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SUBJECT: Forwards responses ro Round One questions,Set 7 from Mechanical Engineering Branch.

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Docket No. 50-397

March 26, 1980
G02-80-81

Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Attention: Mr. L. Rubenstein, Chief
Branch No. 4
Division of Project Management

Subject: WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO ROUND ONE QUESTIONS
SET 7 - MECHANICAL ENGINEERING BRANCH (MEB)

Dear Mr. Rubenstein:

Attached please find sixty (60) copies of the responses to Round One, Set Seven (Mechanical Engineering Branch). These responses are to be incorporated formally into the FSAR in the next amendment.

The responses have been delayed due to efforts related to our work on Three Mile Island Lessons Learned and the realization that your review of our docket had been temporarily suspended due to similar reasons.

Very truly yours,

D L Renberger
D. L. RENBERGER
Assistant Director-Technology

DLR:CDT:ct
Attachment
cc: JJ Verderber, B&R, w/o attachment
RC Root, B&R, w/o attachment
RE Snaith, B&R, w/o attachment
J Ellwanger, B&R, w/attachment
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WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO ROUND ONE QUESTIONS
SET 7 - MECHANICAL ENGINEERING BRANCH (MEB)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Technology, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that he is authorized to submit the foregoing on behalf of said applicant; that he has read the foregoing and knows the contents thereof; and believes the same to be true to the best of his knowledge.

DATED March 25, 1980

D L Renberger
D. L. RENBERGER

On this day personally appeared before me D. L. RENBERGER to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 25th day of March, 1980

Reba B. Helgeson
Notary Public in and for the State of Washington
Residing at Richland

Q. 010.011
(3.5)

We require you to provide an evaluation of the environmental effects resulting from a postulated failure of the main steam lines and the main feedwater lines. Your evaluation should demonstrate conformance with our requirements that:

a. Those compartments and tunnels which house the main steam lines, the feedwater lines, including the isolation valves for these lines, are designed to withstand the environmental effects (pressure, temperature and humidity) and the potential flooding resulting from a postulated crack equivalent to the flow area of a single-ended pipe rupture in these lines.

b. The essential equipment located within these compartments, including the main steam line isolation valves and the feedwater valves and their associated valve operators, are capable of operating in the environment resulting from the crack postulated in Item (a) above.

c. If the forces resulting from this postulated crack could cause the structural failure of these compartments, the consequent failure of these compartments will not jeopardize the safe shutdown of the plant.

d. The remaining portion of the pipe in the tunnel between the outboard safety valve and the Turbine Building meet the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", with respect to the stress levels in this portion of the pipe and with respect to the location of the postulated break points.

We further require that you submit an analysis of the subcompartment pressure buildup following a postulated pipe break, including the structural evaluation of the affected subcompartments, to demonstrate that the design of the pipe tunnel conforms with our positions as stated above. If you cannot demonstrate conformance with our positions in this matter, indicate any design changes which may be required to comply with our positions. This evaluation should demonstrate

that the methods used to calculate the pressure transient in the subcompartments outside of the primary containment are the same as those used for subcompartments inside the containment for postulated pipe break. Demonstrate that the margin against a structural failure resulting from the pressure transient, are the same as those in subcompartments inside the primary containment. If you propose to use methods of analysis for subcompartments outside of containment which are different from those used inside containment, demonstrate that the methods of analysis for subcompartments outside containment assure adequate design margins. Identify the computer codes and the assumptions regarding the mass and energy release rates which you used in your analysis. Provide sufficient design data so that we may perform independent calculations.

Response:

The compartments and tunnels which house the main steam lines and the reactor feedwater lines, including the isolation valves for these lines between the primary containment vessel and the turbine generator building, are the main steam tunnel in the reactor building and the main steam tunnel extension in the turbine generator building. Overpressurization of the main steam tunnel and tunnel extension due to a postulated pipe break is prevented by venting the main steam tunnel and tunnel extension to the turbine building, by way of the tunnel extension, and to the atmosphere, by way of the ventway structure. The following sections, tables and figures address, either totally or in part, the main steam tunnel, ventway and tunnel extension:

a. 3.6.1.18.3.1, 3.6.1.18.3.2, 3.6.1.20,* 3.8.4.1.1.4, 3.8.4.1.3, 3.8.4.3.3f, 3.8.4.4.1

b. Tables 3.6-11 through 3.6-17

c. Figures 1.2-5, 1.2-6, 3.6-6g through 3.6-6k, 3.6-38, 3.6-39, 3.6-40a, 3.6-40b, 3.6-44, 3.6-49, 3.6-123 through 3.6-146, 3.8-2, 3.8-30 through 3.8-33, 3.8-38, 3.8-39, 3.8-54, and 3.8-55.

An evaluation of the environmental effects resulting from a postulated pipe break in the main steam line or the reactor feedwater line demonstrates conformance with NRC

*Draft FSAR page changes attached.

requirements, as described in the following paragraphs a, b, c and d, which paragraphs correspond to paragraphs of the same designation in the question:

a. The main steam tunnel, including the blast door, removable concrete plugs, ventway and tunnel extension, are designed to withstand the environmental effects of the predicted pressure, temperature and humidity ~~(given in these tables)~~ and the potential flooding effects (described in paragraph b below) resulting from a postulated crack equivalent to the flow area of a single-ended pipe break in the main steam line or the reactor feedwater line.

b. The essential equipment, including the main steam isolation valves and the feedwater valves and their associated valve operators, are, or will be, shown capable of operating in the environment resulting from the crack postulated in (a) above. WPPSS has an extensive environmental qualification program ongoing to verify required equipment operability. Equipment listed in Tables 10.11-1 and 10.11-2 will be appropriately listed in Tables in 3.11 with qualification information to the listed criteria.

Table Q. 010.11-1 and Table Q. 010.11-2 list the essential equipment in the main steam tunnel and tunnel extension. There is no essential equipment in the ventway. The tables compare the design environmental conditions to the maximum predicted environmental conditions (pressure, temperature, humidity) resulting from the crack postulated in (a) above.

The main steam tunnel and tunnel extension face no potential flooding problem resulting from the postulated crack in the main steam line, because the closure of the main steam isolation valve terminates flow from the reactor side of the main steam line within a maximum closure time of 5.5 seconds, as discussed in the response to Question 010.13.

With regard to flooding resulting from a postulated reactor feedwater line crack occurring in the tunnel extension, water will flow directly upon the mezzanine floor at elevation 471'-0" in

the turbine building by way of the opening in the tunnel extension at elevation 501'-0". However, the water is eventually removed by the turbine building drainage system. If the postulated reactor feedwater line crack occurs in the main steam tunnel rather than in the tunnel extension, the water will flow through the openings created by the displacement of the blow-out panels at the north end of the main steam tunnel and at the north end of the main steam tunnel east wall. The blow-out panels are designed to blow off at a differential pressure of 0.5 psi, as noted in 3.6.1.20.3.2. If the blow-out panels were to fail to open, in the case of a smaller break, the blow-out panels will open under the water pressure developed by the water accumulated on the tunnel floor, before the flood level reaches any safety-related equipment. The water will accumulate to a height of 15-inches before failure of the blow-out panels and consequent release of the headwater occurs.

c. The forces resulting from the postulated crack do not cause structural failure of the tunnel, tunnel extension, or the ventway.

d. The portion of the pipe in the main steam tunnel between the outboard safety valve and the turbine building meets the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", with respect to the stress levels in this portion of the pipe and with respect to the location of the postulated pipe break points.

The method used to calculate blowdown is discussed in 3.6.1.20.1.3.* The mechanism which terminates the blowdown is discussed in the response to Question 010.03.

An analysis of the subcompartment pressure buildup in the main steam tunnel, ventway and tunnel extension following a postulated main steam line crack (equivalent to the flow area of a single-ended pipe rupture) in the main steam tunnel or the tunnel extension, and verification of the structural adequacy of the tunnel, ventway and tunnel extension are discussed in 3.6.1.20.3.* The subcompartment analysis following the postulated crack in the reactor feedwater line (equivalent to the flow area of a single-ended pipe rupture) is not discussed because, in

*Draft FSAR page changes attached.

comparing the postulated crack analyses of both the reactor feedwater and the main steam lines, the postulated main steam line crack is the limiting case as stated in 3.6.1.20.3.*

The methods used to calculate the pressure transient in the main steam tunnel, tunnel extension and ventway are the same as those used for subcompartment pressure analyses inside the primary containment vessel for a postulated pipe break. The structural design of the tunnel, tunnel extension and ventway has the same margin against structural failure resulting from the pressure transient as the structural design of the subcompartments inside the primary containment vessel.

The computer codes used in the analysis are identified in 3.6.1.20 and in 3.12.* The assumptions regarding the mass and energy release rates used in the analysis are identified in 3.6.1.20.*

*Draft FSAR page changes attached.

TABLE 010.11-1

COMPARISON OF DESIGN AND PREDICTED ENVIRONMENTAL CONDITIONS FOR ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(1)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(4)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL: DESIGN			MAXIMUM PREDICTED, FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL(2)				
NAME	DESIGNATION	LOCATION IN TUNNEL	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	B&R REFERENCE DRAWING	REMARKS
Main Steam Body Drain Shutoff Valves	MS-V-67A	South	340 to 212	45	100	307	12	100	M695,M697	
	MS-V-67B	South	340 to 212	45	100	307	12	100	M695,M697	
	MS-V-67C	South	340 to 212	45	100	307	12	100	M695,M697	
	MS-V-67D	South	340 to 212	45	100	307	12	100	M695,M697	
Main Steam Drain Block	MS-V-19	South	340	-2 to +45	100	307	12	100	M695,M697	
Main Steam Isolation Valves-Outboard	MS-V-28A	South	340	-2 to +45	100	307	12	100	M695,M697	
	MS-V-28B	South	340	-2 to +45	100	307	12	100	M695,M697	
	MS-V-28C	South	340	-2 to +45	100	307	12	100	M695,M697	
	MS-V-28D	South	340	-2 to +45	100	307	12	100	M695,M697	
Main Steam Leakage Control Valves	MSLC-V-2A	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-2B	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-2C	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-2D	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-3A	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-3B	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-3C	South	340	-2 to +45	100	307	12	100	M698	
	MSLC-V-3D	South	340	-2 to +45	100	307	12	100	M698	
Main Steam Leakage Control Valves	MSLC-V-4	North	340	-2 to +45	100	305	12	100	M698	
	MSLC-V-5	North	340	-2 to +45	100	305	12	100	M698	
	MSLC-V-9	North	340	-2 to +45	100	305	12	100	M698	
	MSLC-V-10	North	340	-2 to +45	100	305	12	100	M698	

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(4)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL: DESIGN			MAXIMUM PREDICTED, FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL(2)				
NAME	DESIGNATION	LOCATION IN TUNNEL	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	B&R REFERENCE DRAWING	REMARKS
Reactor Feed Water Valves	RFW-V-32A(3)	South	340 to 212	45	100	307	12	100	M713	
	RFW-V-32B(3)	South	340 to 212	45	100	307	12	100	M713	
	RFW-V-65A	South	340 to 212	45	100	307	12	100	M713	
	RFW-V-65B	South	340 to 212	45	100	307	12	100	M713	
General Electric Temperature Elements	E31- NO 29A	North	350	LATER	LATER	305	12	100	E697	
	E31- NO 29B	North	350			305	12	100	E697	
	E31- NO 29C	North	350			305	12	100	E697	
	E31- NO 29D	North	350			305	12	100	E697	
General Electric Temperature Elements	E31- NO 30A	South	350			307	12	100	E697	
	E31- NO 30B	South	350			307	12	100	E697	
	E31- NO 30C	South	350			307	12	100	E697	
	E31- NO 30D	South	350			307	12	100	E697	
General Electric Temperature Elements	E31- NO 31A	South	350			307	12	100	E697	
	E31- NO 31B	South	350			307	12	100	E697	
	E31- NO 31C	South	350			307	12	100	E697	
	E31- NO 31D	South	350			307	12	100	E697	
Cables, Electrical, and Instrumentation and Control(5)	None	Various	340	-2 to +45	See Footnote 5	307	12	100	E683,E697	0 to 3 hrs.(6)
	None	Various	320	-2 to +45		307	12	100	E683,E697	4 to 9 hrs.(6)
	None	Various	200	0 to 25		307	12	100	E683,E697	10 to 33 hrs.(6)
	None	Various	194	0.5 to 2.0		307	12	100	E683,E697	Remainder(6)

TABLE 010.11-1 (Continued)

NOTES FOR TABLE

(1) The main steam tunnel is in the reactor building. The tunnel extension, Table Q. 010.11-2, is in the turbine-generator building.

