1)-4	400-18	26	Later*	3.6-44
~~114~~	~~MS(1)-4~~	~~400-19~~	~~26~~	~~Later*~~	~~3.6-44~~
~~115~~	~~MS(1)-4~~	~~400-20~~	~~30~~	~~Later*~~	~~3.6-44~~
~~116~~	~~MS(1)-4~~	~~400-21~~	~~30~~	~~Later*~~	~~3.6-44~~
117	CO(3)-2	440-1	2.5	Later*	N/A
118	CO(3)-2	440-2	2.5	Later*	N/A
119	CO(3)-2	440-3	2.5	Later*	N/A
120	HS(5)-1	447-19	6	Later*	N/A
121	HS(5)-1	447-25	6	Later*	N/A
122	HS(5)-1	447-26	6	Later*	N/A
123	HS(5)-1	447-27	6	Later*	N/A
124	HS(1)-1	448-15	6	Later*	N/A
125	HS(1)-1	448-16	6	Later*	N/A

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DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
126	HS(1)-1	448-17	6	Later*	N/A
127	HS(1)-1	448-18	6	Later*	N/A
128	HS(1)-1	448-19	6	Later*	N/A
129	HS(1)-1	448-20	6	Later*	N/A
130	HS(1)-1	448-21	5	Later*	N/A
131	HS(1)-1	448-22	5	Later*	N/A
132	HS(1)-1	448-23	4	Later*	N/A
133	HS(1)-1	448-24	~~6~~ 2"	Later*	N/A
134	HCO(5)-1	449-13	3	Later*	N/A
135	HCO(5)-1	449-14	3	Later*	N/A
136	HCO(5)-1	449-15	3	Later*	N/A
137	HCO(5)-1	449-16	3	Later*	N/A
138	HCO(5)-1	449-17	3	Later*	N/A
139	HCO(5)-1	449-18	3	Later*	N/A
140	HCO(5)-1	449-19	3	Later*	N/A
141	HCO(5)-1	449-20	3	Later*	N/A
142	HCO(5)-1	449-21	3	Later*	N/A
143	HCO(5)-1	449-22	3	Later*	N/A
144	HCO(5)-1	450-33	3	Later*	N/A
145	HCO(5)-1	450-24	3	Later*	N/A
146	HCO(5)-1	450-25	3	Later*	N/A
147	HCO(5)-1	450-26	2.5	Later*	N/A
148	HCO(5)-1	450-27	3	Later*	N/A
149	HCO(5)-1	449-28	3	Later*	N/A
150	MS(9)-4	451-6	3	Later*	N/A

TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

Break No.	Line Designation	Isometric No. (M200)	Diameter (Inches)	Max. Force (kips) or Thrust vs. Time Figure	Plan Location Figure
151	MS(9)-4	451-7	3	Later*	N/A
~~152~~	~~CRD(12)-3~~	~~N/A~~	~~8~~	~~N/A~~	~~See Section~~ 3.6.1.18.3.5 and the response to NRC Question 010.014.

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*Information is scheduled to be ready for Staff review in late 1982.

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TABLE 3.6-7

SEISMIC AND QUALITY CLASSIFICATION

Page 1 of 2

Line Designation	Diameter	Classification Seismic	Classification Quality
RCIC (13)-4	4	I	I
RWCU (1)-4	4,6	I	I
RWCU (2)-4	4,6	I	I
RWCU (1)-3	4	I	I
RWCU (2)-3	4,6	I	I
RWCU (3)-4	4,6	I	I
RWCU (5)-3	4,6	I	I
RWCU (6)-4	6	I	I
RWCU (7)-3	6	I	I
AS (1)-2	4,8	I/II	II
AS (3)-2	2	I	II
AS (10)-2	6,8	I/II	II
AS (11)-2	2,3,4	I	II
AS (16)-2	2.5,3	I	II
CO (3)-2	2,2.5	II	II
HCO (5)-1	2.5,3	II	II
HCO (5)-2	2,2.5,3	I	II
HCO (9)-2	2	I	II
HCO (11)-1	2.5,4,6	I	II
HCO (11)-2	3	I	II
HS (1)-1	4,6	I/II	II
HS (1)-2	2,4	I	II
HS (5)-1	6	I/II	II
HS (5)-2	4.3	I	II

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TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

M220 ISO BREAK	LINE DESIGNATION	PIPE DIA.	MAX FORCE (KIPS) OR THRUST vs TIME GRAPH	PLAN LOCAT. FIGURE
120-1	RCIC (13)-4	4"	FIG 3.6-65	FIG 3.6-49
120-2	RCIC (13)-4	4"	FIG 3.6-65	FIG 3.6-49
120-3	RCIC (13)-4	4"	FIG 3.6-65	FIG 3.6-49
120-4	RCIC (13)-4	4"	FIG 3.6-65	FIG 3.6-48
120-5	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-48
120-6	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-48
120-7	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-48
120-8	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-47
120-11	RCIC (13)-4	4"	FIG 3.6-65	FIG 3.6-49
120-12	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-47
120-13	RCIC (13)-4	4"	FIG 3.6-63	FIG 3.6-47
126-1	RWCU (1)-4	6"	FIG 3.6-80	FIG 3.6-51
126-2	RWCU (1)-4	6"	FIG 3.6-75	FIG 3.6-50
126-3	RWCU (1)-4	2"	FIG 3.6-72	FIG 3.6-50
126-5	RWCU (1)-4	2"	FIG 3.6-72	FIG 3.6-50
126-6	RWCU (1)-4	4"	FIG 3.6-81	FIG 3.6-51
126-51	RWCU (1)-4	4"	FIG 3.6-71	FIG 3.6-50
126-52	RWCU (1)-4	6"	FIG 3.6-80	FIG 3.6-51
126-53	RWCU (1)-4	6"	FIG 3.6-80	FIG 3.6-51
128-7	RWCU (2)-4	4"	FIG 3.6-84	FIG 3.6-51
128-8	RWCU (2)-4	6"	FIG 3.6-68	FIG 3.6-51
128-9	RWCU (2)-4	6"	FIG 3.6-68	FIG 3.6-51
128-10	RWCU (2)-4	6"	FIG 3.6-68	FIG 3.6-50
128-11	RWCU (2)-4	6"	FIG 3.6-67	FIG 3.6-49
128-61	RWCU (2)-4	6"	FIG 3.6-68	FIG 3.6-49

TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

M200-150 -BREAK	LINE DESIGNATION	PIPE DIA.	MAX FORCE (KIPS) OR THRUST VS TIME GRAPH	PLAN LOCATION FIGURE
129-39	RWCU(3)-4	6"	FIG 3.6-77	FIG 3.6-50
129-41	RWCU(3)-4	6"	FIG 3.6-77	FIG 3.6-50
129-42	RWCU(3)-4	6"	FIG 3.6-77, 73	FIG 3.6-50
129-43	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
129-44	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
129-45	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
129-47	RWCU(3)-4	3"	FIG 3.6-73	FIG 3.6-50
129-48	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
129-50	RWCU(3)-4	3"	FIG 3.6-73	FIG 3.6-50
129-54	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
129-55	RWCU(3)-4	4"	FIG 3.6-73	FIG 3.6-50
134-1	MS(20)-4	2"	2.85	FIG 3.6-44
134-2	MS(20)-4	2"	2.85	FIG 3.6-44
134-3	MS(20)-4	2"	2.85	FIG 3.6-44
134-4	MS(20)-4	2"	2.85	FIG 3.6-44
134-35	MS(20)-4	3"	2.85, 6.88	FIG 3.6-44
134-36	MS(20)-4	3"	6.88	FIG 3.6-44
139-1	AS(11)-2	3"	FIG 3.6-97	FIG 3.6-43
139-3	AS(11)-2	3"	FIG 3.6-93	FIG 3.6-43
139-4	AS(11)-2	4"	FIG 3.6-87	FIG 3.6-43
139-7	AS(11)-2	4"	FIG 3.6-87	FIG 3.6-43
139-18	AS(11)-2	4"	12.6, 3.2,	FIG 3.6-43
139-19	AS(11)-2	4"	FIG 3.6-87	FIG 3.6-43
139-20	AS(11)-2	4"	FIG 3.6-87	FIG 3.6-43
139-21	AS(11)-2	4"	FIG 3.6-87	FIG 3.6-43

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

M200-150 BREAK	LINE DESIGNATION	PIPE DIA.	MAX FORCE (KIPS) or FORCE VS TIME GRAPH	PLAN LOCATION FIGURE
141-10	AS (10)-2	6"	7.28	FIG 3.6-43
141-11	AS (10)-2	6"	7.28	FIG 3.6-43
141-12	AS (10)-2	6"	7.28	FIG 3.6-43
141-20	AS (10)-2	8"	12.61	FIG 3.6-43
141-21	AS (7)-2	8"	12.61	FIG 3.6-43
142-20	RWCU (1)-4	4"	FIG 3.6-85	FIG 3.6-51
142-21	RWCU (1)-4	4"	FIG 3.6-85	FIG 3.6-51
142-22	RWCU (1)-4	4"	FIG 3.6-85	FIG 3.6-51
142-23	RWCU (1)-4	4"	FIG 3.6-85	FIG 3.6-51
144-24	RWCU (1)-3	4"	13.34	FIG 3.6-53
144-26	RWCU (1)-3	4"	13.34	FIG 3.6-51
144-27	RWCU (1)-3	4"	13.34	FIG 3.6-51
144-28	RWCU (1)-3	4"	13.34	FIG 3.6-51
144-29	RWCU (1)-3	6"	30.25	FIG 3.6-51
144-31	RWCU (2)-3	4"	13.34	FIG 3.6-51
144-33	RWCU (2)-3	6"	30.25	FIG 3.6-51
144-36	RWCU (2)-3	4"	13.34	FIG 3.6-53
144-56	RWCU (1)-3	4"	13.34	FIG 3.6-51
144-57	RWCU (2)-3	6"	30.12	FIG 3.6-51
144-58	RWCU (1)-3	4"	13.28	FIG 3.6-51
144-59	RWCU (2)-3	6"	30.12	FIG 3.6-49
144-60	RWCU (5)-3	6"	30.12	FIG 3.6-49

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TABLE 3.6-6.

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

M200-150 BREAK	LINE DESIGNATION	PIPE DIA.	MAX FORCE (KIPS) OR THRUST VS TIME FIGURE	LOCATION PLAN FIGURE
148-1	HS (9)-2	3"	FIG 3.6-97	FIG 3.6-60
148-2	HS (1)-2	4"	FIG 3.6-97	FIG 3.6-60
148-3	AS (11)-2	3"	FIG 3.6-97	FIG 3.6-60
148-5	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-6	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-7	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-8	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-9	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-10	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-11	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-12	HS (5)-2	2"	FIG 3.6-97	FIG 3.6-60
148-30	HS (5)-2	3"	FIG 3.6-97	FIG 3.6-60.
149-2	HCO (11)-1	4"	.182	FIG 3.6-58
149-5	HCO (11)-2	3"	.182	FIG 3.6-58
149-30	HCO (11)-2	3"	.182	FIG 3.6-58
149-31	HCO (11)-2	3"	.182	FIG 3.6-58
149-32	HCO (11)-2	3"	.182	FIG 3.6-58
149-33	HCO (11)-2	3"	.182	FIG 3.6-58
149-34	HCO (11)-2	3"	.182	FIG 3.6-60
335-1	RFW (1)-4	24"	433.12	
335-3	RFW (1)-4	24"	433.12	
335-5	RFW (1)-4	24"	433.12	
335-6	RFW (1)-4	24"	433.12.	

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TABLE 3.6-6

DESIGN BASIS BREAK LOCATIONS OUTSIDE PRIMARY CONTAINMENT

M200 ISO BREAK	LINE DESIGNATION	PIPE DIA.	MAX FORCE (KIPS) OR THRUST VS TIME FIGURE	PLAN LOCATION FIGURE
342-13	HS (1) -1	6"	7.28	FIG 3.6-43
342-14	AS (9)-2	4"	3.21	FIG 3.6-43
342-25	AS (1)-2	8"	12.6	N/A
342-26	SS (1)-2	8"	12.6	N/A
342-27	AS (9)-2	4"	3.21	FIG 3.6-43
315-8	MS(1)-4	26"	444.5	FIG 3.6-44
315-30	MS(1)-4	26"	432.2	FIG 3.6-44
400-11	MS(1)-4	26"	444.5	FIG 3.6-44
400-33	MS(1)-4	26"	432.2	FIG 3.6-44
401-14	MS(1)-4	26"	444.5	FIG 3.6-44
401-30	MS(1)-4	26"	432.2	FIG 3.6-14
402-18	MS(1)-4	26"	444.5	FIG 3.6-44
402-31	MS(1)-4	26"	432.2	FIG 3.6-44
440-1	CO (3)-2	2½"	1.63	N/A
440-2	CO (3)-2	2½"	1.63	N/A
440-3	CO (3)-2	2½"	1.63	N/A
447-19	HS (5)-1	6"	1.82	N/A
447-25	HS (5)-1	6"	1.82	N/A
447-26	HS (5)-1	6"	1.82	N/A
447-27	HS (5)-1	6"	1.82	N/A

TABLE 3.6-6

DESIGN BASIS BREAK LOCATION OUTSIDE PRIMARY CONTAINMENT

M200 150 BREAK	LINE DESIGNATION	PIPE DIA	MAX FORCE (KIPS) OR THRUST VS. TIME FIGURE	PLAN LOCATION FIGURE
448-15	HS(1)-1	6"	1.82	N/A
448-16	HS(1)-1	6"	1.82	N/A
448-17	HS(1)-1	6"	1.82	N/A
448-18	HS(1)-1	6"	1.82	N/A
448-19	HS(1)-1	6"	1.82	N/A
448-20	HS(1)-1	6"	1.82	N/A
448-21	HS(1)-1	6"	1.82	N/A
448-22	HS(1)-1	4"	.802	N/A
448-23	HS(1)-1	4"	.802	N/A
448-24	HS(1)-1	2"	.186	N/A
449-13	HCO(5)-1	3"	.093	N/A
449-14	HCO(5)-1	3"	.093	N/A
449-15	HCO(5)-1	3"	.093	N/A
449-16	HCO(5)-1	3"	.093	N/A
449-17	HCO(5)-1	3"	.093	N/A
449-18	HCO(5)-1	3"	.093	N/A
449-19	HCO(5)-1	3"	.093	N/A
449-20	HCO(5)-1	3"	.093	N/A
449-21	HCO(5)-1	3"	.093	N/A
449-22	HCO(5)-1	3"	.093	N/A