(2) The maximum predicted conditions correspond to a time duration of 0 to 2 hours. From 2 hours to 6 hours, the predicted temperature is 212°F and the predicted pressure approaches atmospheric.

(3) Valves RFW-V-32A and 32B are swing check valves. ~~As such, their safety-related function, namely,~~ Their operability is not impaired by the maximum predicted pressure.

(4) There is no tubing in the main steam tunnel and tunnel extension for air lines operating instrumentation and control equipment and components or for any other applications.

(5) Electrical power cables and instrumentation and control cables were given qualification tests for nuclear power services, in accordance with IEEE Standard No. 323-1974. The tests included steam environment exposure under simulated normal operating conditions and LOCA conditions. Associated test documents are:

a. For power cables: The Okonite Company, Ramsey, New Jersey, Engineering Report No. 266, dated July 17, 1975 submitted to Burns and Roe as Transmittal 62A-00-0004, in accordance with Contract Specification 2800-62A.

b. For power cables and instrumentation and control cables: The Raychem Corp., Menlo Park, California, Report No. RABR-62B-75-028, dated November 12, 1975, submitted to Burns and Roe as Transmittal 62B-00-0094, in accordance with Contract Specification 2800-62B.

(6) LOCA ratings

TABLE 010.11-2

COMPARISON OF DESIGN AND PREDICTED ENVIRONMENTAL CONDITIONS FOR ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL EXTENSION(1)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(3)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL EXTENSION							
			DESIGN			MAXIMUM PREDICTED,(2) FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL EXTENSION				
NAME	DESIGNATION	LOCATION IN TUNNEL EXTENSION	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	B&R REFERENCE DRAWING	REMARKS
General Electric Radiation Detectors	D17-N003A	North	LATER	LATER	LATER	313	8	100	E607	
	D17-N003B	North				313	8	100	E607	
	D17-N003C	North				313	8	100	E607	
	D17-N003D	North				313	8	100	E607	
Cables, Electrical and Instrumentation and Control	Data same as in Table Q.010.11-1 and associated footnotes 4 and 5								E590,607	

(1) The main steam tunnel extension is in the turbine-generator building. The main steam tunnel, Table Q. 010.11-1, is in the reactor building.

(2) The maximum predicted conditions correspond to a time duration of 0 to 2 hours. From 2 hours to 6 hours, the predicted temperature is 212°F and the predicted pressure approaches atmospheric.

(3) There is no tubing in the main steam tunnel and tunnel extension for air lines operating instrumentation and control equipment and components or for any other application.

3.6.1.18.3.6 Postulated Ruptures of the Auxiliary Steam, Heating Steam, Auxiliary Condensate and Heating Steam Condensate Piping

The consequences of a rupture in any of the above, including the dynamic effects of pipe whip and the resulting environmental conditions, are investigated as described in 3.6.1.11. In no instance does a postulated rupture of these systems preclude reactor shutdown to a cold condition.

These systems provide no emergency function which would be required to mitigate the consequences of a postulated piping failure. Therefore, normal reactor shutdown as well as the emergency methods described would not be simultaneously impaired.

3.6.1.18.3.7 Postulated Rupture of the Reactor Water Cleanup System Piping

The consequences of a reactor water cleanup system piping rupture are investigated as discussed in 3.6.1.11. In no circumstance, does the postulated failure of reactor water cleanup system piping preclude the availability of all shutdown modes. Since the reactor water cleanup system does not fulfill any safety function, nonoperability has no impact on the safe shutdown of the reactor.

3.6.1.19 Seismic and Quality Classifications of Piping Used in the Synamic Analysis

Table 3.6-7 gives the seismic and quality classifications of all piping listed in Table 3.6-6. Refer to 3.2 for descriptions of the various seismic and quality classifications.

3.6.1.20 Method Used to Predict Blowdown Rates and Subcompartment Pressure Transient After a Postulated Pipe Break

3.6.1.20.1 Blowdown Analysis for a Postulated Pipe Break Outside the Primary Containment

The analytical approach used to determine the blowdown mass and energy rates from a postulated pipe break outside the primary containment are described in 3.6.1.20.1.1 through 3.6.1.20.1.3.

3.6.1.20.1.1 Method of Analysis for a Postulated Pipe Break in Larger Pipes

For larger pipes, the blowdown mass and energy release are predicted by using the computer programs RELAP3 (reference 3.6-9) or RELAP4/MOD5 (reference 3.6-21). In the computer model, the piping system is nodalized into a number of volumes connected by flow junctions. A multiplier of 1.0 is used with the choked flow correlation. Except in special cases, all breaks are assumed to be the double-ended circumferential type which open instantaneously. Initial conditions and other assumptions necessary for the analysis are such that the result is on the conservative side.

3.6.1.20.1.2 Method of Analysis for a Postulated Pipe Break in Smaller Pipes

For smaller pipes, a constant blowdown profile with an applicable choked flow correlation is used. The initial conditions are chosen to maximize the blowdown mass and energy release rates.

3.6.1.20.1.3 Blowdown Mass and Energy Release Rates for a Postulated Pipe Break in the Main Steam Line and the Reactor Feedwater Line in the Main Steam Tunnel

Refer to 3.6.1.20.3.2 for a description of the arrangement and features of the main steam tunnel. For subcompartment analysis in the main steam tunnel, the postulated break in the main steam line and in the feedwater line is assumed to be a crack with the flow area equivalent to the flow area of a single-ended pipe. The blowdown mass and energy release rates are computed by the RELAP3 Program.

Figures 3.6-123 and 3.6-124 show the mass and energy release rates after a postulated crack in the main steam line in the main steam tunnel. Figures 3.6-125 and 3.6-126 show the mass and energy release rates after a postulated crack in the reactor feedwater line in the main steam tunnel.

3.6.1.20.2 Subcompartment Analysis for Postulated Pipe Break Outside the Primary Containment Excluding the Main Steam Tunnel, Ventway and Tunnel Extension

3.6.1.20.2.1 Method of Analysis

The pressure transient in the reactor building after a postulated pipe break is analyzed with the computer programs RELAP3 (reference 3.6-9) or RELAP4/MOD5 (reference 3.6-21). In the computer model, subcompartments are represented by nodes, and flow paths between two nodes are represented by flow junctions. Volumes, vent areas, flow resistances, initial atmospheric conditions, as well as the blowdown mass and energy release rates from the pipe breaks, are input to the computer program. Since the absolute pressure within the subcompartments after a pipe break outside the primary containment ~~are~~ low in all cases, no significant pressure gradient exists within a subcompartment itself. Therefore, a subcompartment is not nodalized in the analysis and sensitivity study is not performed.

3.6.1.20.2.2 Initial Atmospheric Conditions

The initial atmospheric conditions within the subcompartments used for the analysis are:

a. Pressure = 14.7 psia

b. Temperature = 110°F

c. Relative Humidity = 0.0%

These conditions are simulated in the computer analysis as a homogeneous saturated steam - water mixture at 14.7 psia with an average density equivalent to the density of air at the above conditions.

3.6.1.20.2.3 Vent Flow

The vent flow between the subcompartments is assumed to be a homogeneous steam - water mixture with 100% water entrainment. For choked flow, a multiplier of 0.6 is used for Moody two-phase flow correlations. For unchoked flow, the flow resistance consists of an entrance loss, an exit loss, and frictional losses. For conservatism, an entrance loss of 0.5 and an exit loss of 1.0 are assumed for most of the vents.

3.6.1.20.2.4 Results of Subcompartment Analyses

Subcompartment analyses are performed for all subcompartments containing high energy piping. The results are summarized in Table 3.6-12.

3.6.1.20.2.5 Verificaton of Structural Adequacy

Verification of structural adequacy of compartments, or of structural elements thereof, subjected to pressure generated by a postulated pipe break and to the local effects in the

structure generated by the postulated pipe break, namely, a broken pipe reaction, jet impingement and pipe whipping impact, is furnished in 3.6.1.6 through 3.6.1.10.

3.6.1.20.3 Subcompartment Analysis for a Postulated Pipe Break in the Main Steam Tunnel

Refer to 3.6.1.20.3.2 for a description of the arrangement and features of the main steam tunnel. Subcompartment analysis in the main steam tunnel, ventway and tunnel extension is performed for a postulated crack in the main steam line (refer to 3.6.1.20.1.3). Comparison of mass and energy release rates for the postulated crack in the main steam line to that of a postulated crack in the reactor feedwater line (Figures 3.6-123 through 3.6-126) shows that the main steam line crack is the limiting case. Other lines inside the main steam tunnel or its extension are of smaller sizes and a break in those lines is less severe.

3.6.1.20.3.1 General Approaches

The pressure and temperature transients in the main steam tunnel, ventway and tunnel extension after a postulated crack in the main steam line are computed by the RELAP4/MOD5 program (reference 3.6-21). The general approaches discussed in 3.6.1.20.2.1 through 3.6.1.20.2.3 also apply to this case.

3.6.1.20.3.2 Description of the Main Steam Tunnel, Ventway and Tunnel Extension

Descriptive information of the main steam tunnel, ventway and tunnel extension is provided in 3.8.4.1.1.4. Figures 3.6-127 and 3.6-128 show a sectional plan view and a sectional elevation view, respectively, of the main steam tunnel, ventway and tunnel extension.

In plan, the main steam tunnel is located at 0° azimuth of the north side of the reactor building; in elevation, it extends from elevation 501'-0" to elevation 522'-0". At the interface of the reactor building and the turbine generator building, the main steam tunnel continues for a short distance into the turbine generator building; the portion in the turbine generator building is referred to as the tunnel extension. ~~In elevation 522'-0", the tunnel extension extends from elevation 501'-0" to elevation, as does the main steam tunnel in the reactor building.~~ The ventway starts at the same level as the main steam tunnel and extends horizontally from the main steam tunnel in the easterly direction; and continues upward to the underside of the corridor floor above at elevation 548'-0", where a blow-out panel in the north wall of the ventway provides a ventilating path to the atmosphere.

Four blow-out panels are used, as shown in Figures 3.6-127 and 3.6-128:

a. Panel A, vertical, part of secondary containment, located between the north end of the main steam tunnel (in the reactor building) and the tunnel extension (in the turbine generator building), bolted in place, of sheet steel.

b. Panel B, vertical, part of secondary containment, located in the east wall of the main steam tunnel, bolted in place, of sheet steel.

c. Panel C, horizontal, part of secondary containment, located at the top of the main steam tunnel, the north edge of panel is hinged, other edges are free and not bolted in place, of sheet steel.

d. Panel D, vertical, not part of secondary containment, located in the north exterior wall of the ventway, bolted in place, of insulated metal siding.

The fasteners of the blow-out panels which are bolted in place (namely, panels A, B and D) are designed to fail in single shear, and all blow-off panels (namely, panels A, B, C and D) are designed to blow-off and permit venting when the pressure generated in the main steam tunnel, ventway and tunnel extension by a postulated pipe break within the main steam tunnel or tunnel extension exceeds 1/2 psi.

3.6.1.20.3.3 Analysis for a Postulated Pipe Break in the Main Steam Tunnel

Section 3.6.1.20.3.4 discusses the analysis for a postulated pipe break in the tunnel extension.

Figure 3.6-129 shows the nodalization scheme for a postulated pipe break in the main steam tunnel. For conservatism blow-out panels A and B are assumed to remain in place during the pressure transient. Therefore, the tunnel extension and the turbine generator building are not modeled in this case. Nodes 1 and 2 represent the main steam tunnel. Nodes 3, 4, 5 and 6 represent the ventway. Tables 3.6-13 and 3.6-14 provide the volume and flow junction data, respectively.

The hinged panel C is modeled as an inertia valve in the RELAP4/MOD5 analysis (refer to reference 3.6-21). The differential equation of motion for the valve gate is:

$$I\ \ddot{\theta}\ (t) = \bar{A}\cdot P\ (t) - K\ \dot{\theta}\ (t) \qquad \text{(Eq. 3.6.1.20.3.3-1)}$$

where:

θ = opening angle (radians)

t = time (sec.)

$\dot{\theta} = \frac{d\theta}{dt}$

$\ddot{\theta} = \frac{d^2\theta}{dt^2}$

I = moment of inertia ($lbs._{mass} - ft^2$)

$\bar{A}$ = area X moment arm ($ft.^3$)

P = differential pressure (lbs/ft^2)

K = damping constant $\left(\frac{lbs._{mass}-ft^2}{sec.}\right)$

Let ω = angular velocity in radians/sec. and substituting $\omega = \dot{\theta}$ and $\dot{\omega} = \ddot{\theta}$ in Eq. 3.6.1.20.3.3-1.

$$I\dot{\omega} + K\omega = \bar{A}P$$

It has the solution:

$$\theta = \theta_o + \omega_o\ t + \left(\frac{AP}{K} - \omega_o\right)\left[t - \frac{I}{K}\cdot\left(1 - e^{\frac{-Kt}{I}}\right)\right] \qquad \text{(Eq. 3.6.1.20.3.3-2)}$$

Where θ_o and ω_o are values for θ and ω at $t = 0$.

For panel D, the differential equation of motion is:

$$M\ddot{s}\ (t) + AP\ (t) \qquad \text{(Eq. 3.6.1.20.3.3-3)}$$

where:

M = mass of panel $\left(\frac{lbs._{mass} - sec.^2}{ft}\right)$

s = displacement (ft.)

v = velocity (ft/sec)

$\ddot{s}$ = linear acceleration (ft/sec^2)

A = area of panel (ft^2)

P = average pressure (lbs/ft^2)

t = time (sec)

The frictional force is neglected. The solution of the above equation is:

$$s = s_o + (v_o + \frac{F}{2m} t) t \qquad \text{(Eq. 3.6.1.20.3.3-4)}$$

where s_o and v_o are values for s and v at t = O.

The displacement, s, of the panel and the opening area as functions of time are determined by iterative procedures.

Pertinent properties of blow-out panels C and D are furnished in Table 3.6-15.

Figures 3.6-130 and 3.6-131 are plots of the pressure transients and Figures 3.6-132 and 3.6-133 are plots of the temperature transients for a postulated pipe break in Node 1.

Figures 3.6-134 and 3.6-135 are plots of the pressure transients and Figures 3.6-136 and 3.6-137 are plots of the temperature transients for a postulated pipe break in Node 2.

Blow-out panels C and D are assumed to blow off at the differential pressure noted in 3.6.1.20.3.2.

3.6.20.3.4 Analysis for a Postulated Pipe Break in the Tunnel Extension

Figure 3.6-138 shows the nodalization scheme for a postulated pipe break in the tunnel extension. Nodes 1 and 2 represent the tunnel extension. The vertical pipe restraint (see Figures 3.6-6g, 3.6-6h, 3.6-6j and 3.6-6k) divides Node 1 and 2. Nodes 3, 4 and 5 represent the following portions of the turbine generator building:

a. Node 3 represents the portion between the mezzanine floor at elevation 471'-0" and the operating floor at elevation 501'-0".

b. Node 4 represents the portion between the ground floor at elevation 441'-0" and the mezzaine floor at elevation 471'-0"

c. Node 5 represents the portion between the operating floor elevation 501'0" and the roof of the turbine generator building

For conservatism, panel A (in Figures 3.6-127 and 3.6-128) is assumed to remain in place during the pressure transient, and only 10% of the insulated metal siding comprising the exterior walls above the operating floor (see Figure 1.2-8) of the turbine generator building is assumed to blow off the structural steel frame at a differential pressure of 1/2 psi. Tables 3.6-16 and 3.6-17 provide the volume and flow junction data, respectively.

Figures 3.6-139 and 3.6-140 are plots of the pressure transients and Figures 3.6-141 and 3.6-142 are plots of the temperature transients for a postulated pipe break in Node 1.

Figures 3.6-143 and 3.6-144 are plots of the pressure transients and Figures 3.6-145 and 3.6-146 are plots of the pressure transients for a postulated pipe break in Node 2.

3.6.1.20.3.5 Verification of Structural Adequacy

Verification of structural adequacy of the main steam tunnel ventway and tunnel extension, or of structural elements thereof, subject to load combinations involving pressure generated by a postulated pipe break and local effects in the structure generated by the postulated pipe break, namely, broken pipe reaction, jet impingement and pipe whip impact, is furnished in 3.6.1.6 through 3.6.1.10.

3.6.1.21 Description of Methods of Analyses to Ensure That Primary or Secondary Containment Integrity Is Not Compromised by a Postulated Passive Component Failure

The previous twenty sections present the results of analyses that indicate that postulated piping failures do not adversely affect safe reactor operation. Implicit in these analyses is the necessity of conforming to relevant standards with regard to offsite radiological consequences.

3.6-10 Newmark, N.M., and Richart, F.E., Impact Tests of Reinforced Concrete Beams, NDRC Report No. A-125, A-213 and A-304, 1941-1946.

3.6-11 AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings", American Institute of Steel Construction, New York, N. Y., February 12, 1969.

3.6-12 ACI 318-71, "Building Code Requirements for Reinforced Concrete", American Concrete Institute, Detroit, Michigan, 1971.

3.6-13 Linderman, R. B., Rotz, J. V., Yeh, G. C. K., Design of Structures for Missile Impact, Bechtel Power Corporation, Topical Report BC-TOP-9A, Revision 2, San Francisco, California, September 1974.

3.6-14 Roark, R. J., Formulas for Stress and Strain, Fourth Edition, McGraw-Hill Book Company, Inc., New York, 1965.

3.6-15 Amarikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, Report NP-3726, August 1950.

3.6-16 Moody, F. J., Maximum Flow Rate of a Single Component, Two-Phase Mixture, American Society of Mechanical Engineers Journal of Heat Transfer, pp. 134-142, February 1965.

3.6-17 Gwaltney, R. C., Missile Generation and Protection in Light-Water Cooled Power Reactor Plants, ORNL NSIC-22, Oak Ridge National Laboratory, Oak Ridge Tennessee, for the U. S. Atomic Energy Commission, September, 1968.

3.6-18 Deleted. Replaced by 3.6-20 below.

3.6-19 Harris and Crede, Shock and Vibration Handbook, McGraw Hill Book Company, Inc., 1961.

3.6-20 Kennedy, R. P., A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects, Nuclear System Sciences Group, Holmes & Narver Inc., Anaheim, California, September 1975.

3.6-21 Idaho National Engineering Laboratory, RELAP4/MOD5, A Computer Program For Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, Volume I: RELAP4/MOD5 Description, Volume II: Program Implementation, Volume III: Checkout Application, National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia,

TABLE 3.6-12

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)
422	R11/R106	HPCS Pump Room	4" AS (11)-2	0.09	R11, R106/R206	1.6	0.15
				0.09	R11, R106/R12, R114	1.6	0.15
				0.09	R11, R106/R10, R105	1.6	0.15
422	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	0.35	0.08
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
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.

TABLE 3.6-12 (Continued)

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)
522	R409	RWCU Pump Room	6" RWCU(1)-4	11.0	R409/R404, R504	0.7	16.5
				6.3	R409/R405	0.5	9.5
				5.6	R405/R404	0.8	8.4
				11.0	R406, R407/R404, R305	0.7	16.5
522	R406/R407*	RWCU Pump Room	4" RWCU(2)-4	15.0	R406/R404, R305	1.3	22.5
				11.0	R406/R407, R409	1.1	16.5
				4.2	R409/R504, R404	1.8	6.3
				1.7	R405/R305, R404, R504	1.5	2.6
				2.4	R409/R405	1.7	3.6
522	R408	Valve Room	6" RWCU(2)-4	1.0	R408/R404	0.2	1.5
				1.0	R408/R305	0.2	1.5
				1.0	R408/R509	0.2	1.5
548	R509	Valve Room	6" RWCU(2)-4	2.1	R509/R508, R408	0.44	3.2
				2.1	R509/R607	0.44	3.2
548	R510	Valve Room	6" RWCU(1)-4	1.8	R510/R504, R508	1.1	2.7
				1.8	R510/R404, R604	1.1	2.7
548	R511/R511A	Valve Room	6" RWCU(1)-4	4.4	R511/R404, R504	0.75	6.6
				4.4	R511/R604	0.75	6.6
572	R604	El. 572' Open Floor Area	4" HS (1)-2	0.045	R604/R504	0.35	0.05
				0.045	R604/R704	0.35	0.05

* Break could occur in either room; break assumed in R406

TABLE 3.6-13

SUBCOMPARTMENT ANALYSIS

NODAL VOLUME DATA FOR A POSTULATED PIPE BREAK IN THE MAIN STEAM TUNNEL[(a)]

NODE NUMBER	DESCRIPTION	VOLUME (CU.FT.)	ELEVATION (FT.)
1	MAIN STEAM TUNNEL, SOUTH	7427	501
2	MAIN STEAM TUNNEL, NORTH	4345	501
3	VENTWAY, EL. 501'-0" TO EL. 519'-0"	3629	501
4	VENTWAY, EL. 519'-0" TO EL. 532'-0", WEST	3672	519
5	VENTWAY, EL. 519'-0" TO EL. 532'-0", EAST	2340	519
6	VENTWAY, EL. 532'-0" TO EL. 548'-0"	7855	532

(a) For nodalization scheme, see Figure 3.6-129.

TABLE 3.6-14

SUBCOMPARTMENT ANALYSIS

FLOW JUNCTION DATA FOR A POSTULATED PIPE BREAK IN THE MAIN STEAM TUNNEL[a]

FROM NODE	TO NODE	JUNCTION FLOW AREA (SQ.FT.)	JUNCTION ELEVATION (FT.)	JUNCTION INERTIA ($FT.^{-1}$)	FORM LOSS COEFFICIENT[b] FORWARD FLOW	FORM LOSS COEFFICIENT[b] REVERSE FLOW	FRICTIONAL LOSS COEFFICIENT[b]
1	2	438.7	509	0.02656	1.06	1.14	0.1
2	3	SEE FOOTNOTE (c)					
2	4	218.4	519	0.06044	2.66	2.69	0.1
3	5	170.0	519	0.08014	0.163	0.116	0.1
4	5	84.6	525	0.378	0.6	0.6	0.1
4	6	310.4	532	0.368	0.6	0.6	0.1
5	6	170.0	532	0.0486	0.145	0.206	0.1

(a) For nodalization scheme, see Figure 3.6-129.

(b) These data are dimensionless.

(c) No data furnished since Panel B between Nodes 2 and 3 is assumed closed during postulated pipe break.

TABLE 3.6-15

MAIN STEAM TUNNEL SUBCOMPARTMENT ANALYSIS
INFORMATION FOR BLOW-OUT PANELS C AND D

1. PANEL C (HORIZONTAL, HINGED, NON-BOLTED, SHEET STEEL BLOW-OUT PANEL)

    TOTAL WEIGHT: 2060 LBS.