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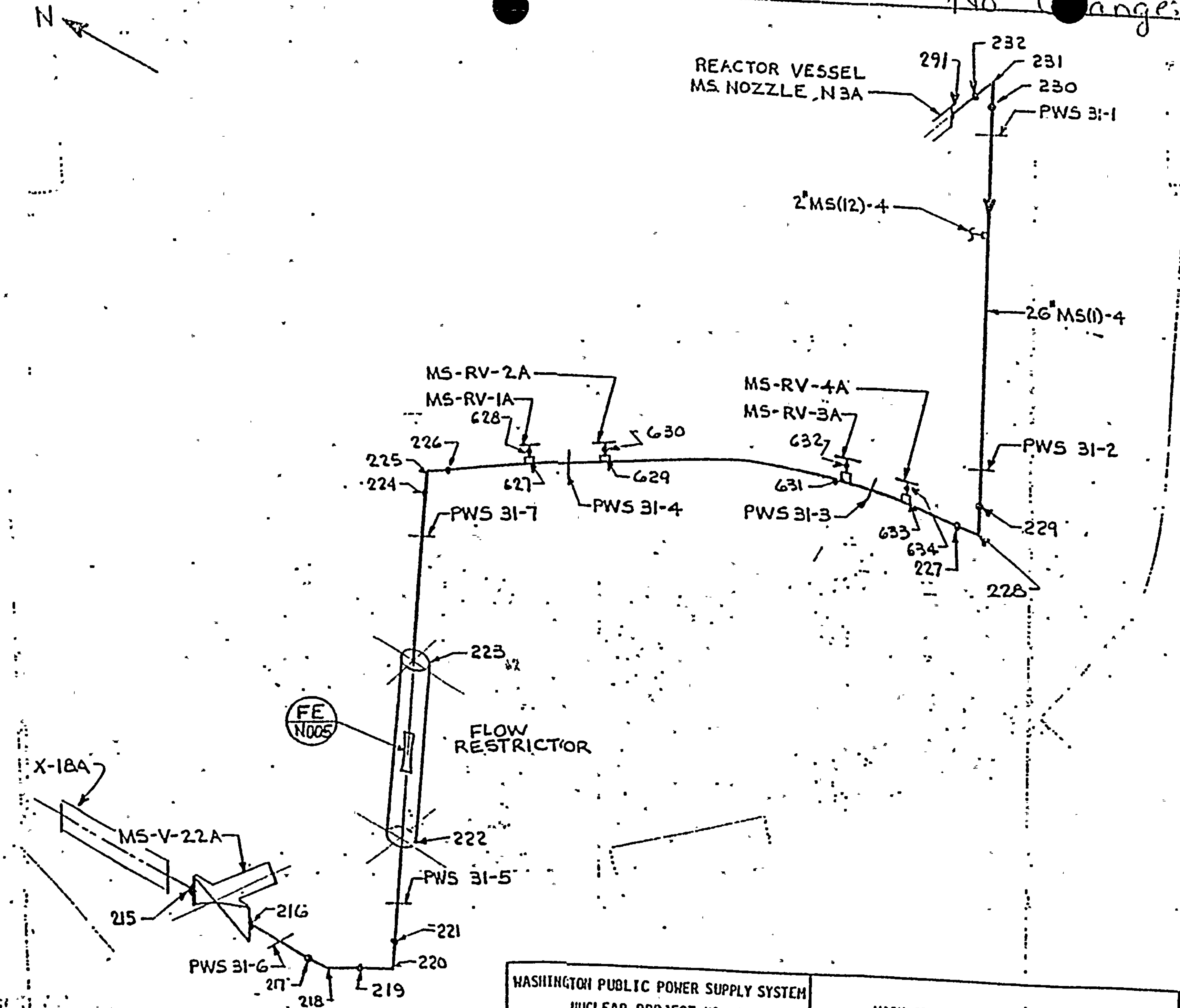
TABLE 3.6-6.

DESIGN BASIS BREAK LOCATION OUTSIDE PRIMARY CONTAINMENT

M200 ISO BRK-1	LINE DESIGNATION	PIPE DIA	MAX FORCE (KIPS) or THRUST VS TIME FIGURE	PLAN LOCATION FIGURE
450-23	HCO(5)-1	3"	.093	N/A
450-21	HCO(5)-1	3"	.093	N/A
450-25	HCO(5)-1	2½"	.060	N/A
450-26	HCO(5)-1	2½"	.060	N/A
450-27	HCO(5)-1	2½"	.060	N/A
450-28	HCO(5)-1	2½"	.060	N/A
451-6	MS(9)-4	3"	6.88	FIG 3.6-44
451-30	MS(9)-4	3"	6.88	FIG 3.6-44

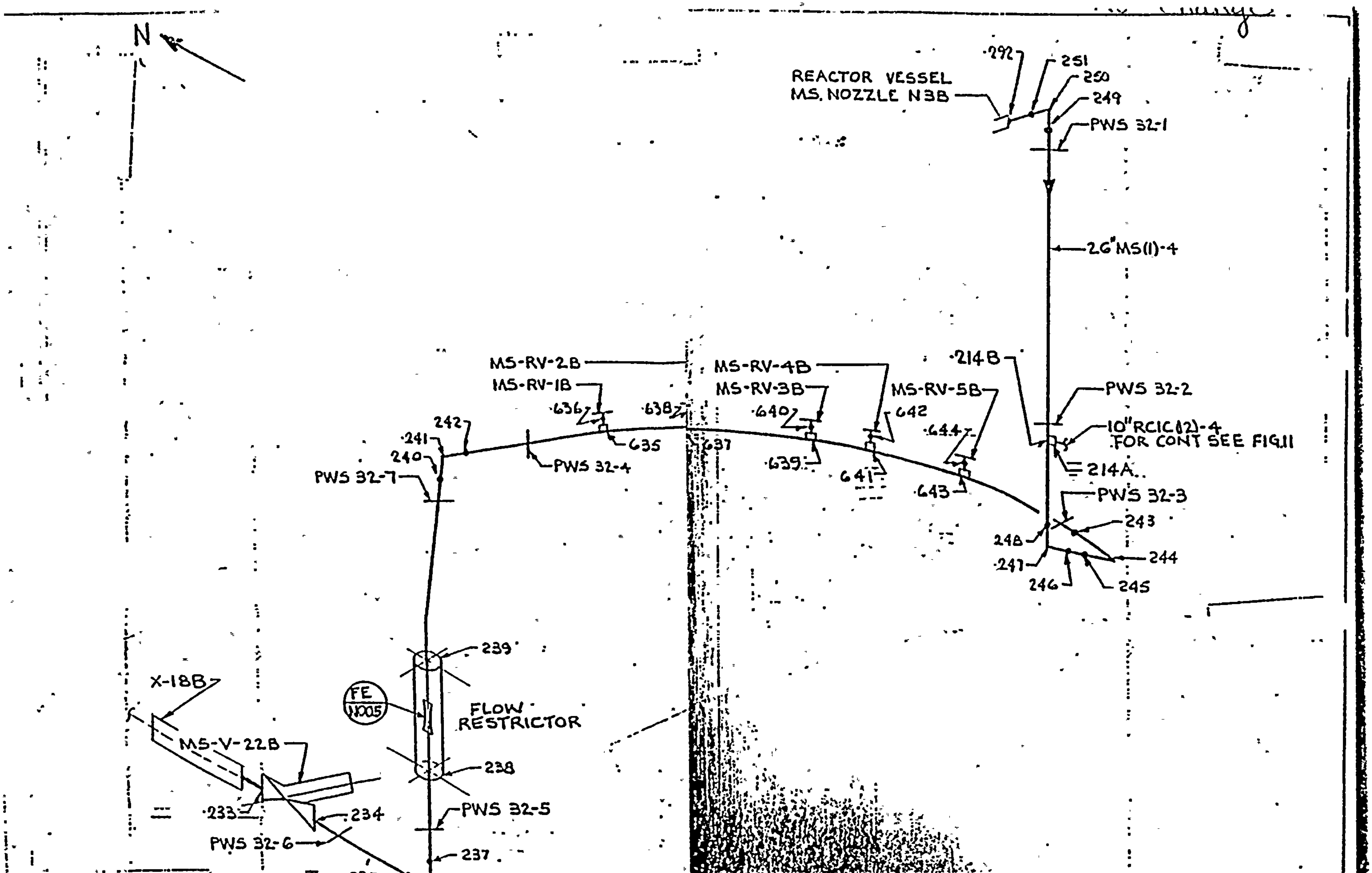
No Changes



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM LOOP A ISOMETRIC	FIGURE 3.6-3

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
~~Node 215~~	~~Node 218~~
Node 216	Node 220
~~Node 217~~	Node 225
Node 219	Node 228
Node 221	Node 231
~~Node 222~~	NODE 633
~~Node 223~~	NODE 631
Node 224	NODE 629
Node 226	NODE 627
Node 227	
Node 229	
Node 230	
Node 232	
Node 291	
~~Node 628~~	
~~Node 630~~	
~~Node 632~~	
~~Node 634~~	
~~NODE 633~~	
~~NODE 631~~	
~~NODE 629~~	
~~NODE 627~~	



MAIN STEAM LOOP B ISOMETRIC

FIGURE 3.6-13

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

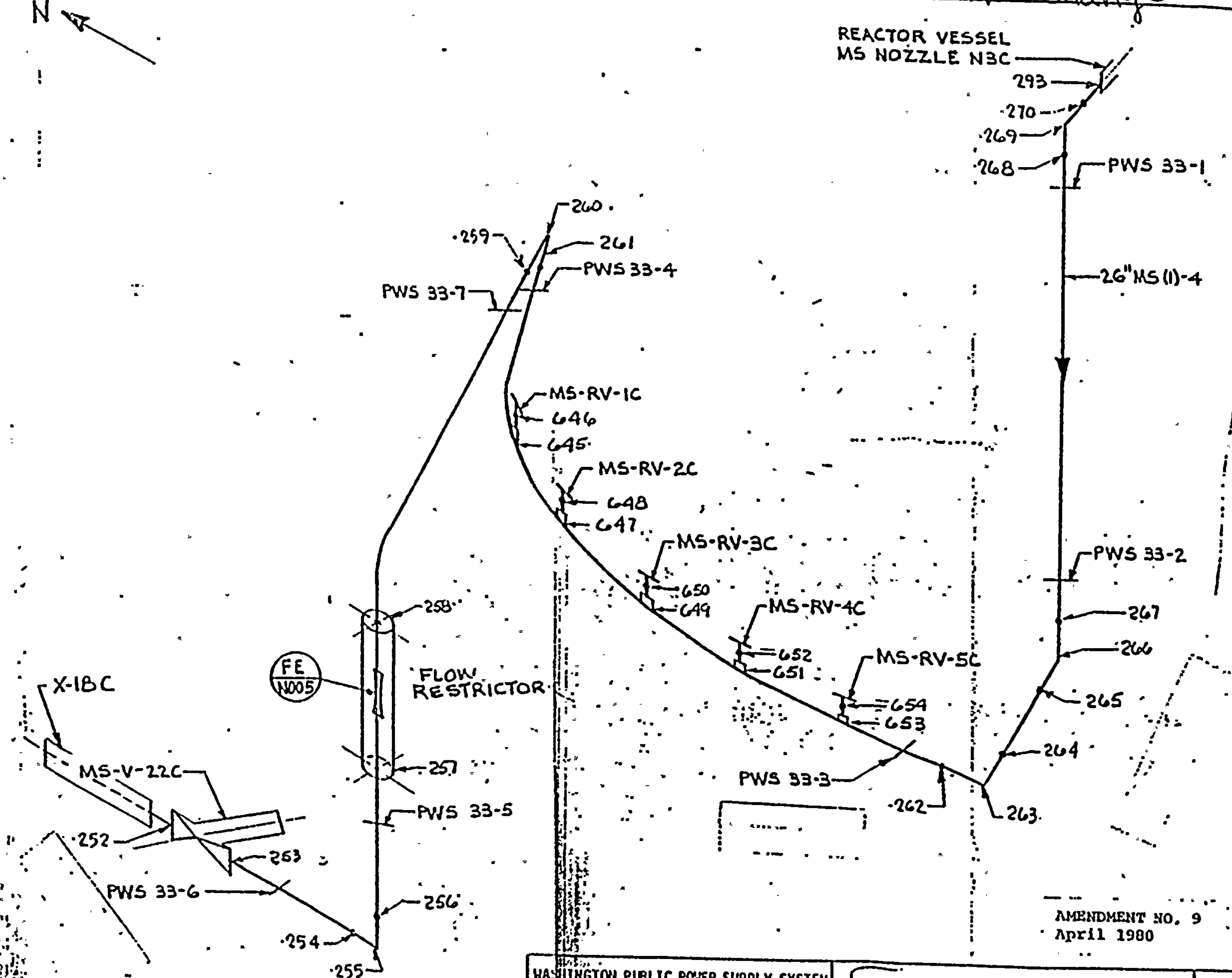
CIRCUMFERENTIAL BREAKS

~~Node 233~~
~~Node 234~~
Node 235
Node 237
~~Node 238~~
~~Node 239~~
Node 240
Node 242
~~Node 243~~
~~Node 245~~
Node 246
Node 248
Node 249
Node 251
Node 292
Node 636
~~Node 638~~
Node 640
~~Node 642~~
Node 644
NODE 214A
~~NODE 643~~
~~NODE 641~~
~~NODE 639~~
~~NODE 637~~
~~NODE 635~~

LONGITUDINAL BREAKS

Node 236
Node 241
~~Node 244~~
Node 247
Node 250
NODE 643
NODE 641
NODE 639
NODE 637
NODE 635

No Change



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April 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM LOOP C ISOMETRIC	FIGURE 3.6-14a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