    AREA: 230.6 $FT.^2$

    MOMENT ARM: 3.375 FT.

    MOMENT OF INERTIA: 31,286 $LBS._{MASS} - FT.^2$

    DAMPING CONSTANT: NEGLECTED

2. PANEL D (VERTICAL, BOLTED, INSULATED METAL BLOW-OUT PANEL)

    TOTAL WEIGHT: 16,000 LBS.

    AREA: 1060.8 $FT.^2$

Reference: Figures 3.6-127 and 3.6-128

TABLE 3.6-16

SUBCOMPARTMENT ANALYSIS

NODAL VOLUME DATA FOR A POSTULATED PIPE BREAK IN THE MAIN STEAM TUNNEL EXTENSION

NODE NUMBER	DESCRIPTION	VOLUME (CU.FT.)	ELEVATION (FT.)
1	MAIN STEAM TUNNEL EXTENSION, SOUTH	2320	501
2	MAIN STEAM TUNNEL EXTENSION, NORTH	2799	501
3	TURBINE GENERATOR BUILDING, EL. 471'-0" FLOOR	728610	471
4	TURBINE GENERATOR BUILDING, EL. 441'-0" FLOOR	658938	441
5	TURBINE GENERATOR BUILDING, EL. 501'-0" FLOOR	1270590	501

NOTE: For nodalization scheme, see Figure 3.6-138

TABLE 3.6-17

SUBCOMPARTMENT ANALYSIS

FLOW JUNCTION DATA FOR A POSTULATED PIPE BREAK IN THE MAIN STEAM TUNNEL EXTENSION[a]

FROM NODE	TO NODE	JUNCTION FLOW AREA (SQ.FT.)	JUNCTION ELEVATION (FT.)	JUNCTION INERTIA ($FT.^{-1}$)	FORM LOSS COEFFICIENT[b] FORWARD FLOW	REVERSE FLOW	FRICTIONAL LOSS COEFFICIENT[b]
1	2	379.0	509	0.01395	0.6	0.56	0.1
1	3	73.6	501	0.1722	1.29	1.07	0.1
2	3	114.6	501	0.1434	1.16	0.99	0.1
3	4	230.0	471	0.1091	1.28	1.49	0.1
3	5	507.0	501	0.01176	1.54	1.55	0.1

(a) For nodalization scheme, see Figure 3.6-129.

(b) These data are dimensionless.

K•E SEMI-LOGARITHMIC 5 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

46 6212

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FIG. 3.6-123

K•E SEMI-LOGARITHMIC 5 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

46 6212

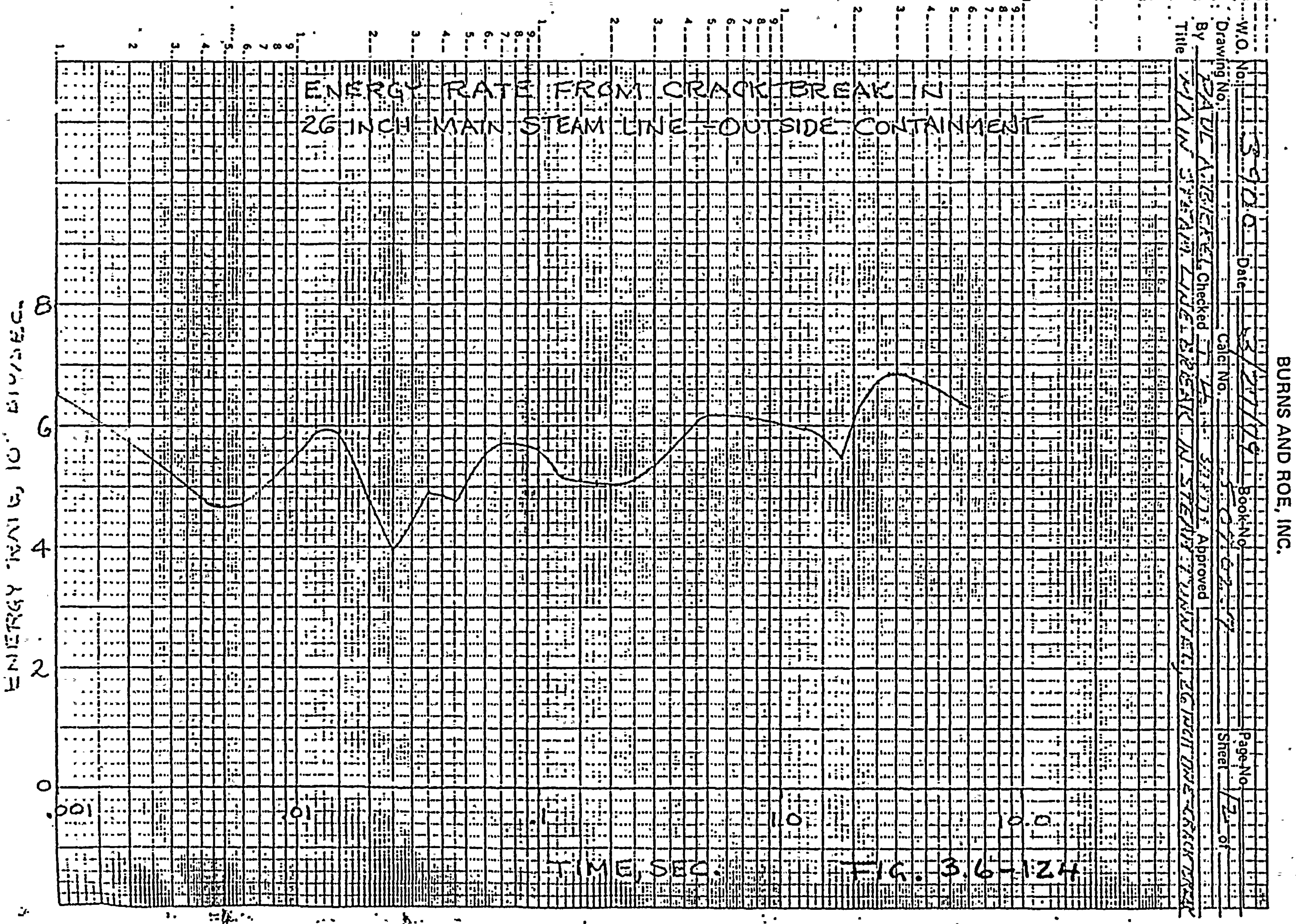

FIG. 3.6-124

K+E SEMI-LOGARITHMIC 5 CYCLES X 70 DIVISIONS KEUFFEL & ESSER CO. MADE IN U.S.A. 46 6212

BLOWDOWN MASS FLOW RATE FROM CRACK BREAK IN 24 INCH FEEDWATER LINE - OUTSIDE CONTAINMENT

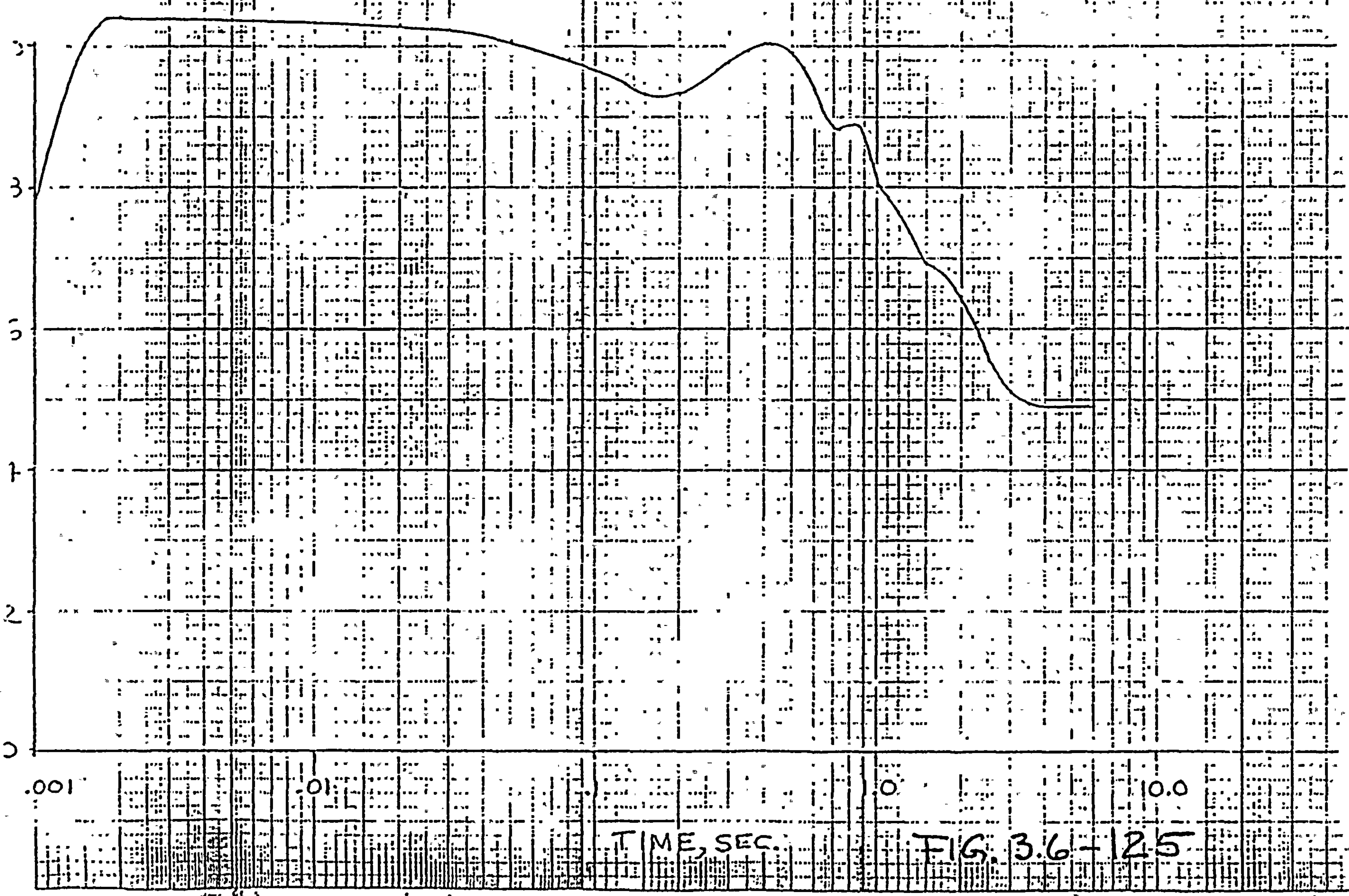

FIG. 3.6-125

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Drawing No. Calc. No. 5.07.104.3 Sheet 632 of

By P.A. BICKEL Checked NOT REQUIRED Approved

Title FEEDWATER LINE BREAK OUTSIDE CONTAINMENT, 24 INCH LINE-CRACK BREAK

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KEUFFEL & ESSER CO. MADE IN U.S.A.

46 6212

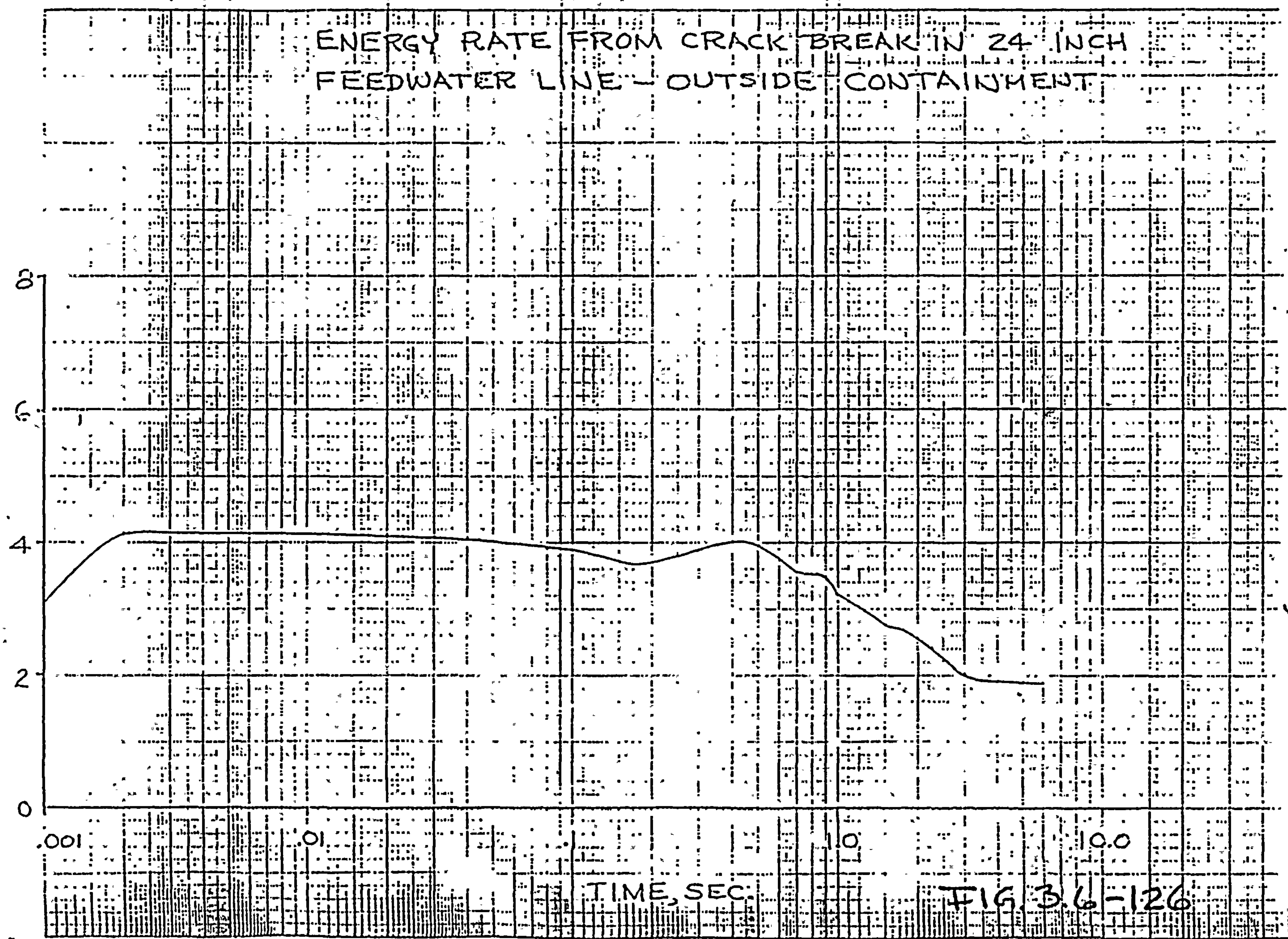

FIG. 3.6-126

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Drawing No. Calc. No. 5.07.04.3 Sheet 37 of

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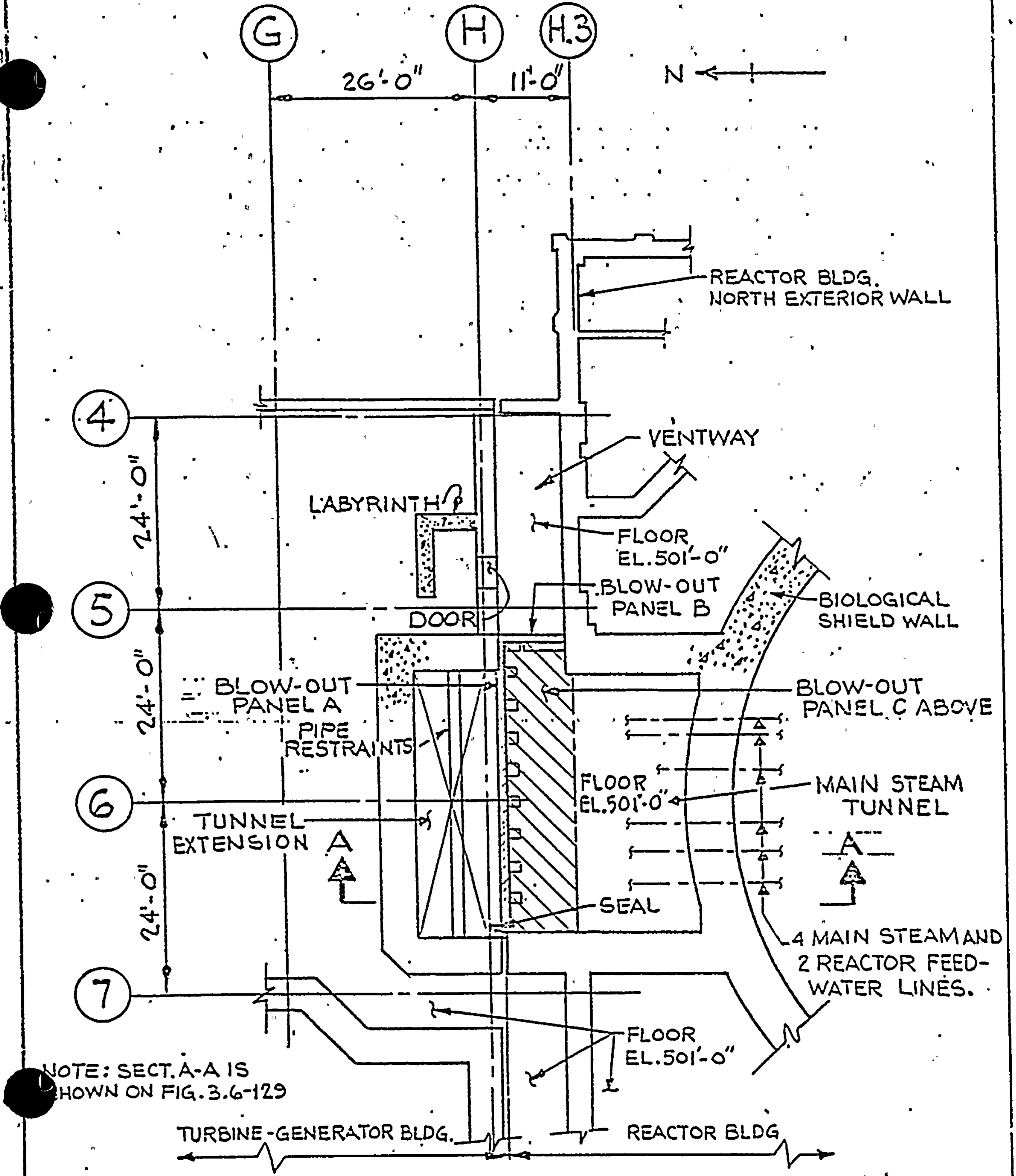
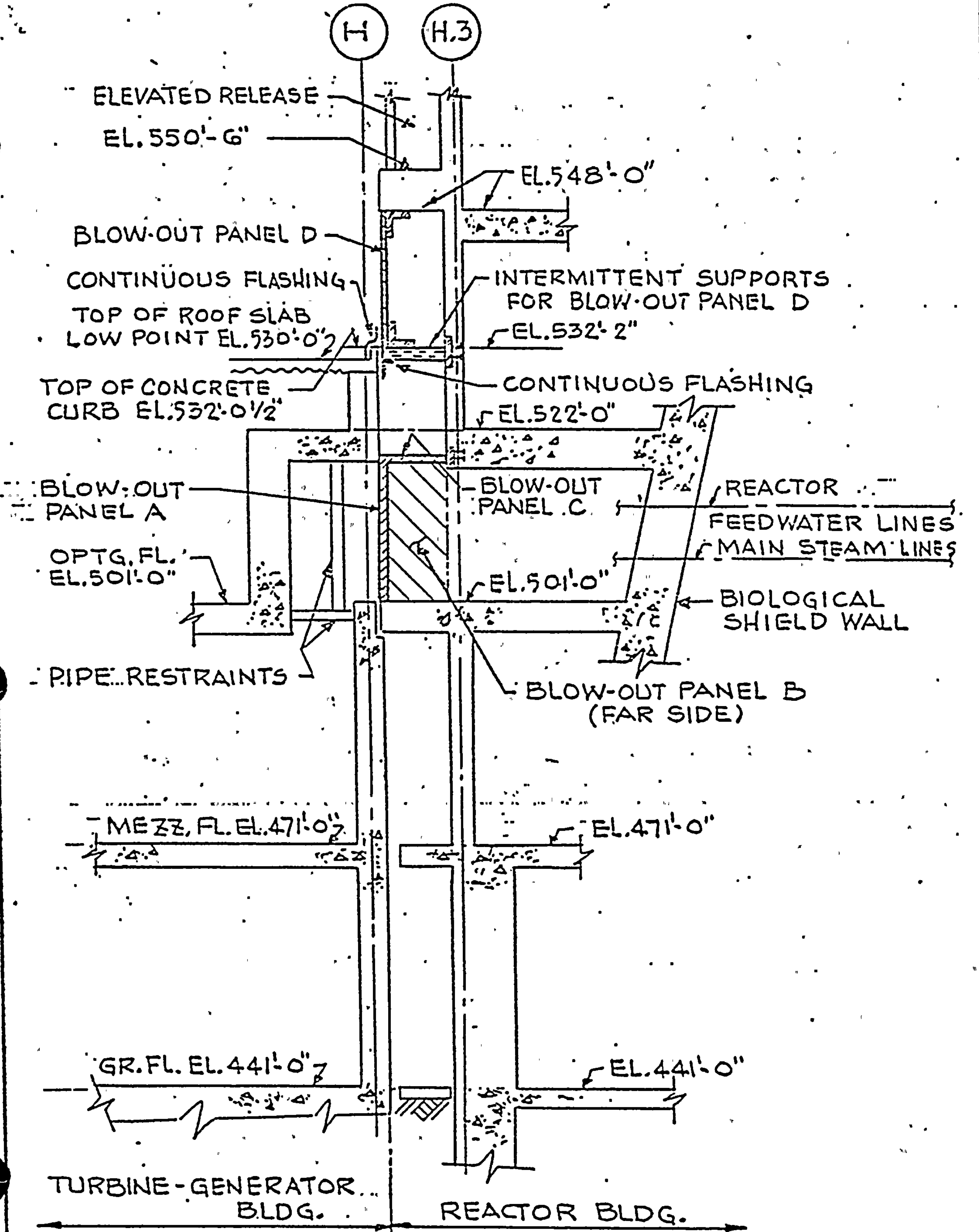
Title: FEEDWATER LINE BREAK OUTSIDE CONTAINMENT, 24 INCH LINE - CRACK BREAK