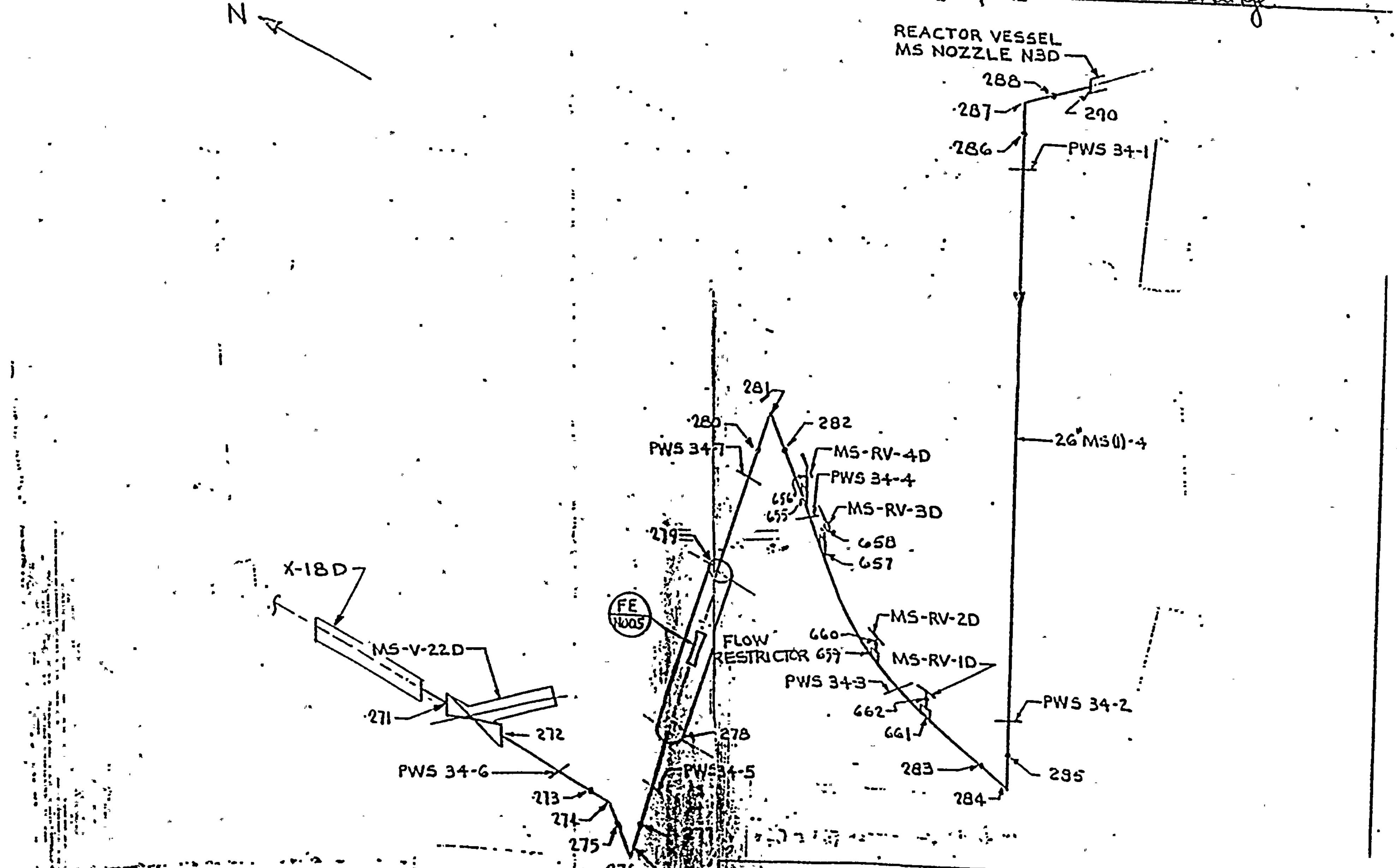
CIRCUMFERENTIAL BREAKS

~~Node 252~~
~~Node 253~~
Node 254
Node 256
~~Node 257~~
~~Node 258~~
Node 259
Node 261
~~Node 262~~
~~Node 264~~
Node 265
Node 267
Node 268
Node 270
Node 293
~~Node 646~~ NODE 646
~~Node 648~~
~~Node 650~~ NODE 650
~~Node 652~~
Node 654

LONGITUDINAL BREAKS

Node 255
Node 260
~~Node 263~~
Node 266
Node 269
NODE 653
NODE 651
NODE 649
NODE 647
NODE 645

No Change

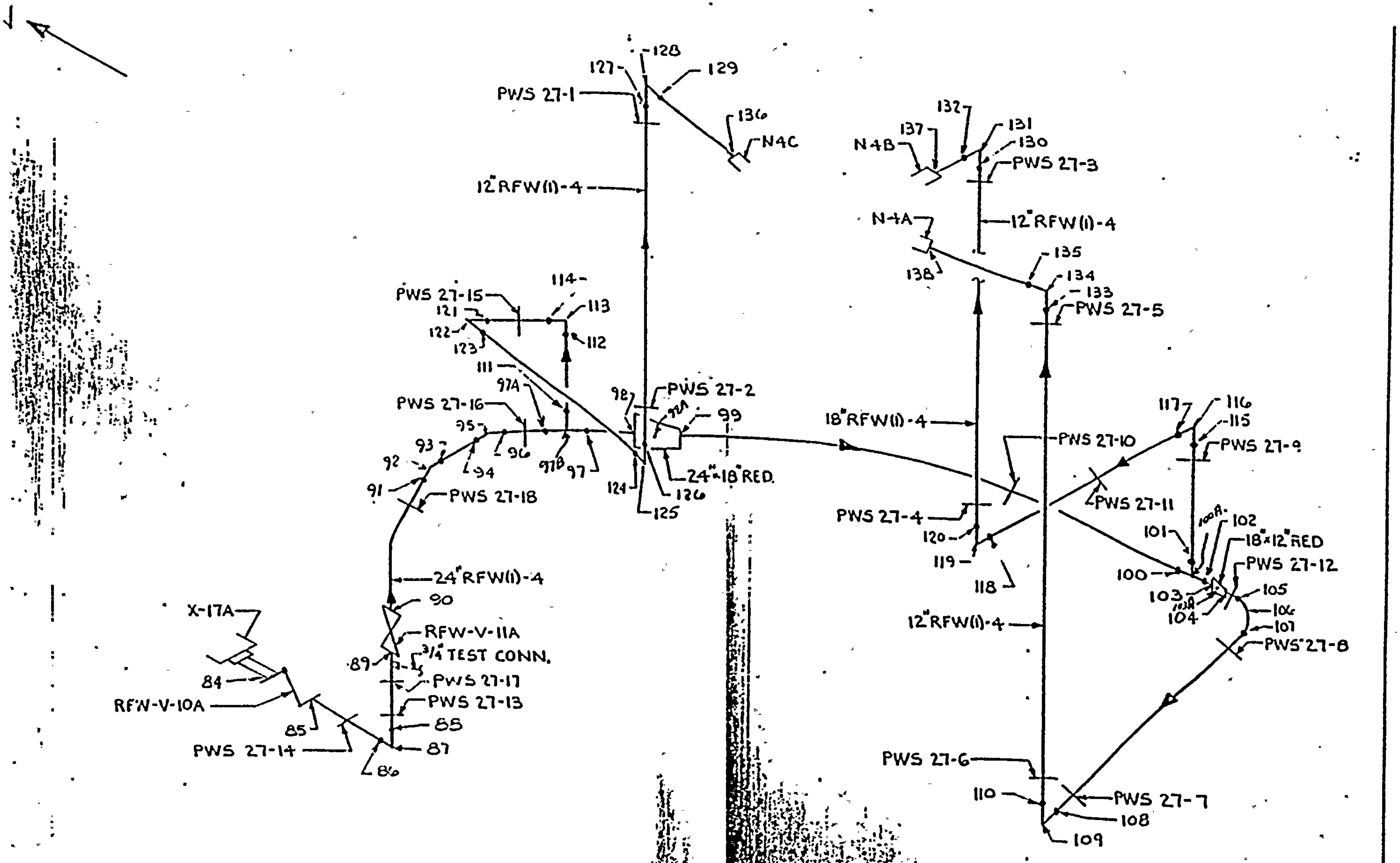


WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

MAIN STEAM LOOP D ISOMETRIC

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
~~Node 271~~	~~Node 274~~
Node 272	Node 276
~~Node 273~~	Node 281
Node 275	Node 284
Node 277	Node 287
~~Node 278~~	NODE 661
~~Node 279~~	NODE 659
Node 280	NODE 657
Node 282	NODE 655
Node 283	
Node 285	
Node 286	
Node 288	
Node 290	
~~Node 656~~	
~~Node 658~~	
~~Node 660~~	
~~Node 662~~	



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR FEEDWATER (LINE 'A') ISOMETRIC

FIGURE 3.6-16

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SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

NODES

85	124	115
86	126	117
88	127	118
89	129	120
90	136	130
91	100	132
93	101	137
94 96 97A	102 103	103
97	104	
111	105	
98	107	
99	108	
112	110	
114	133	
121	135	
123	138	

LONGITUDINAL BREAKS

NODES

87	97B
92	98A
113	100A
122	103A
125	
128	
106	
109	
134	
116	
119	
131	

Note: Break locations based on change in piping flexibility. ~~because stress analysis not complete.~~

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2

REACTOR FEEDWATER (LINE A)

FIG. 3.6-16b

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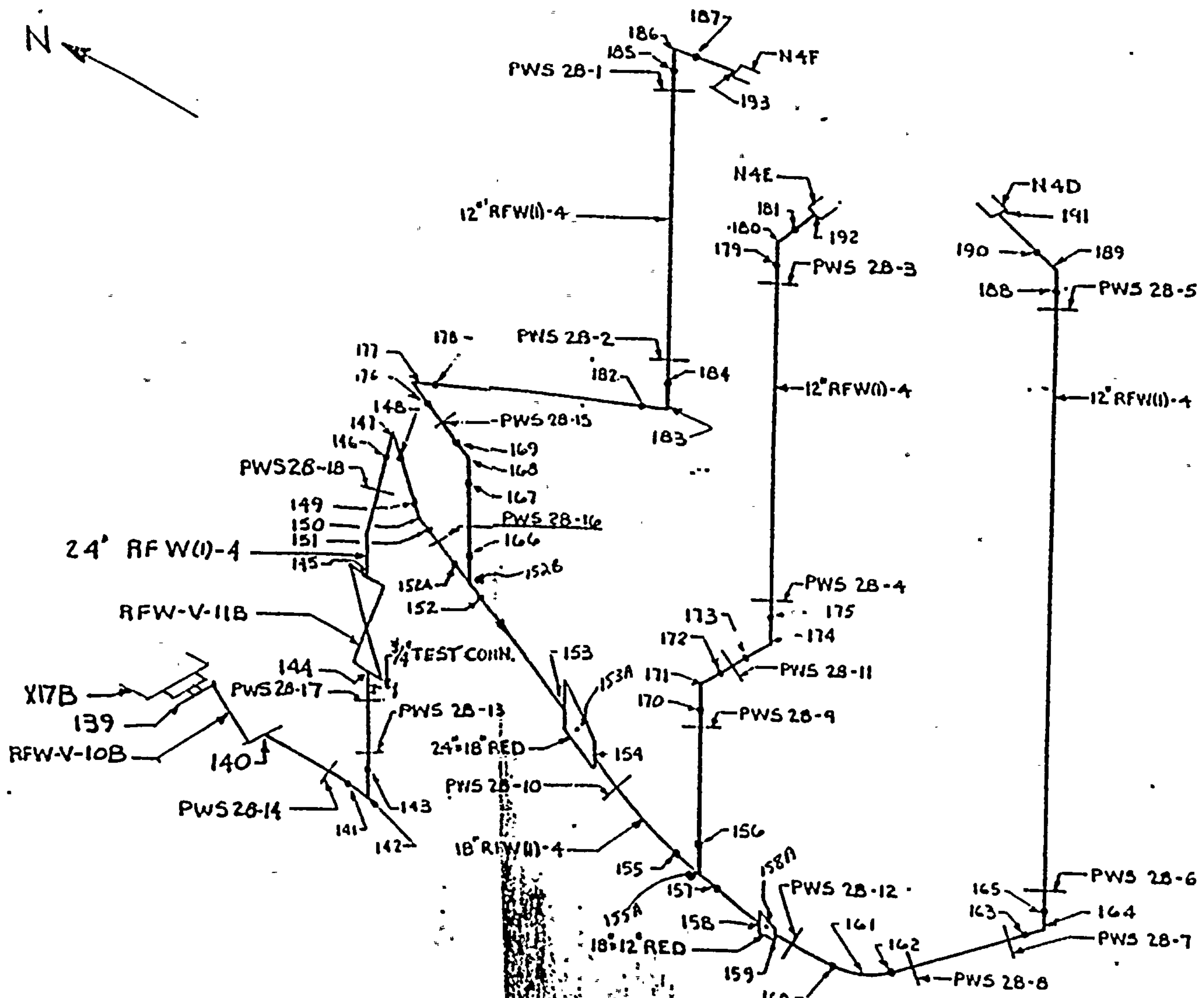

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR FEEDWATER (LINE B)
ISOMETRIC

FIGURE
3.6-17a

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SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAK

NODES

140	154
141	155
143	156
144	157
145	170
146	172
148	173
152 A	175
152	179
166	181
167	192
169	159
176	160
178	162
182	163
184	165
185	188
187	190
193	191
153	158

move

LONGITUDINAL BREAK

NODES

142	152 B
147	153 A
168	155 A
177	15[illegible] A
183	
186	
171	
174	
180	
161	
164	
189	

Note: Break locations based on change in piping flexibility. ~~because stress analysis not complete.~~