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM TUNNEL, VENTWAY AND TUNNEL EXTENSION SECTIONAL PLAN	FIG. 3.6-127



NOTE: SECTION A-A IS CUT FROM PLAN ON FIG. 3.6-127

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM TUNNEL, VENTWAY AND TUNNEL EXTENSION SECTION A-A	FIGURE 3.6.128

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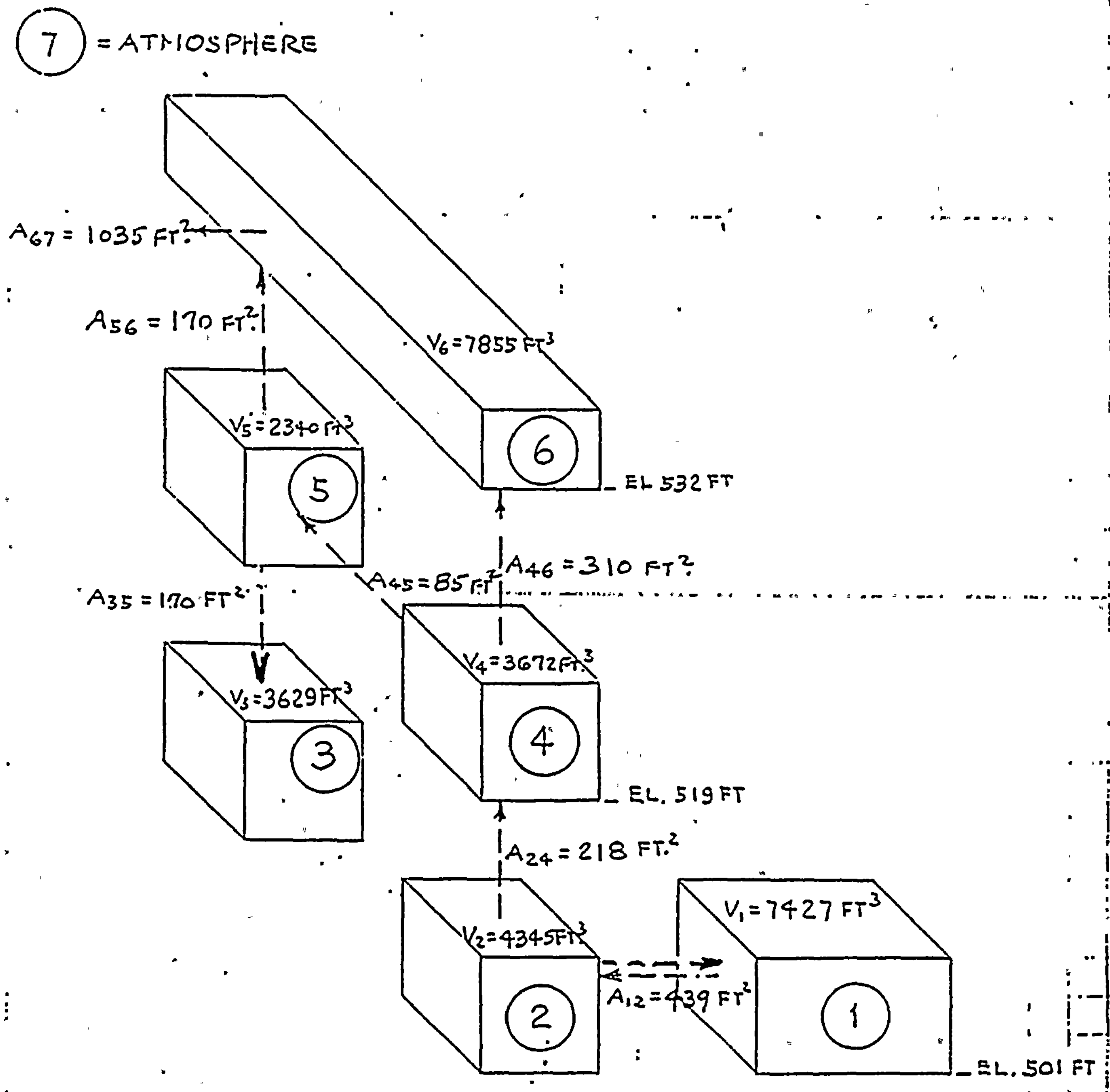
Drawing No. ______ Calc. No. 5.6.7.72 Sheet 22 Cont. on Sheet 23
By [illegible] Checked [illegible] Approved ______
Title PRESSURIZATION OF MAIN STEAM TUNNEL FROM A MS LINE BREAK



REACTOR BUILDING MODEL

FIGURE 3.6-129

(900 MST SHORT TERM PRESSURE ROM MSLB IN RBN1R1RELAP4/C05 03/76 C

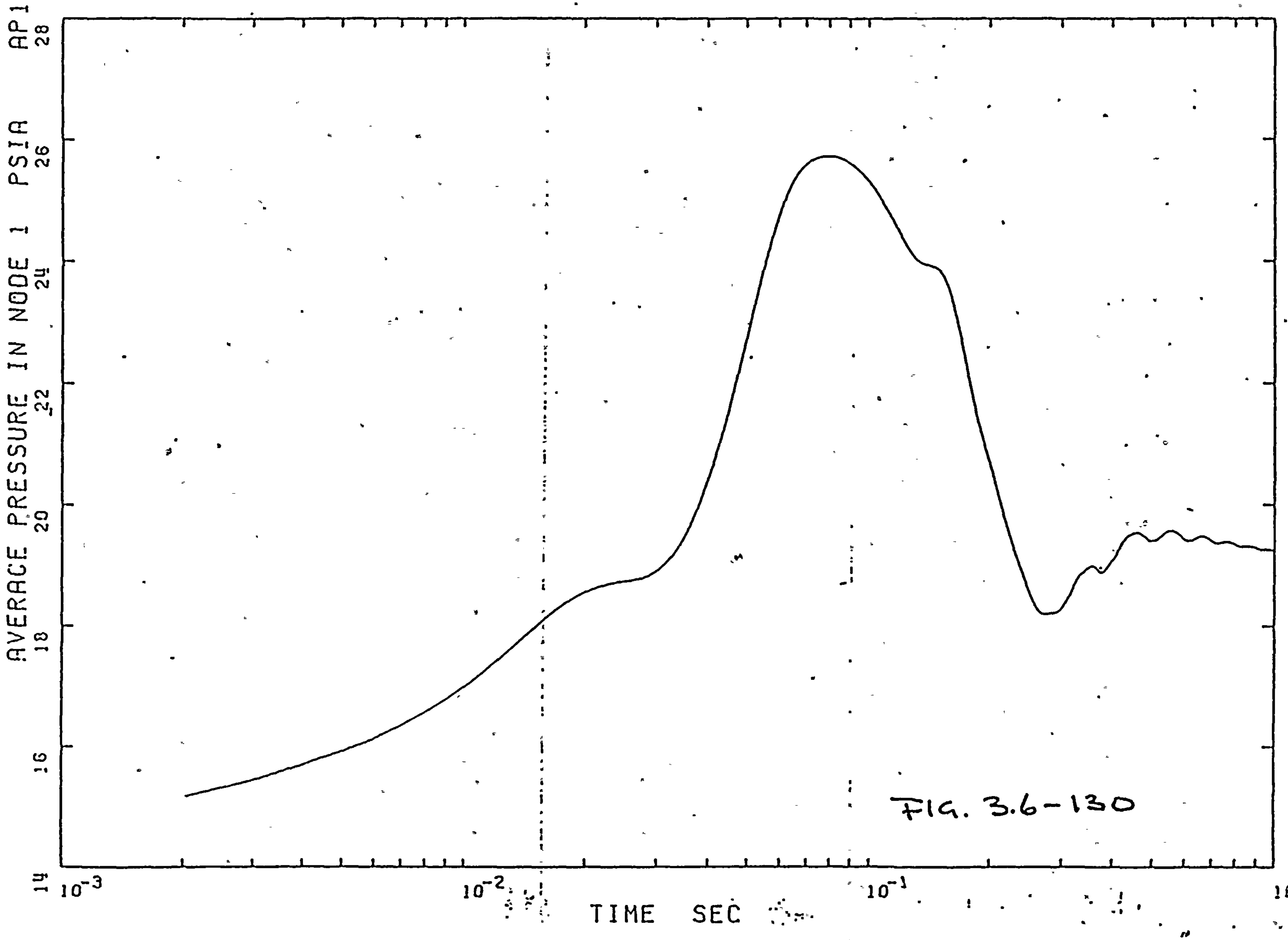


FIG. 3.6-130

W[illegible]900 MST SHORT TERM PRESSURE FROM MSLB IN RBN1R1RELAP4/C05 [illegible]/03/76 0

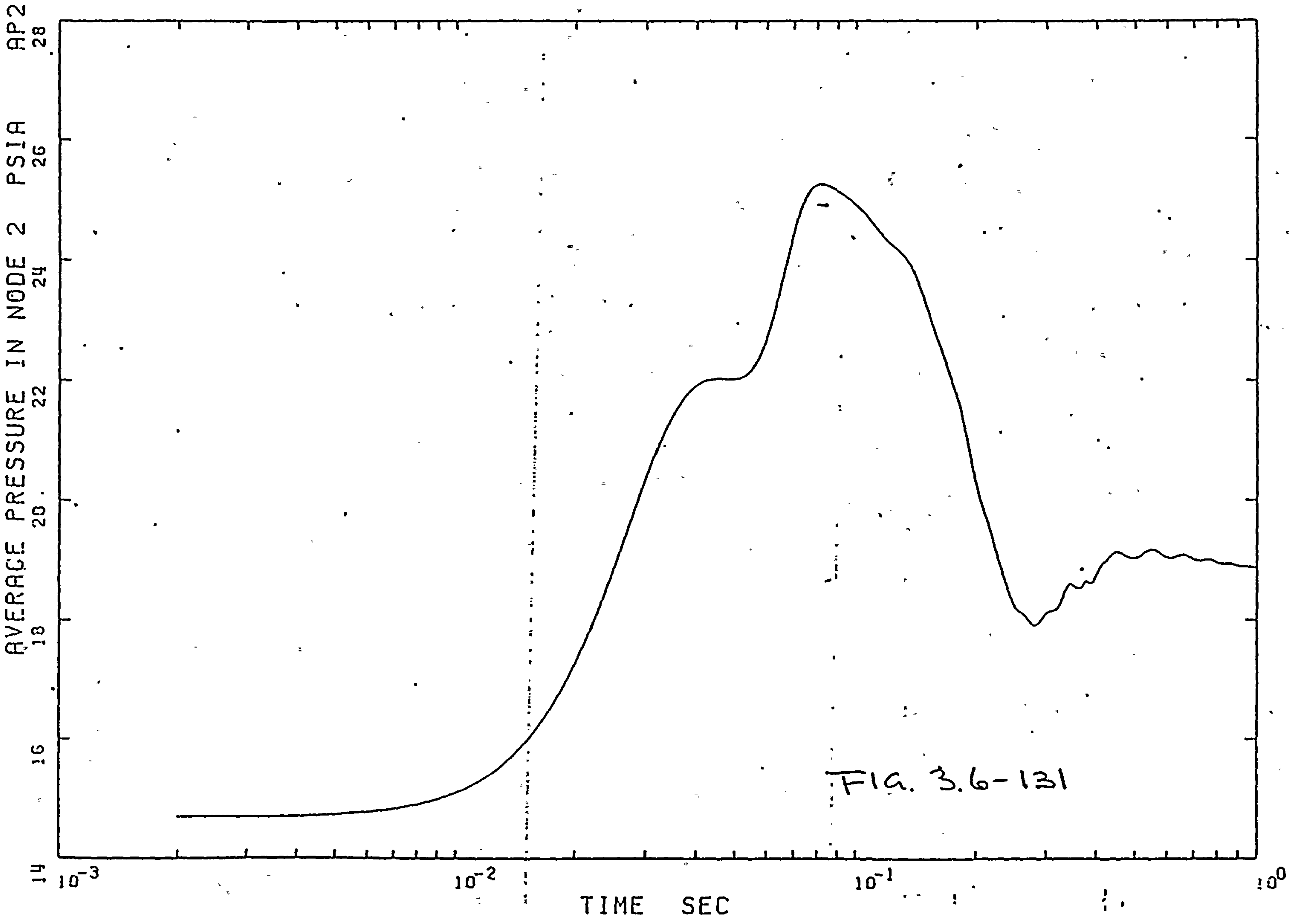


FIG. 3.6-131

WO3900 MST SHORT TERM PRESSURE(OM MSLB IN RBN1R1RELAP4/C05 03/76 07

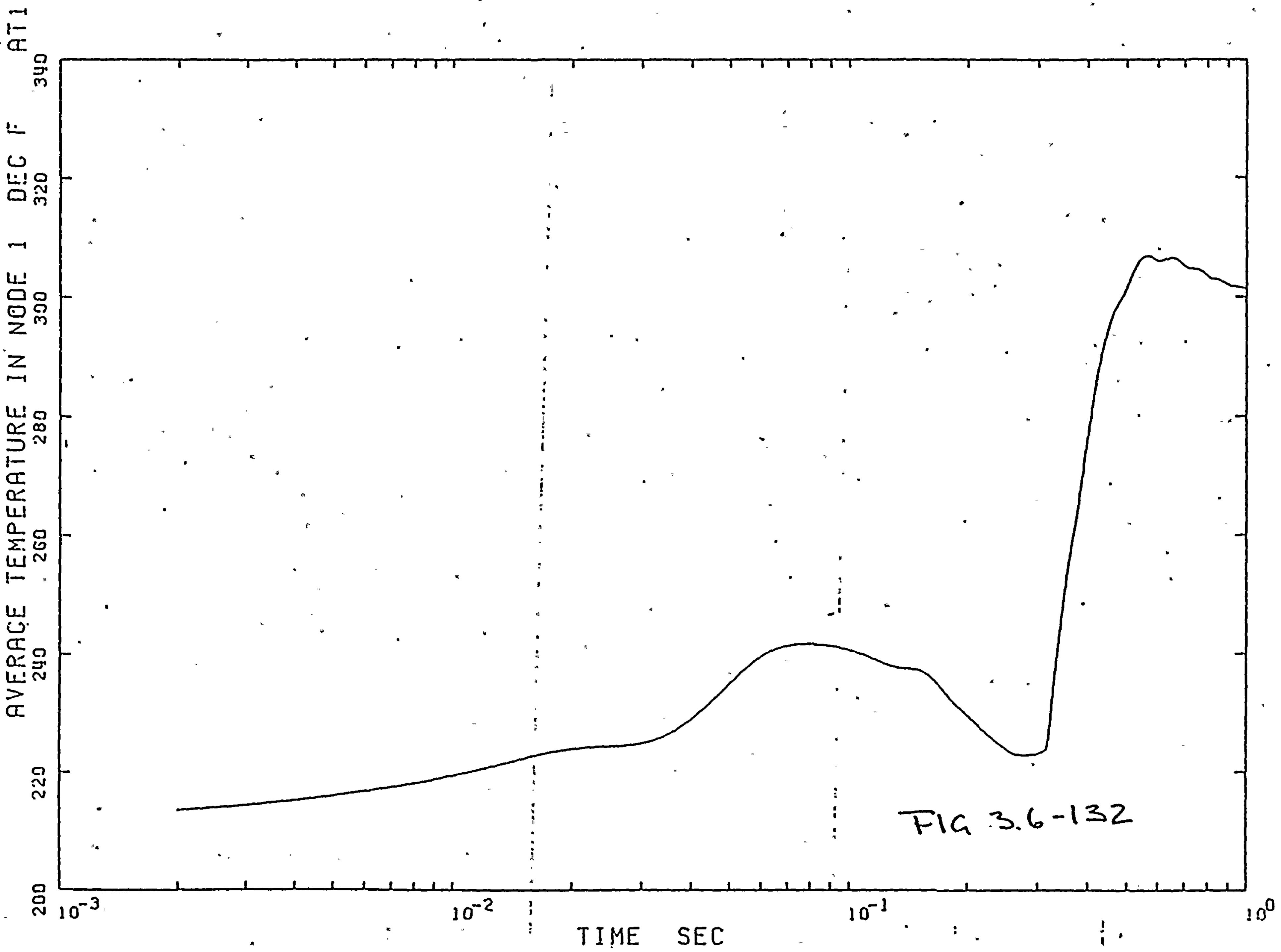


FIG 3.6-132

W 300 MST SHORT TERM PRESSURE ROM MSLB IN RBN1R1RELAP4/C05 03/76 0'

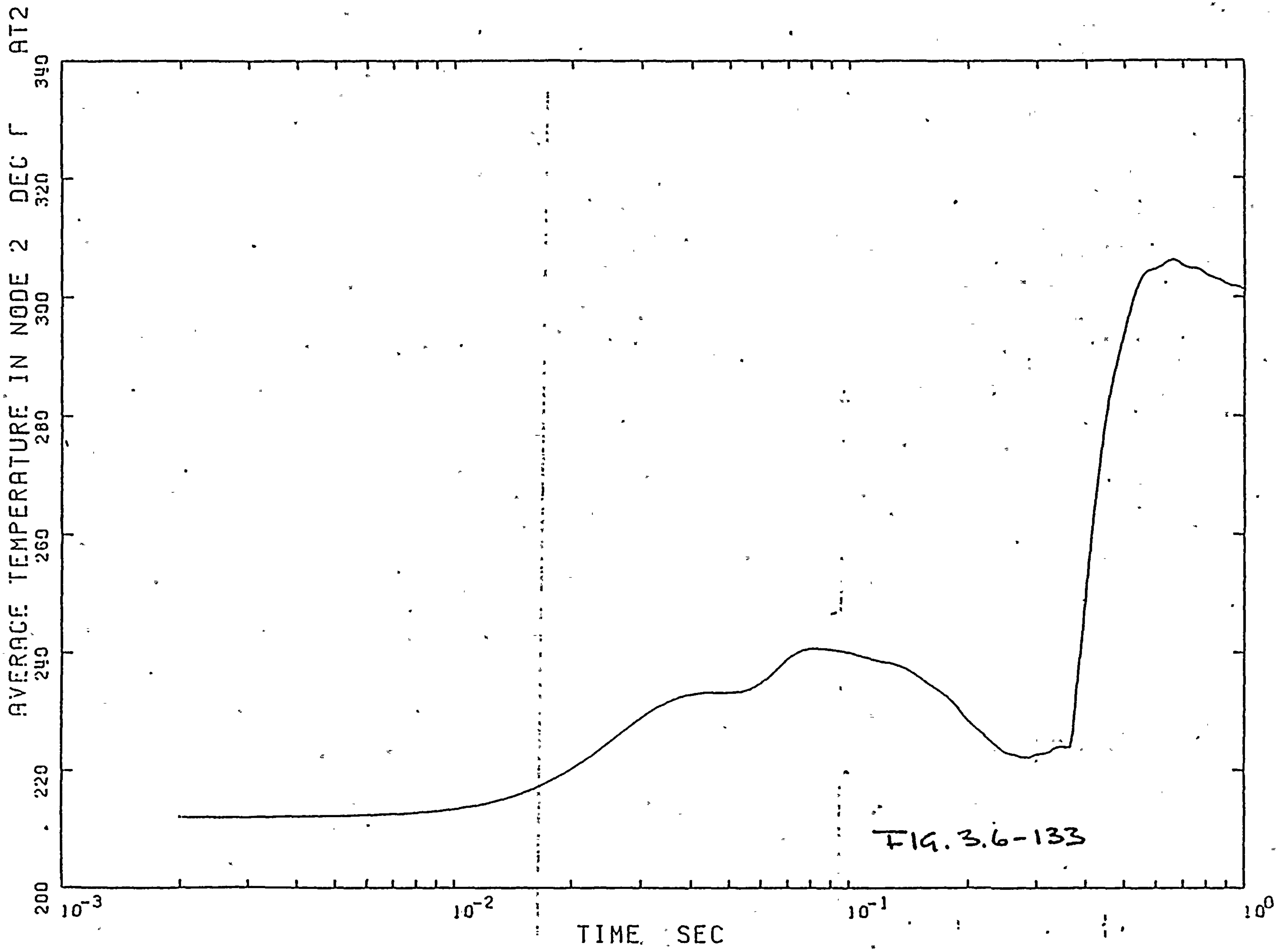


FIG. 3.6-133

W[illegible]00 MST SHORT TERM PRESSURE [illegible]OM MSLB IN RBN2R1RELAP4/C05 [illegible]03/76 07