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER (LINE B)	FIGURE 3.6-17b

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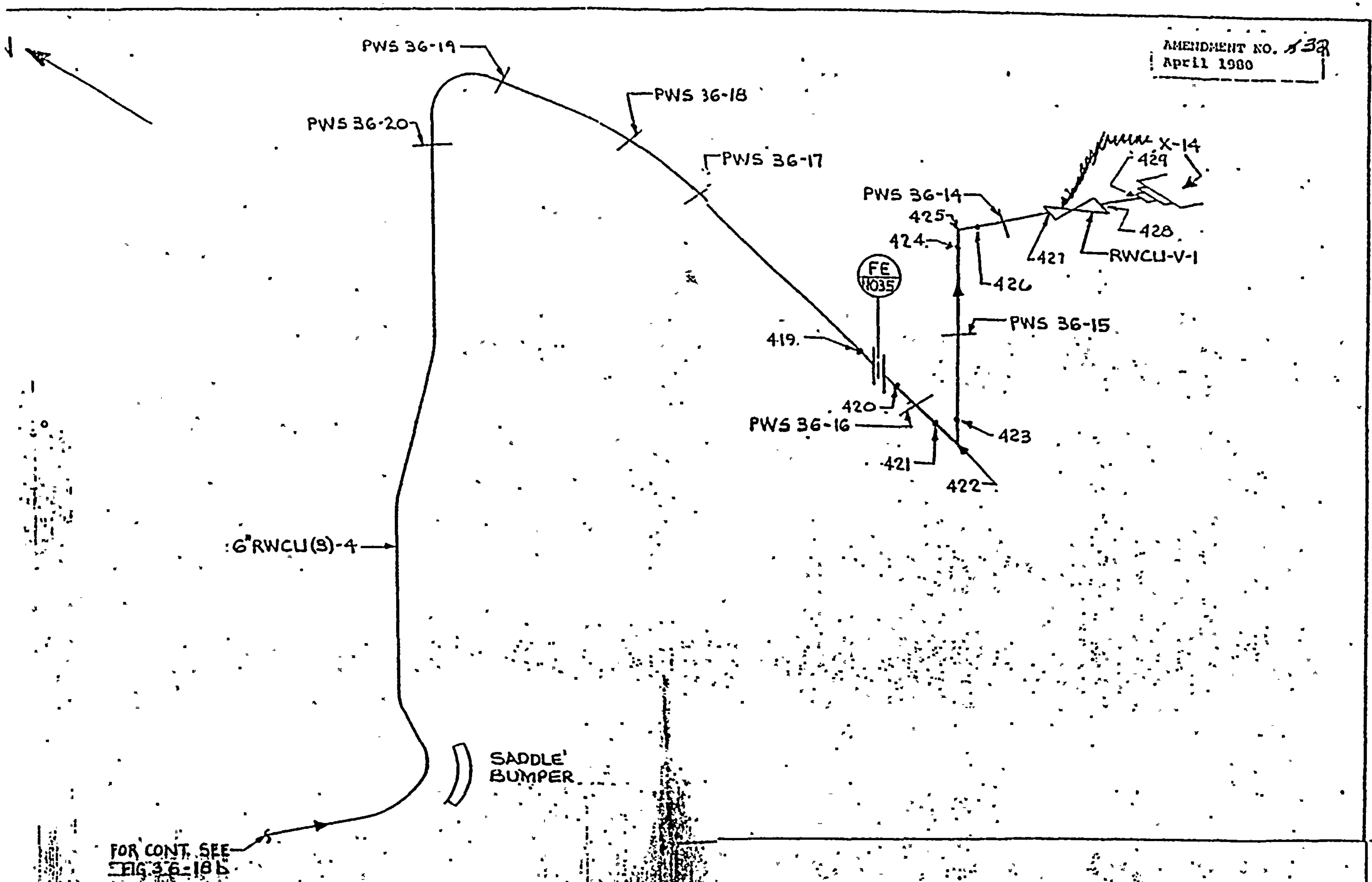
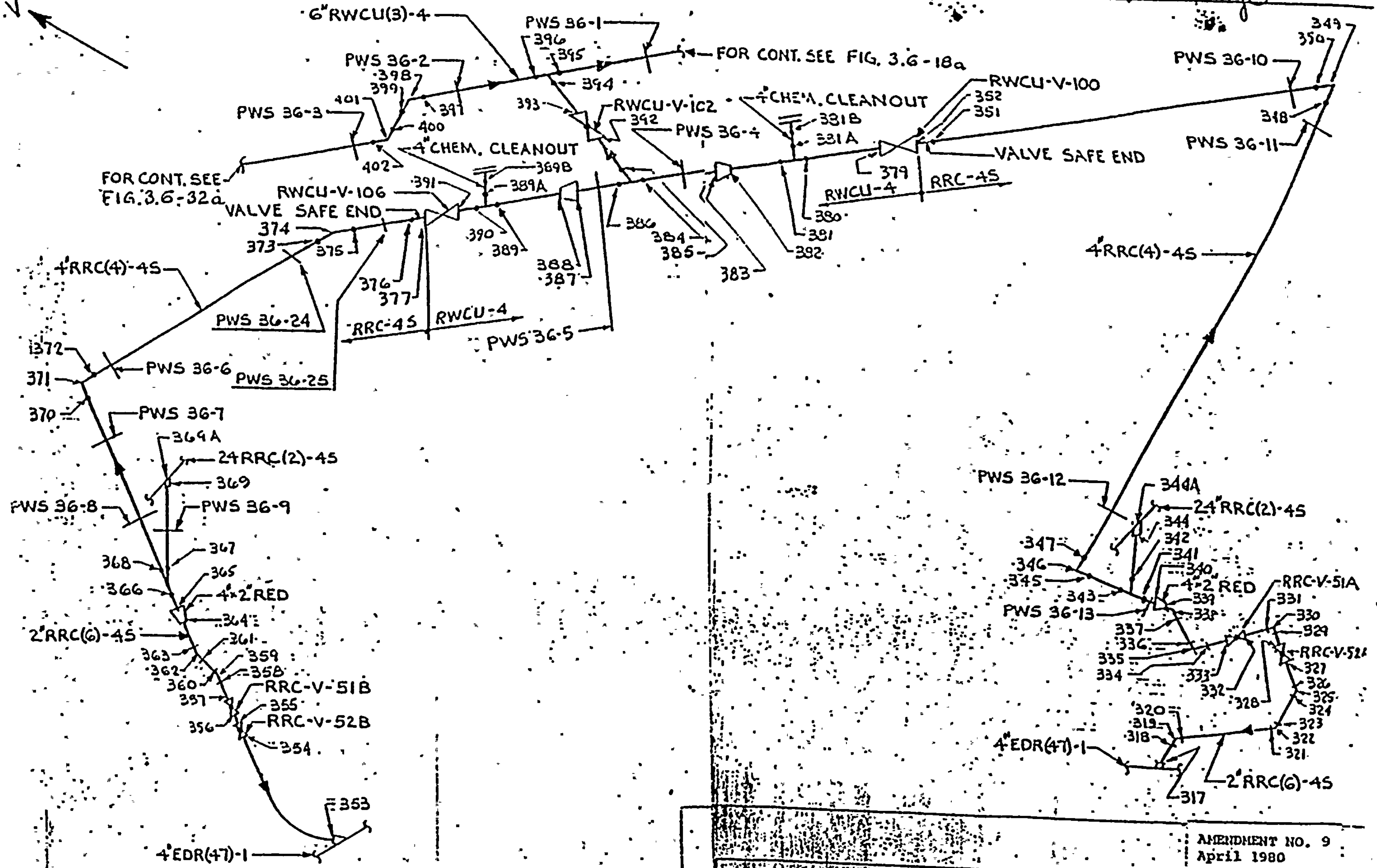


WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR WATER CLEANUP ISOMETRIC

FIGURE 3.6-18a

No Change



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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR WATER CLEANUP ISOMETRIC

FIGURE
3.6-18b

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

~~Node 419~~
~~Node 420~~
~~Node 421~~
~~Node 423~~
~~Node 424~~
~~Node 426~~
Node 427
~~Node 428~~
~~Node 429~~

LONGITUDINAL BREAKS

~~Node 422~~
~~Node 425~~

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR WATER CLEANUP	FIGURE 3.6-18c

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

NODE 333
~~Node 340~~
~~Node 341~~
~~Node 342~~
~~Node 343~~
Node 344
~~Node 345~~
~~Node 347~~
~~Node 348~~
~~Node 350~~
~~Node 352~~
~~Node 365~~
Node 366
Node 367
Node 368
~~Node 369~~
~~Node 370~~
NODE 357

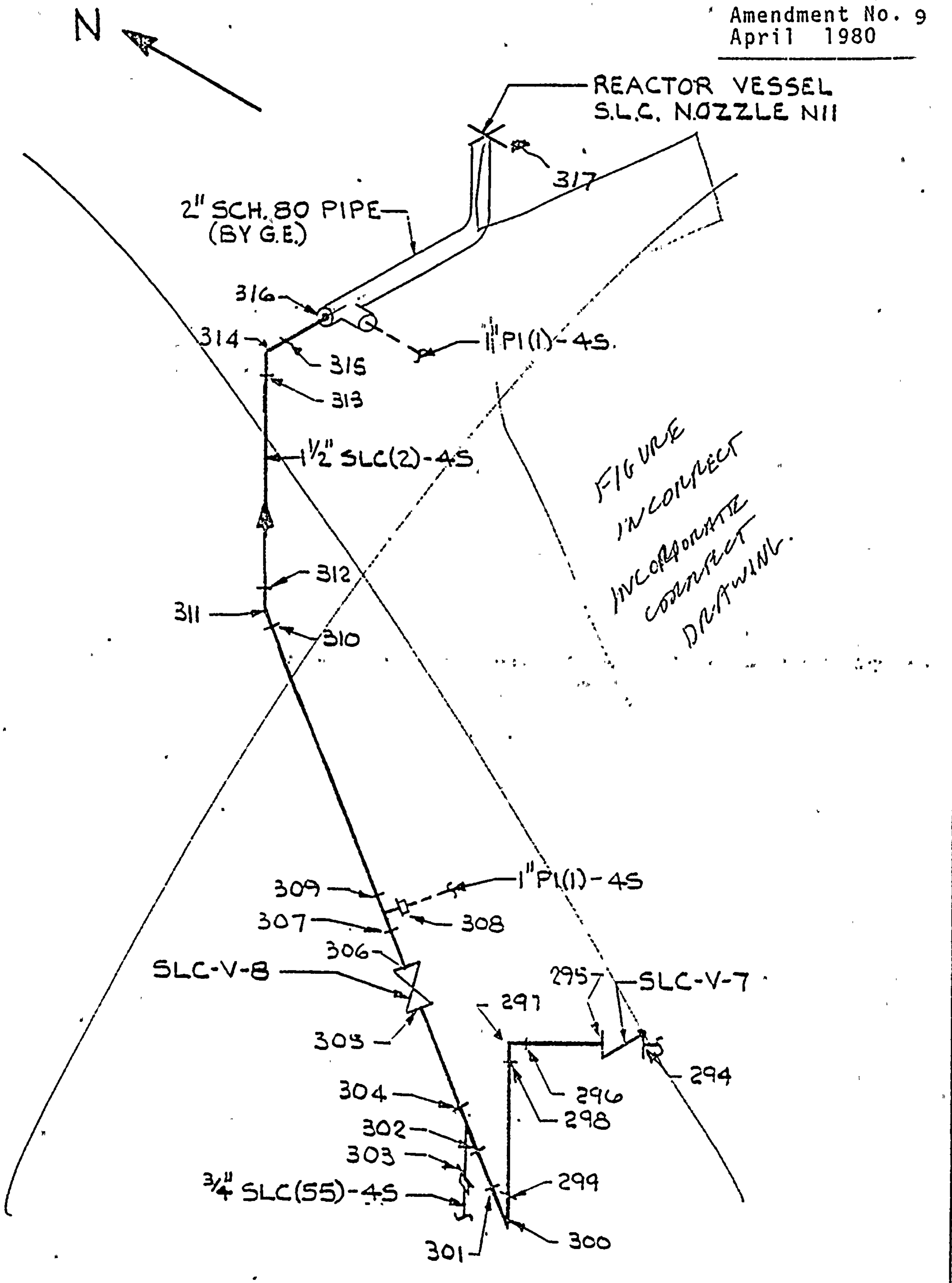
~~Node 372~~
~~Node 373~~
~~Node 375~~
~~Node 377~~
Node 379
~~Node 380~~
~~Node 381~~
~~Node 381A~~
~~Node 381B~~
~~Node 382~~
~~Node 383~~
~~Node 384~~
~~Node 385~~
~~Node 386~~
~~Node 387~~

~~Node 388~~
~~Node 389~~
~~Node 389A~~
~~Node 389B~~
~~Node 390~~
Node 391
~~Node 392~~
~~Node 393~~
Node 394
Node 395
Node 396
~~Node 397~~
~~Node 399~~
~~Node 400~~
~~Node 402~~

LONGITUDINAL BREAKS

~~Node center of [341, 342, 343] (TEE)~~
~~346~~
~~349~~
~~Node center of [384, 385, 386] (TEE)~~
~~374~~
~~371~~
~~Node center of [366, 367, 368] (TEE)~~
~~Node center of [394, 395, 396]~~ (TEE)
~~398~~
~~401~~
~~Node center of [389, 389A, 390]~~ (TEE)
~~Node center of [380, 381, 381A] (TEE)~~

Amendment No. 9
April 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL ISOMETRIC	FIGURE 3.6-19a

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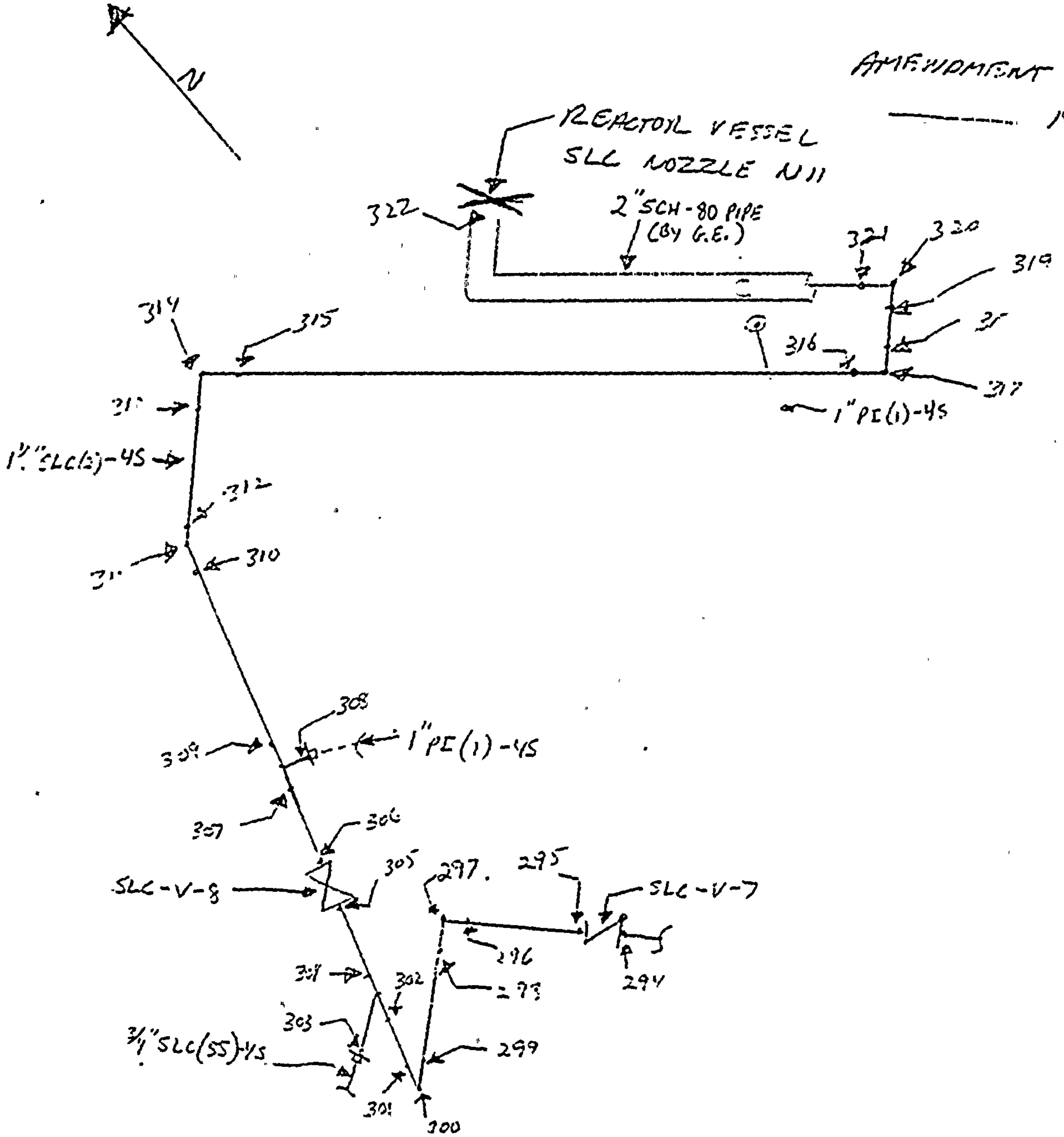

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Amendment No. 33

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR P[illegible] NO 2	STANDBY LIQUID CONTROL ISOMETRIC	FIGURE 3.6-19a

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SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAK

NODE 295
NODE 308
NODE 310
NODE 312
NODE 316
NODE 318
NODE 319
NODE 321
NODE 322

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL	FIGURE 3.6-19b

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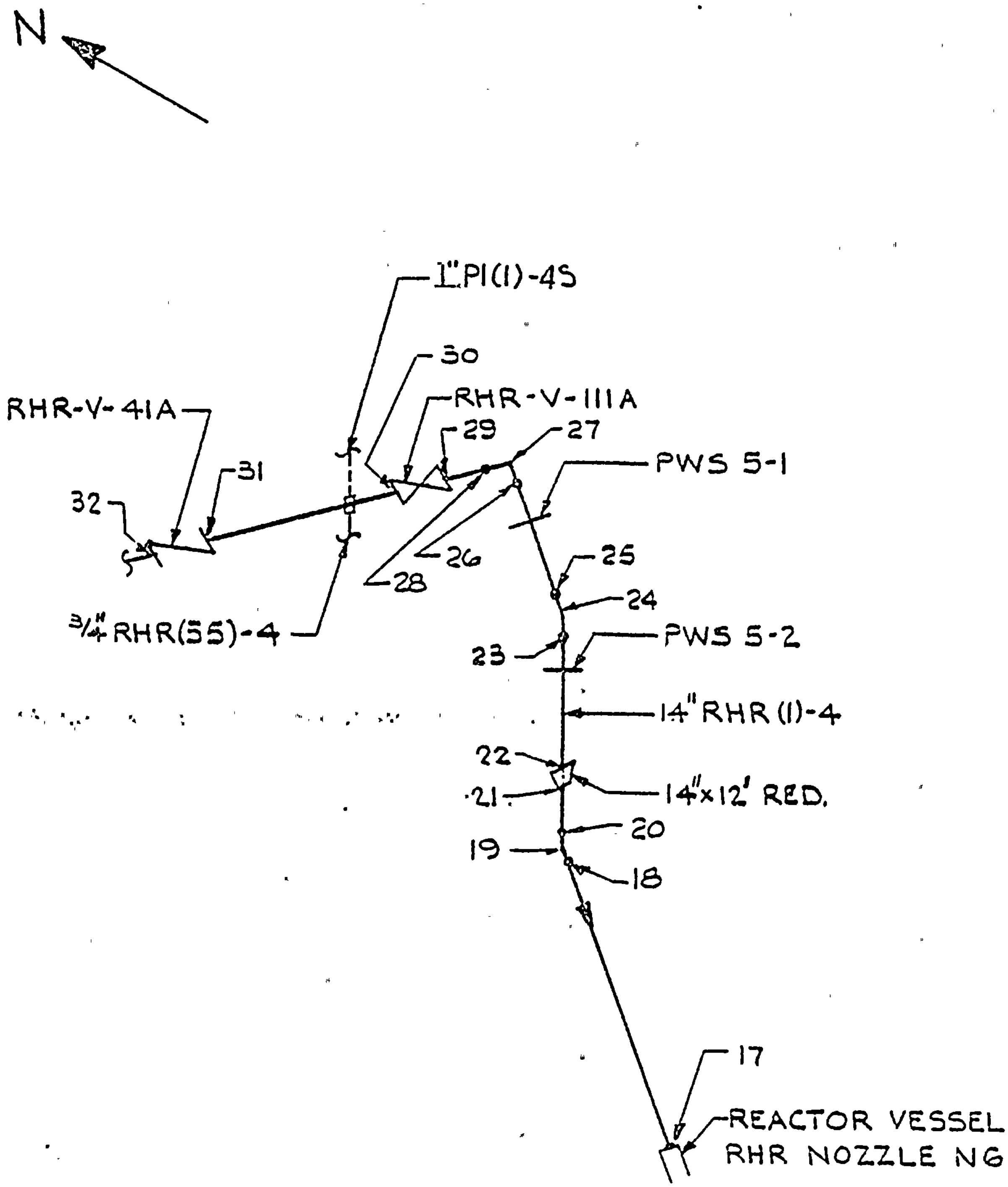
No Change



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDULAL HEAT REMOVAL LPCI MODE (LOOP A) ISOMETRIC	FIGURE 3.6-20a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

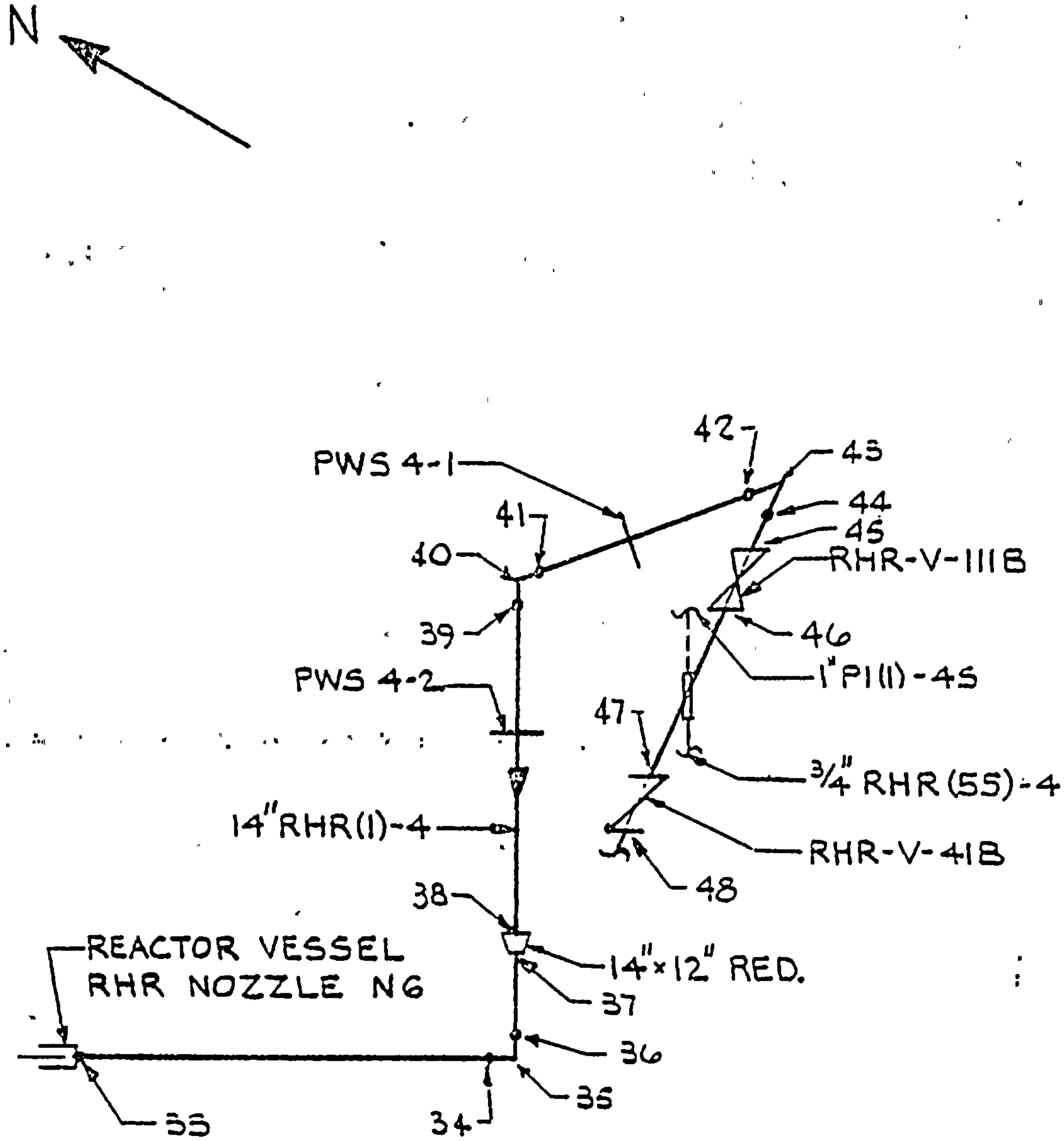
CIRCUMFERENTIAL BREAKS

Node 17
Node 18
Node 20
~~Node 21~~
~~Node 22~~
~~Node 23~~
~~Node 25~~
Node 26
Node 28
~~Node 29~~
~~Node 30~~
Node 31

LONGITUDINAL BREAKS

Node 19
~~Node 24~~
Node 27

No Change

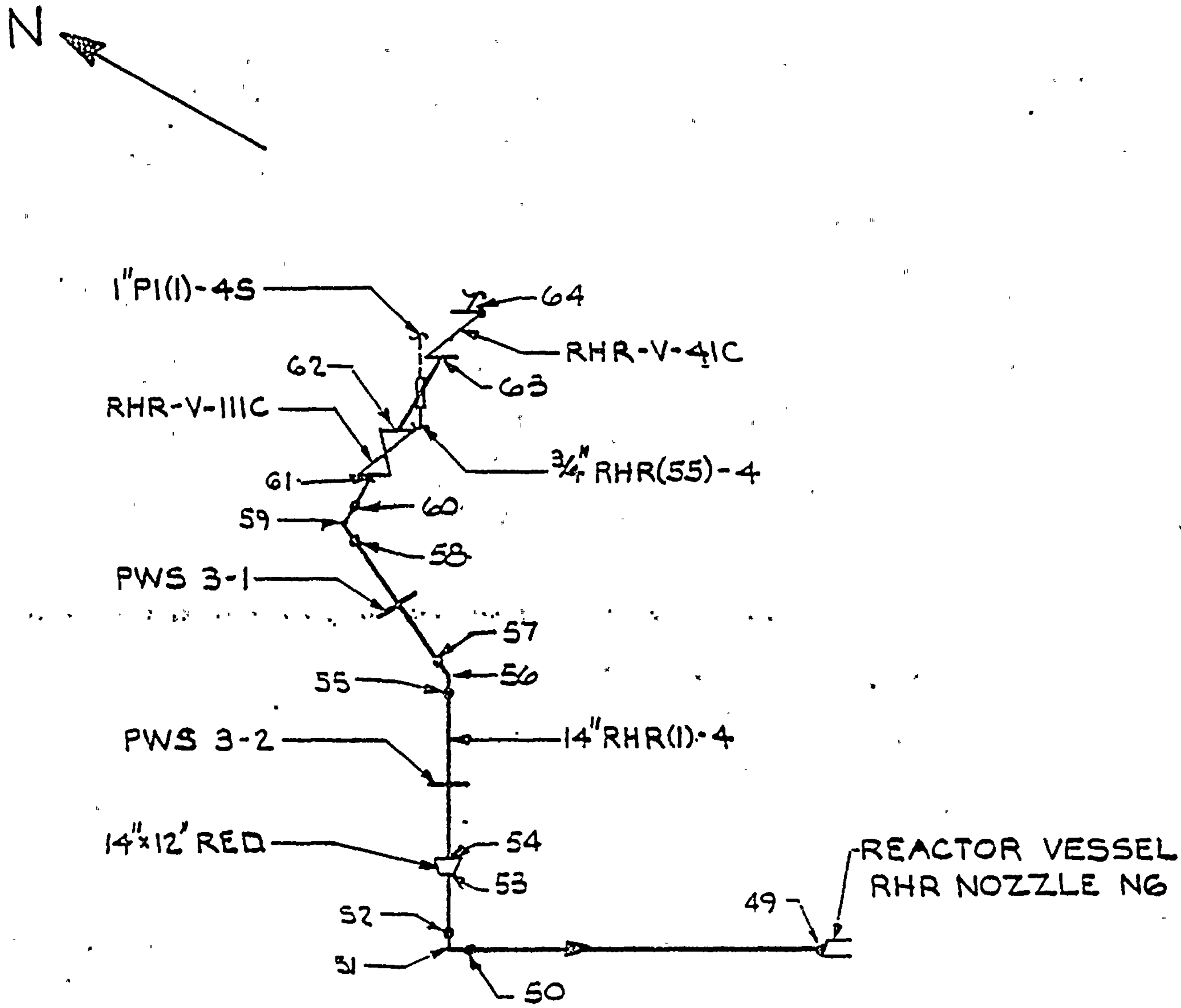


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL LPCI MODE (LOOP B) ISOMETRIC	FIGURE 3.6-21a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 33	Node 35
Node 34	~~Node 40~~
Node 36	Node 43
~~Node 37~~	
~~Node 38~~	
~~Node 39~~	
~~Node 41~~	
Node 42	
Node 44	
~~Node 45~~	
~~Node 46~~	
Node 47	

No Change



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL LPCI MODE (LOOP C) ISOMETRIC	FIGURE 3.6-22a

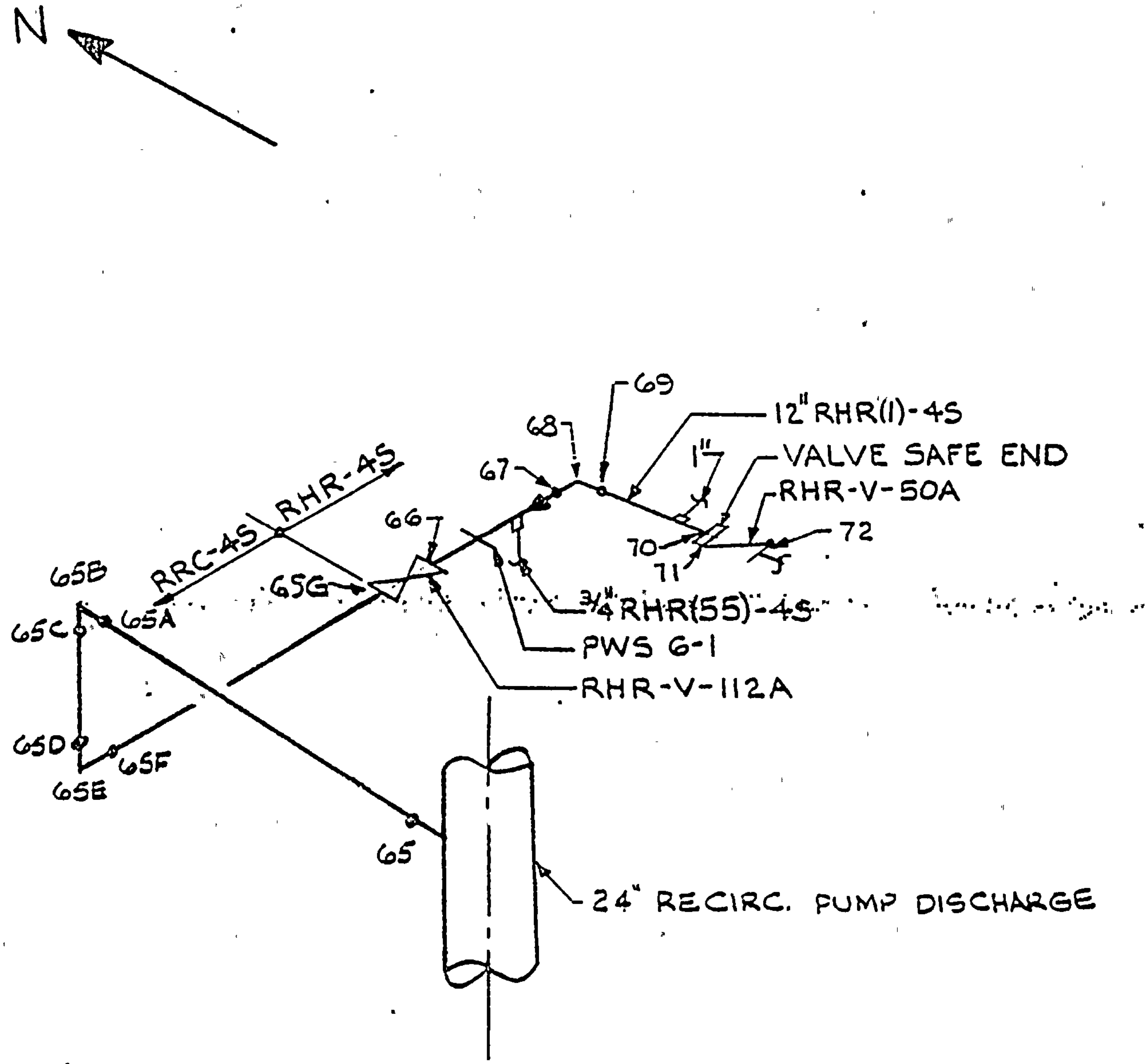
SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 49	Node 51
Node 50	~~Node 56~~
Node 52	Node 59
~~Node 53~~	
~~Node 54~~	
~~Node 55~~	
~~Node 57~~	
Node 58	
Node 60	
~~Node 61~~	
~~Node 62~~	
Node 63	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL LPCI MODE LOOP C	FIGURE 3.6-22b

No Change

Amendment No. 9
April 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SHUTDOWN COOLING (LOOP A) ISOMETRIC	FIGURE 3.6-23a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

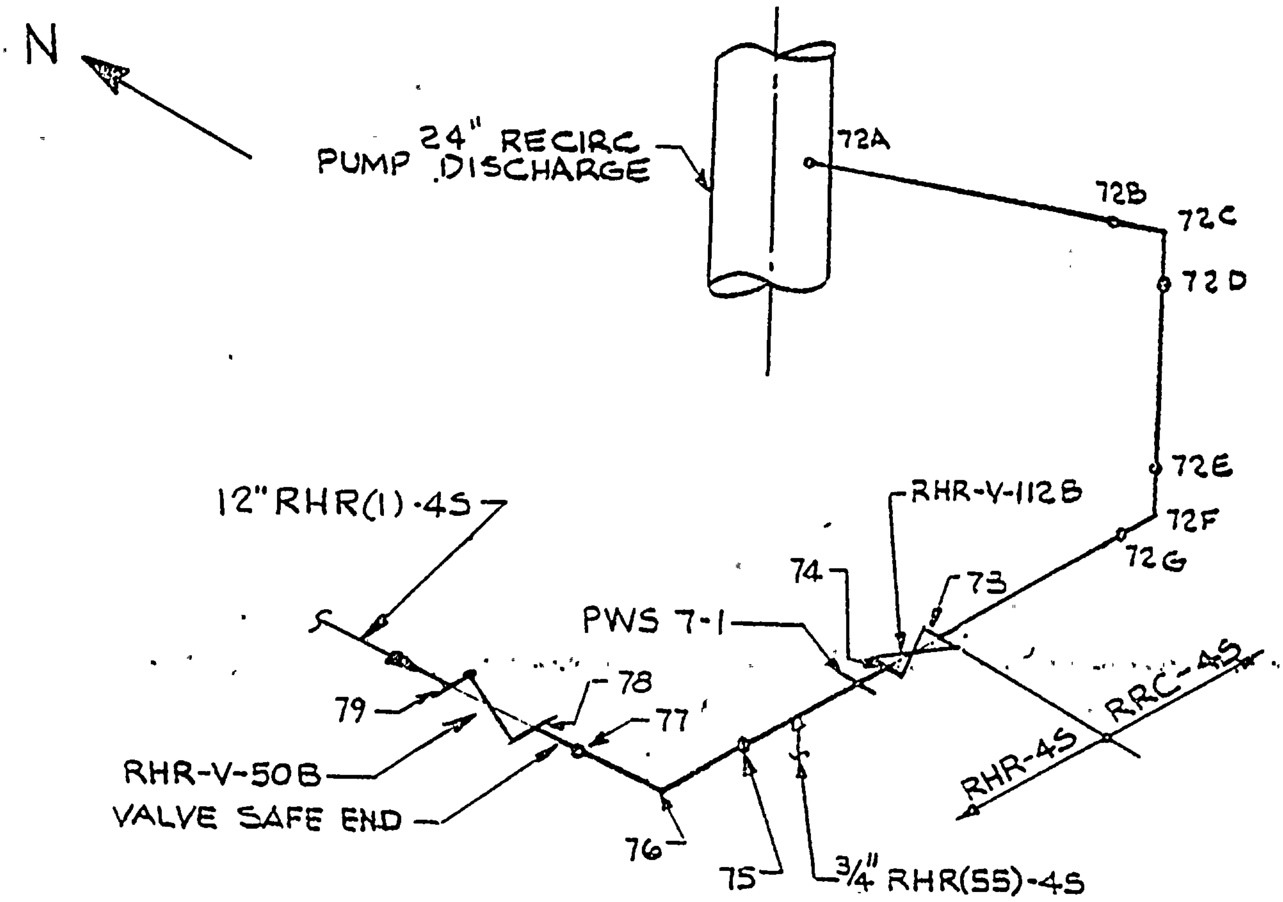
Node 65
Node 65A
Node 65C
~~Node 65D~~
~~Node 65F~~
Node 65G
Node 66
Node 67
Node 69
~~Node 71~~
NODE 70

LONGITUDINAL BREAKS

Node 65B
~~Node 65E~~
Node 68

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SHUTDOWN COOLING LOOP A	FIGURE 3.6-23b

No Change



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SHUTDOWN COOLING (LOOP B) ISOMETRIC	FIGURE 3.6-24a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

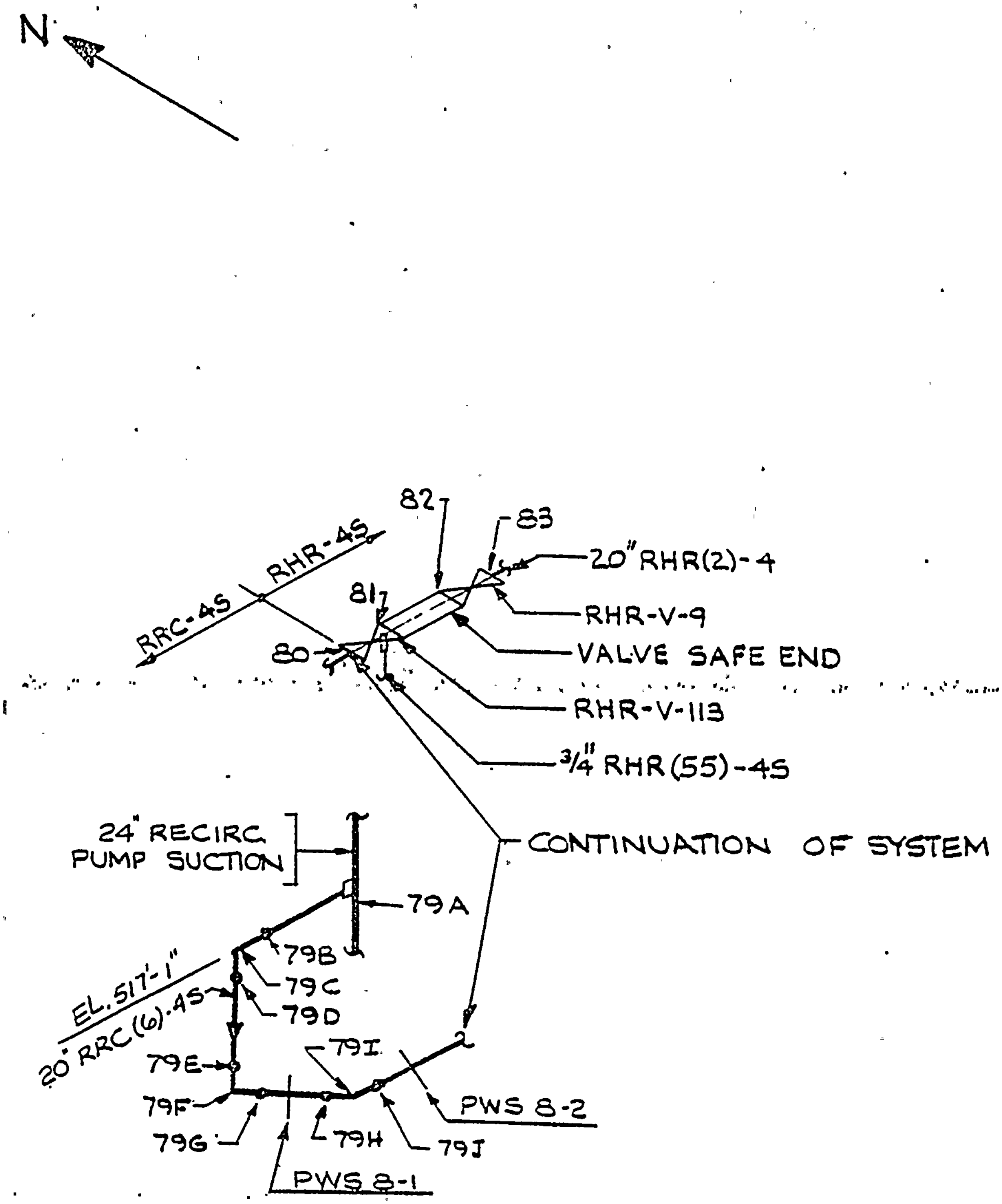
Node 72A
Node 72B
Node 72D
Node 72E
Node 72G
~~Node 73~~
~~Node 74~~
Node 75
~~Node 78~~
NODE 77

LONGITUDINAL BREAKS

Node 72C
Node 72F
Node 76

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SHUTDOWN COOLING LOOP B	FIGURE 3.6-24b

No Change.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SHUTDOWN COOLING SUPPLY ISOMETRIC	FIGURE 3.6-25a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

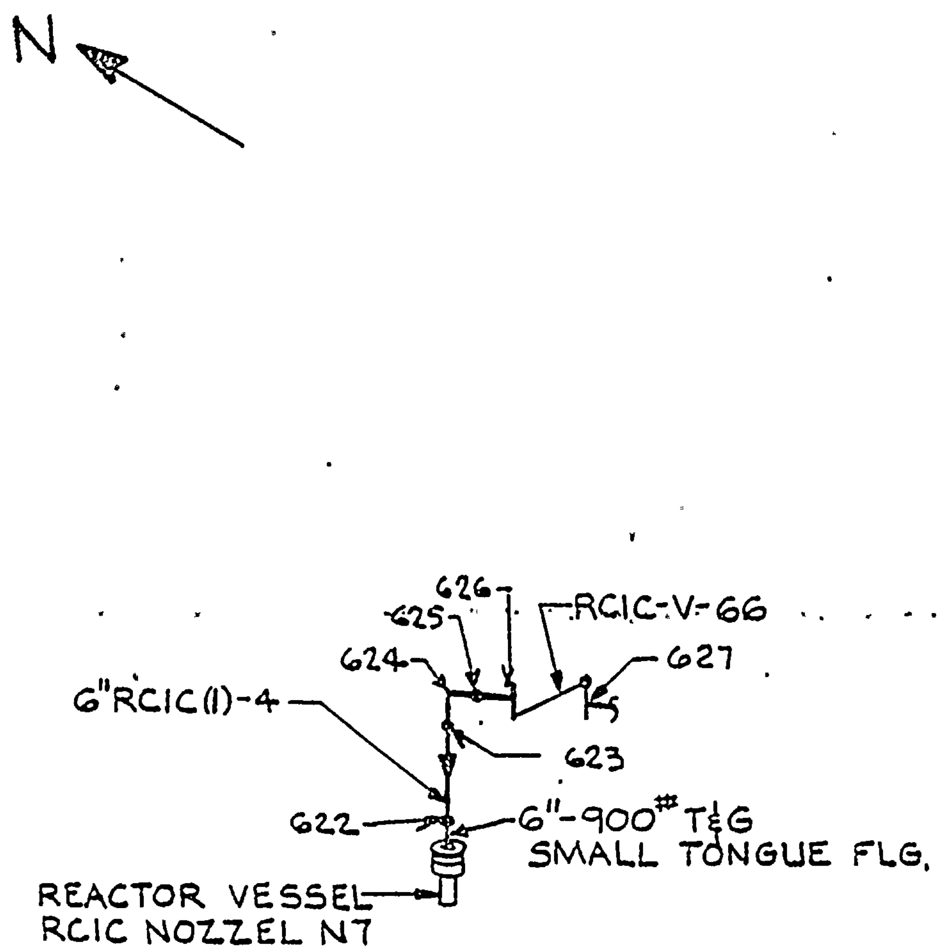
CIRCUMFERENTIAL BREAKS

Node 79A
~~Node 79B~~
~~Node 79D~~
Node 79E
Node 79G
Node 79H
Node 79J
~~Node 80~~
~~Node 81~~
Node 82

LONGITUDINAL BREAKS

~~Node 79C~~
Node 79F
Node 79I

No Change



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC RPV HEAD SPRAY ISOMETRIC	FIGURE 3.6-26a

Amendment No. ~~25~~ 32
~~June 1982~~

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SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 622
Node 623
Node 625
Node 626

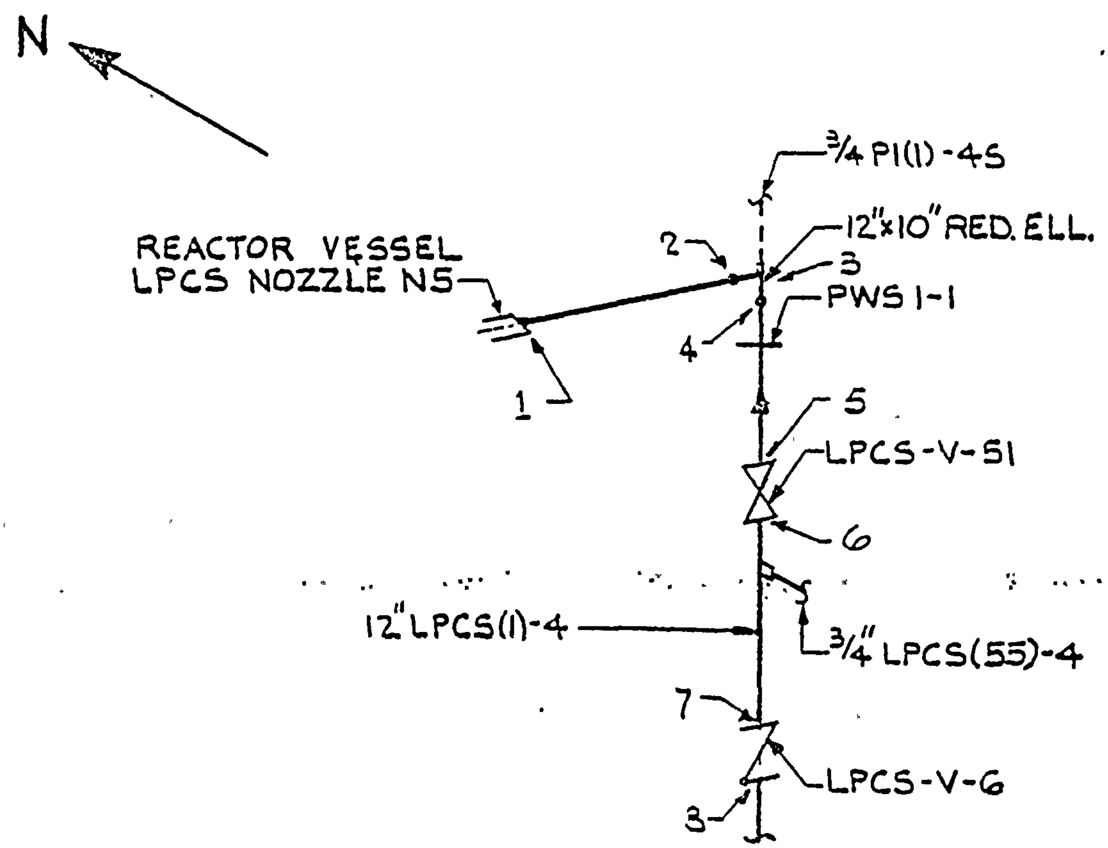
LONGITUDINAL BREAKS

Node 624

RCIC RPV HEAD SPRAY

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	~~RESIDUAL HEAT REMOVAL SHUTDOWN COOLING SUPPLY~~	FIGURE 3.6-26b

No Change



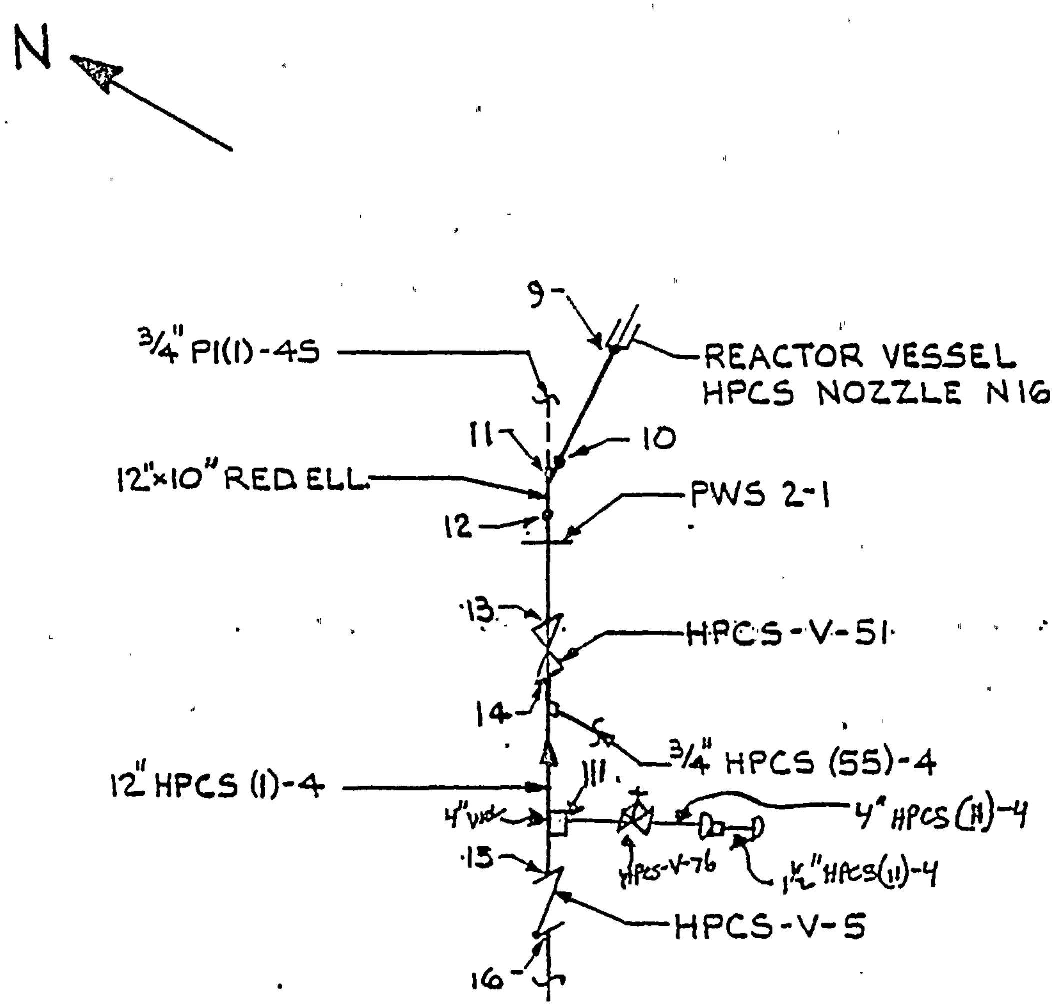
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LOW PRESSURE CORE SPRAY ISOMETRIC	FIGURE 3.6-27a

No Change

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 1	Node 3
Node 2	
Node 4	
Node 5	
Node 6	
Node 7	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LOW PRESSURE CORE SPRAY	FIGURE 3.6-27b

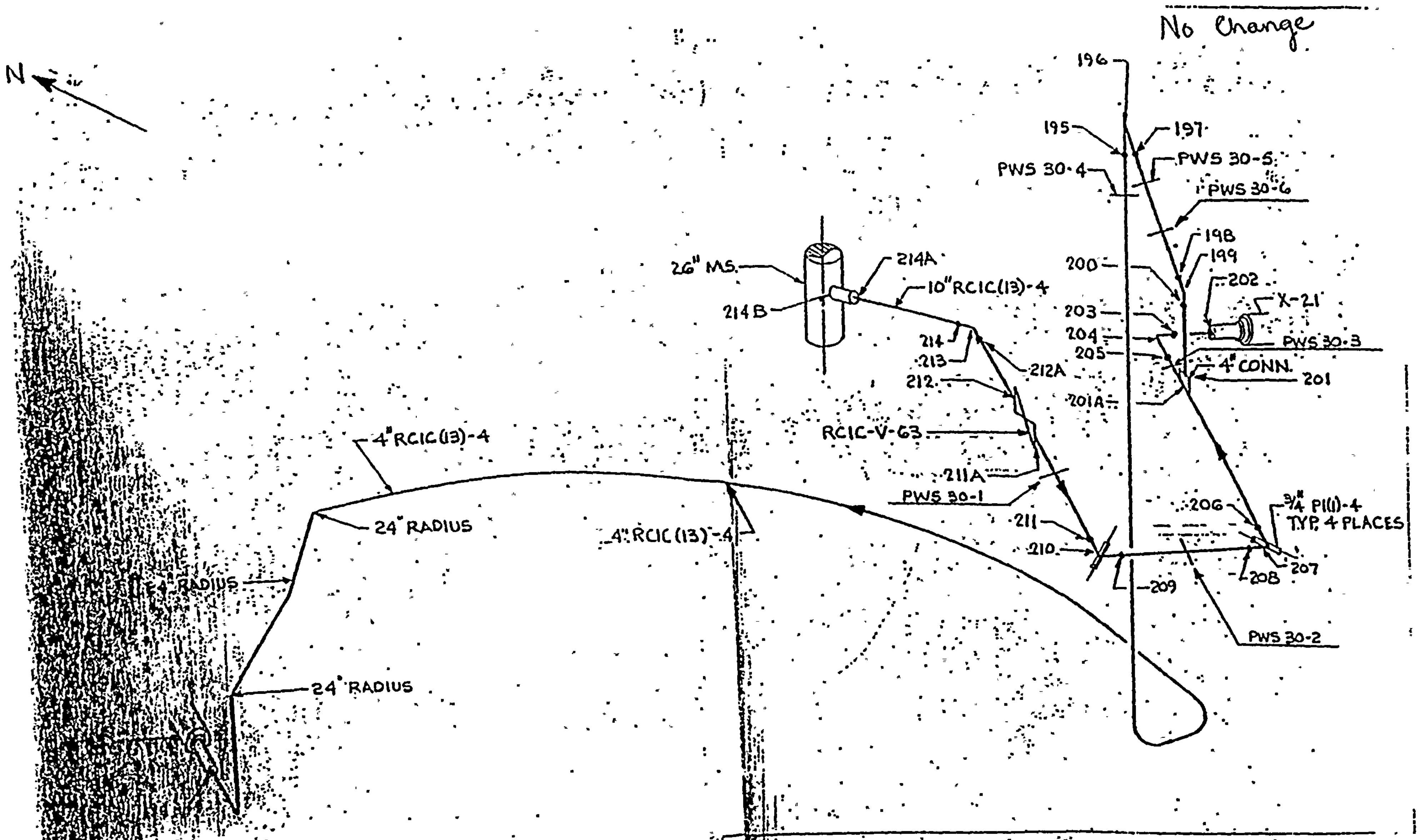


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY ISOMETRIC	FIGURE 3.6-28a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 9	Node 11
Node 10	
Node 12	
Node 13	
Node 14	
Node 15	
NODE 111	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY	FIGURE 3.6-28b



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR CONDENSING MODE RCIC TURBINE STEAM ISOMETRIC	FIGURE 3.6-29a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

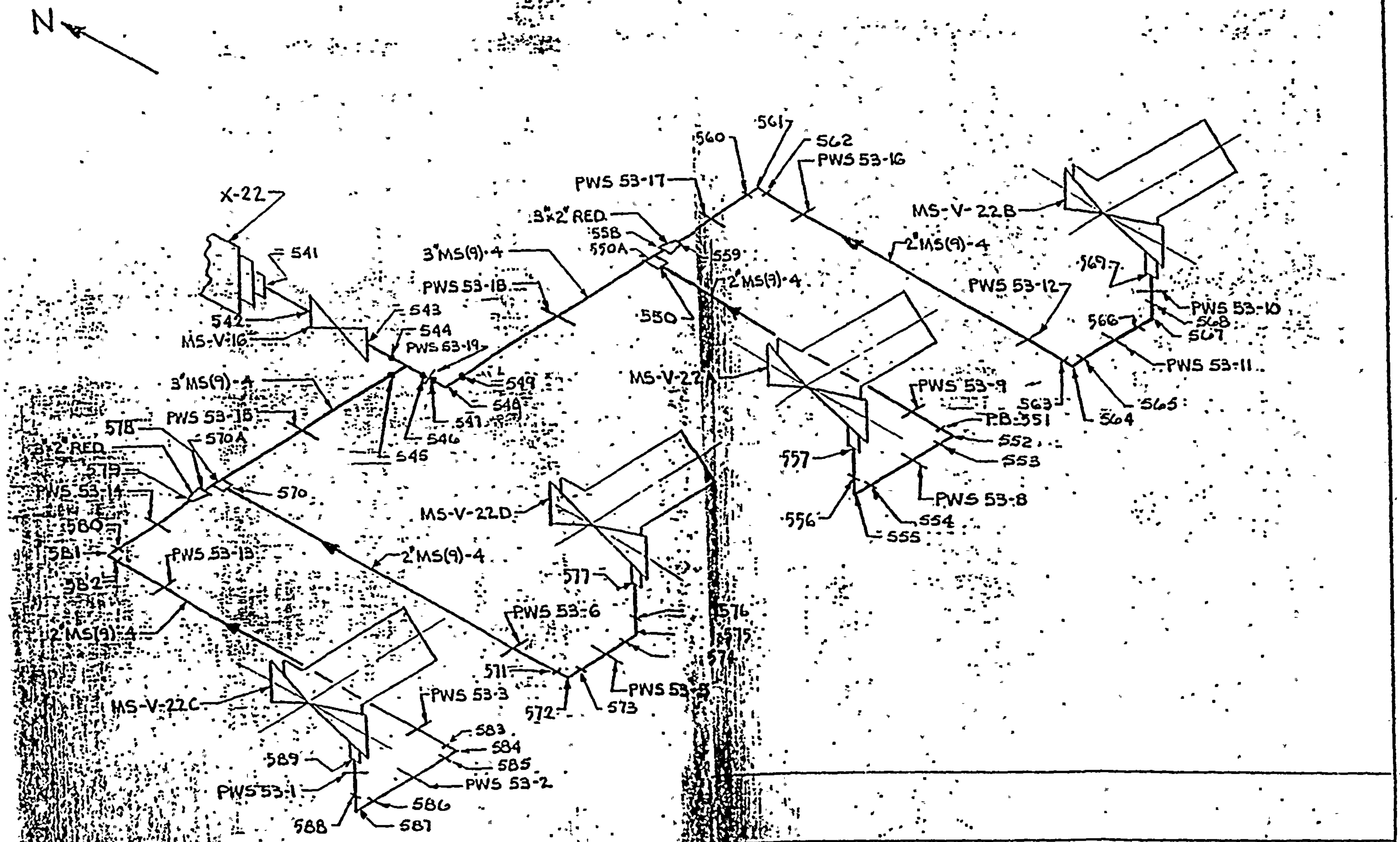
~~Node 194~~
~~Node 195~~
~~Node 197~~
~~Node 198~~
~~Node 200~~
~~Node 201~~
~~Node 202~~
~~Node 203~~
~~Node 205~~
~~Node 206~~
~~Node 208~~
~~Node 209~~
~~Node 211~~
~~Node 211A~~
Node 212
Node 212A
Node 214
~~[illegible]~~ A
~~[illegible] 214B~~
~~[illegible] 214A~~
NODE 214B

LONGITUDINAL BREAKS

~~Node 196~~
~~Node 199~~
~~Node 201A~~
~~Node 204~~
~~Node 207~~
~~Node 210~~
Node 213

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR CONDENSING MODE RCIC TURBINE STEAM	FIGURE 3.6-29b

No Change



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

MAIN STEAM VALVES DRAINAGE PIPING ISOMETRIC

FIGURE 3.6-30a

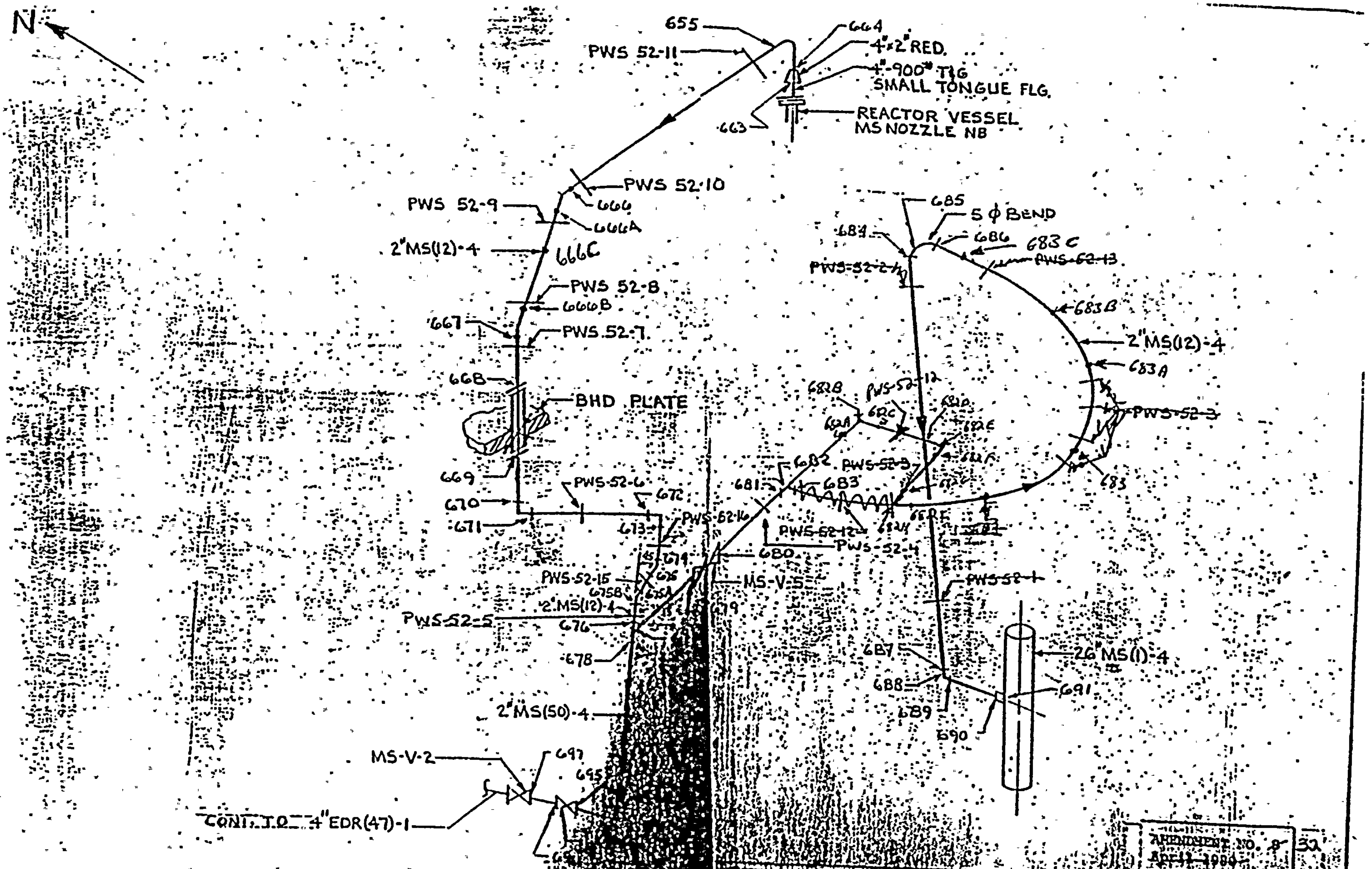
SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

~~Node 541~~
~~Node 542~~
Node 543
~~Node 544~~
~~Node 545~~
~~Node 546~~
~~Node 547~~
~~Node 549~~
Node 550
~~Node 551~~
~~Node 553~~
Node 554
Node 556

Node 557
~~Node 558~~
~~Node 559~~
Node 560
Node 562
~~Node 563~~ NODE 563
Node 565
Node 566
Node 568
Node 569
Node 570
~~Node 571~~
~~Node 573~~

Node 574
Node 576
Node 577
~~Node 578~~
~~Node 579~~
Node 580
Node 582
Node 583
Node 585
Node 586
Node 588
Node 589



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April 1980

MAIN STEAM RPV HEAD VENT ISOMETRIC

FIGURE 3.6-31a

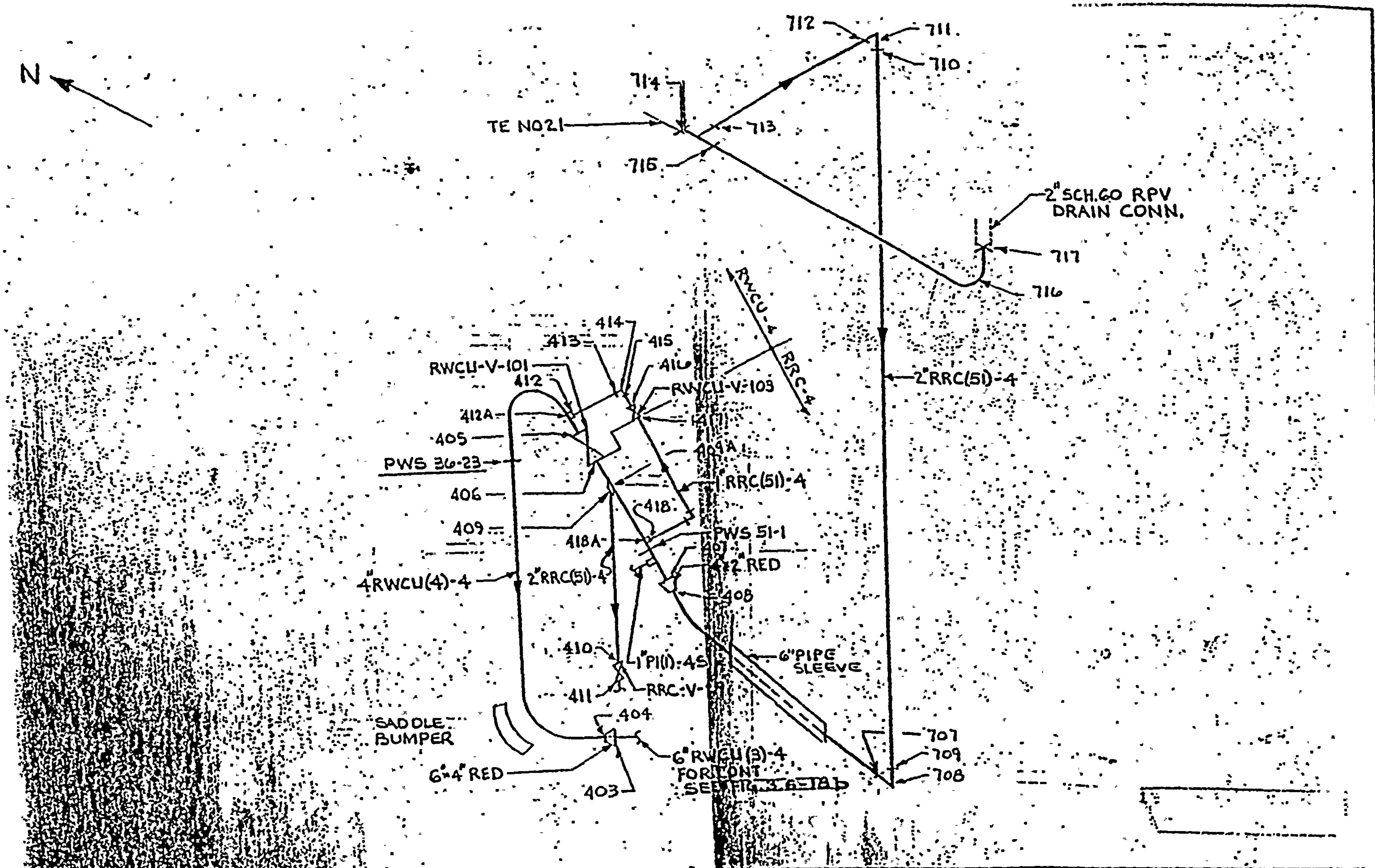
SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 663•
~~Node 664~~
~~Node 666~~
~~Node 666A~~
~~Node 666B~~
~~Node 667~~
Node 668•
~~Node 669~~
~~Node 670~~
~~Node 671~~
~~Node 672~~
~~Node 673~~
~~Node 674~~
~~Node 675~~
Node 675A•
Node 675B•
NODE 666C

Node 676•
Node 677•
Node 678•
Node 679•
~~Node 680~~
~~Node 681~~
Node 683•
~~Node 687~~
~~Node 689~~
Node 690•
~~Node 692~~
~~Node 694~~
Node 695•
~~Node 696~~
~~Node 697~~
~~Node 698~~
NODE 682A
NODE 682C
NODE 682D
NODE 682F
NODE 682G
NODE 682I
~~NODE [illegible]~~
NODE 683A
NODE 683B
NODE 683C

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM RPV HEAD VENT	FIGURE 3.6-31b



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RRC REACTOR PRESSURE VESSEL DRAIN ISOMETRIC	FIGURE 3.6-32a

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September 1983

SUMMARY OF PIPE BREAK LOCATIONS.

CIRCUMFERENTIAL BREAK.

NODE 717
NODE 715
NODE 713
NODE 410
NODE 409
NODE 418
NODE 416

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RRC REACTOR PRESSURE VESSEL DRAIN	FIGURE 3.6-32b

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No Change

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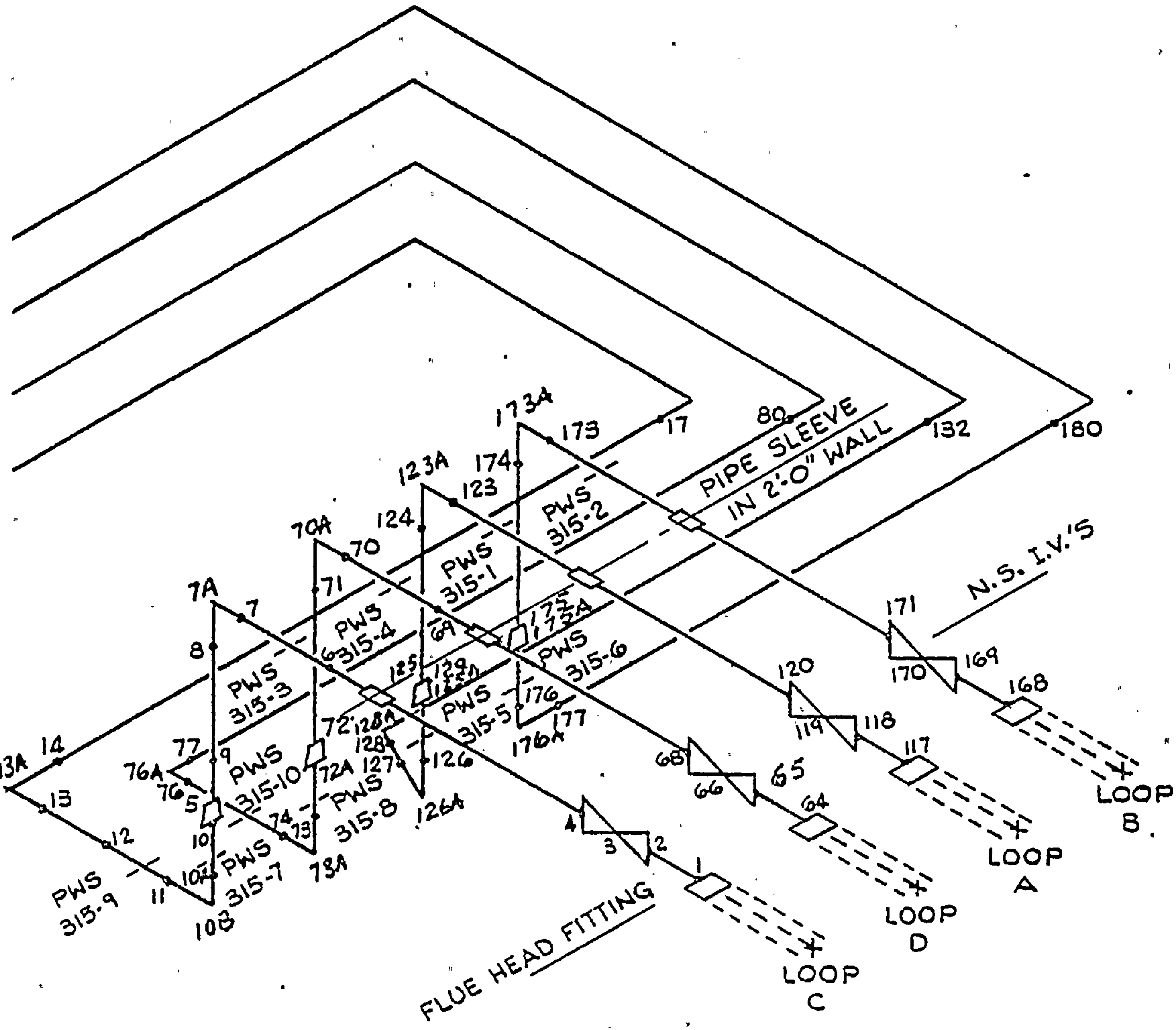
April 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM PIPING (LOOP A,B,C & D) INSIDE MAIN STEAM TUNNEL	FIGURE 3.6-33a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

~~Node 1~~	~~Node 80~~
~~Node 2~~	~~Node 117~~
Node 4	~~Node 118~~
~~Node 5~~	Node 120
Node 7	Node 123
Node 8	Node 124
~~Node 10~~	~~Node 125~~
~~Node 12A~~	~~Node 125A~~
~~Node 12~~	~~Node 126~~
~~Node 13~~	~~Node 127~~
~~Node 14~~	~~Node 128~~
~~Node 17~~	~~Node 129~~
~~Node 64~~	~~Node 132~~
~~Node 65~~	~~Node 168~~
Node 68	~~Node 169~~
Node 70	Node 171
Node 71	Node 173
~~Node 72~~	Node 174
~~Node 72A~~	~~Node 175~~
~~Node 73~~	~~Node 175A~~
~~Node 74~~	~~Node 176~~
~~Node 76~~	~~Node 177~~
~~Node 77~~	~~Node 180~~

LONGITUDINAL BREAKS

Node 7A
~~Node 10B~~
~~Node 13A~~
Node 70A
~~Node 73A~~
~~Node 76A~~
Node 123A
~~Node 126A~~
~~Node 128A~~
Node 173A
~~Node 176A~~

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM LOOP A, B, C, & D INSIDE MAIN STEAM TUNNEL	FIGURE 3.6-33b