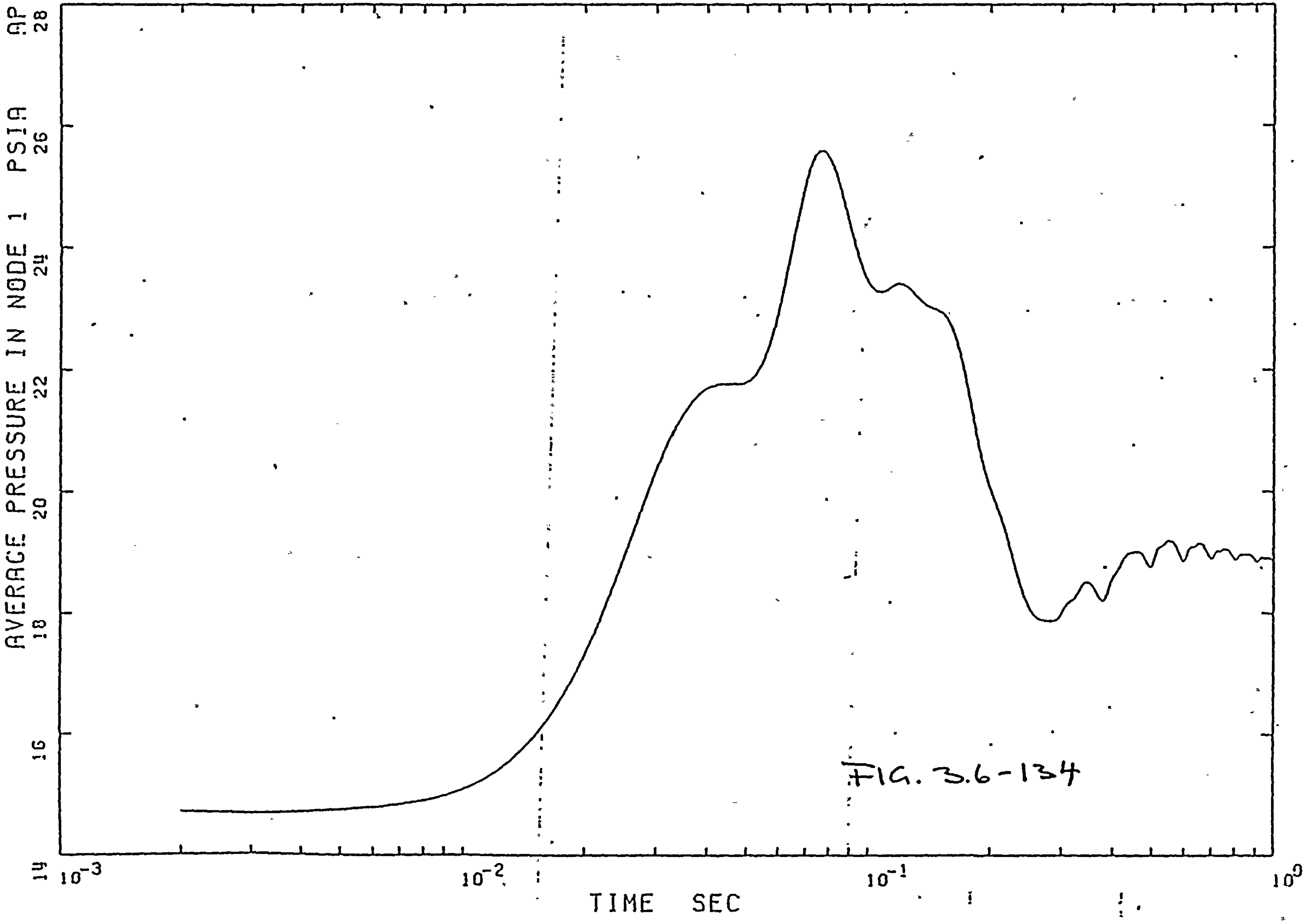


FIG. 3.6-134

W05900 MST SHORT TERM PRESSURE FROM MSLB IN RBN2R1RELAP4/C05 03/76 07

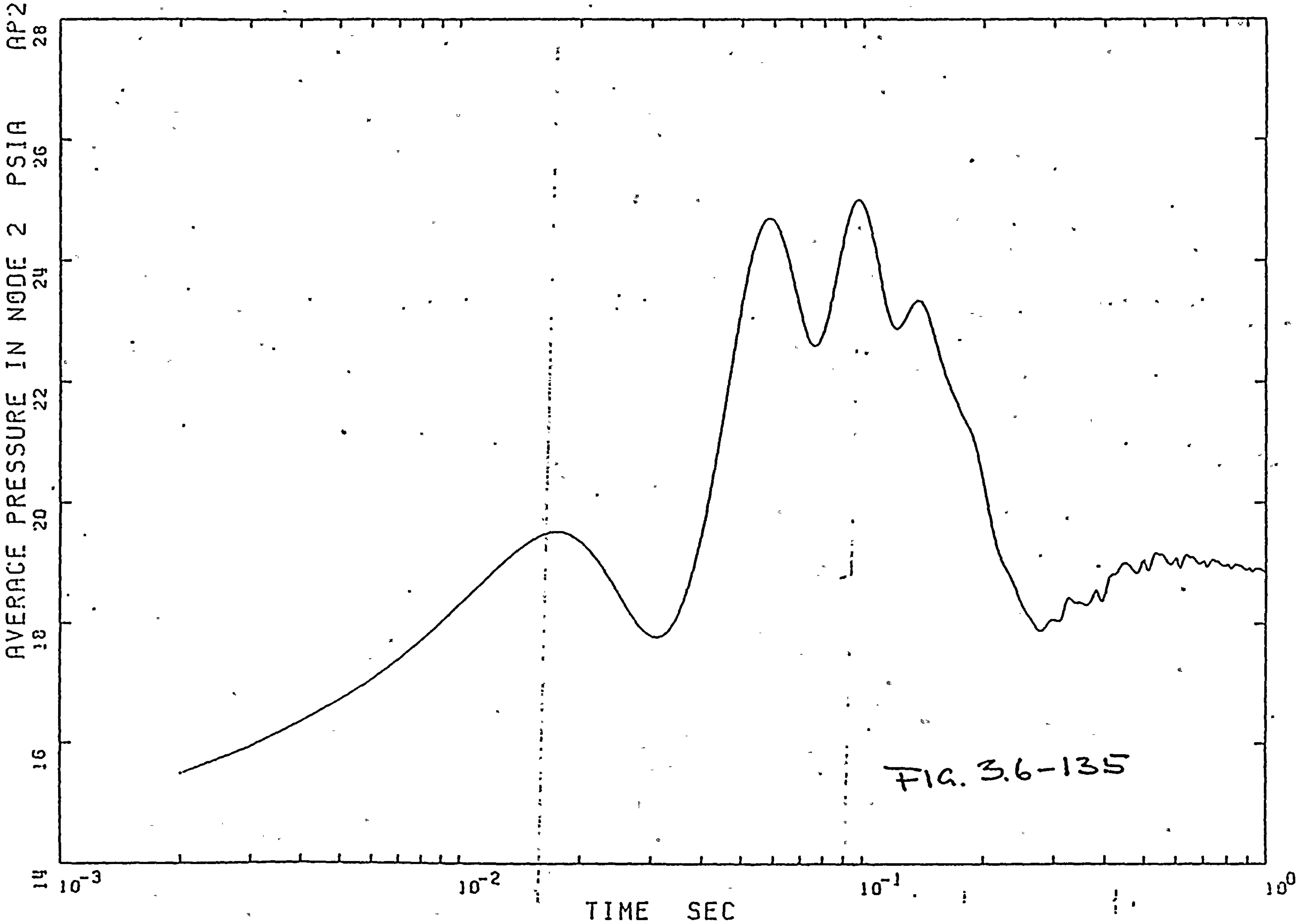


FIG. 3.6-135

3900 MST SHORT TERM PRESSUR ROM MSLB IN RBN2R1RELAP4/C05 /03/76 0

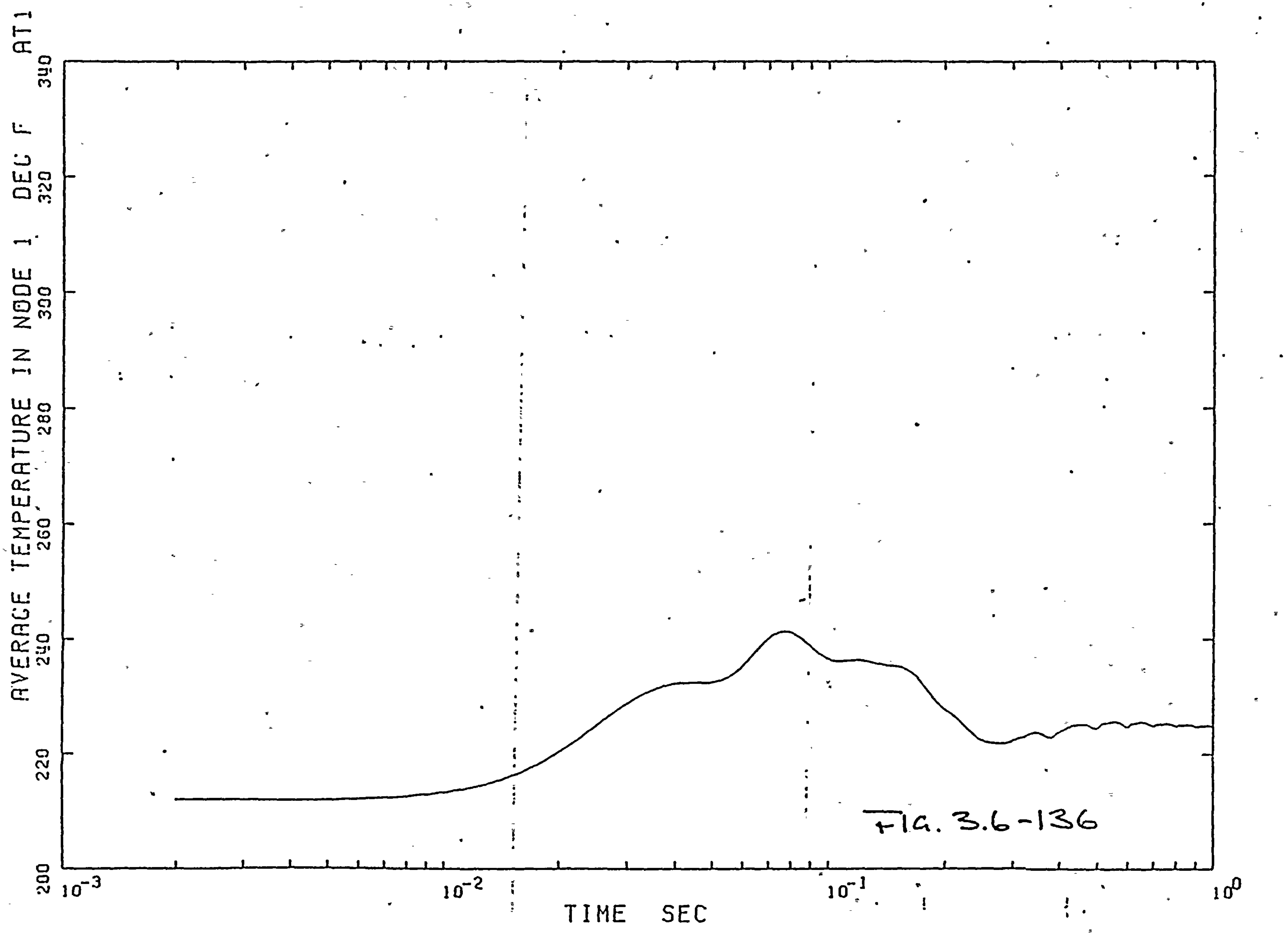


FIG. 3.6-136

(. 900 MST SHORT TERM PRESSUR ROM MSLB IN RBN2R1RELAP4/C05 '03/76 C

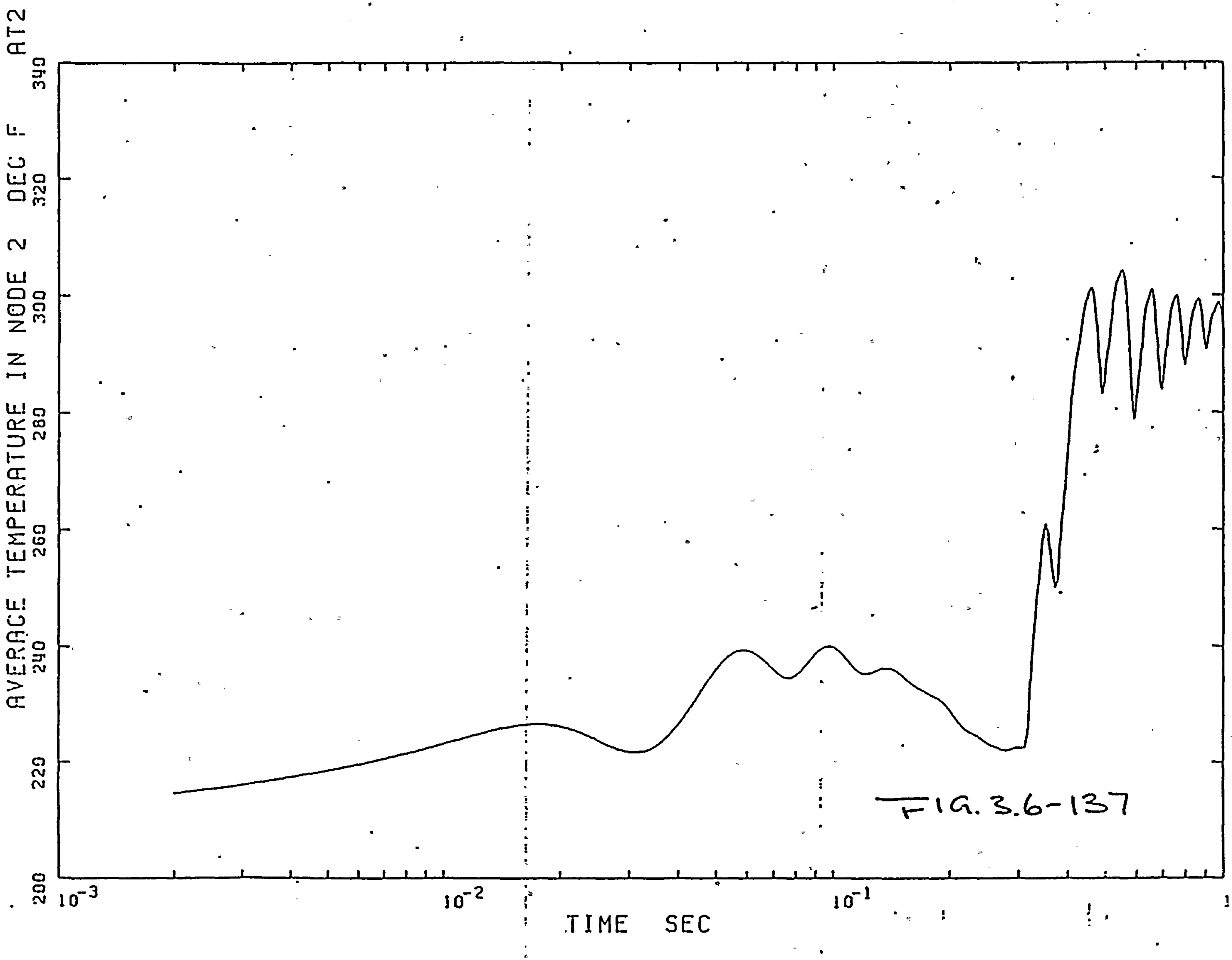


FIG. 3.6-137

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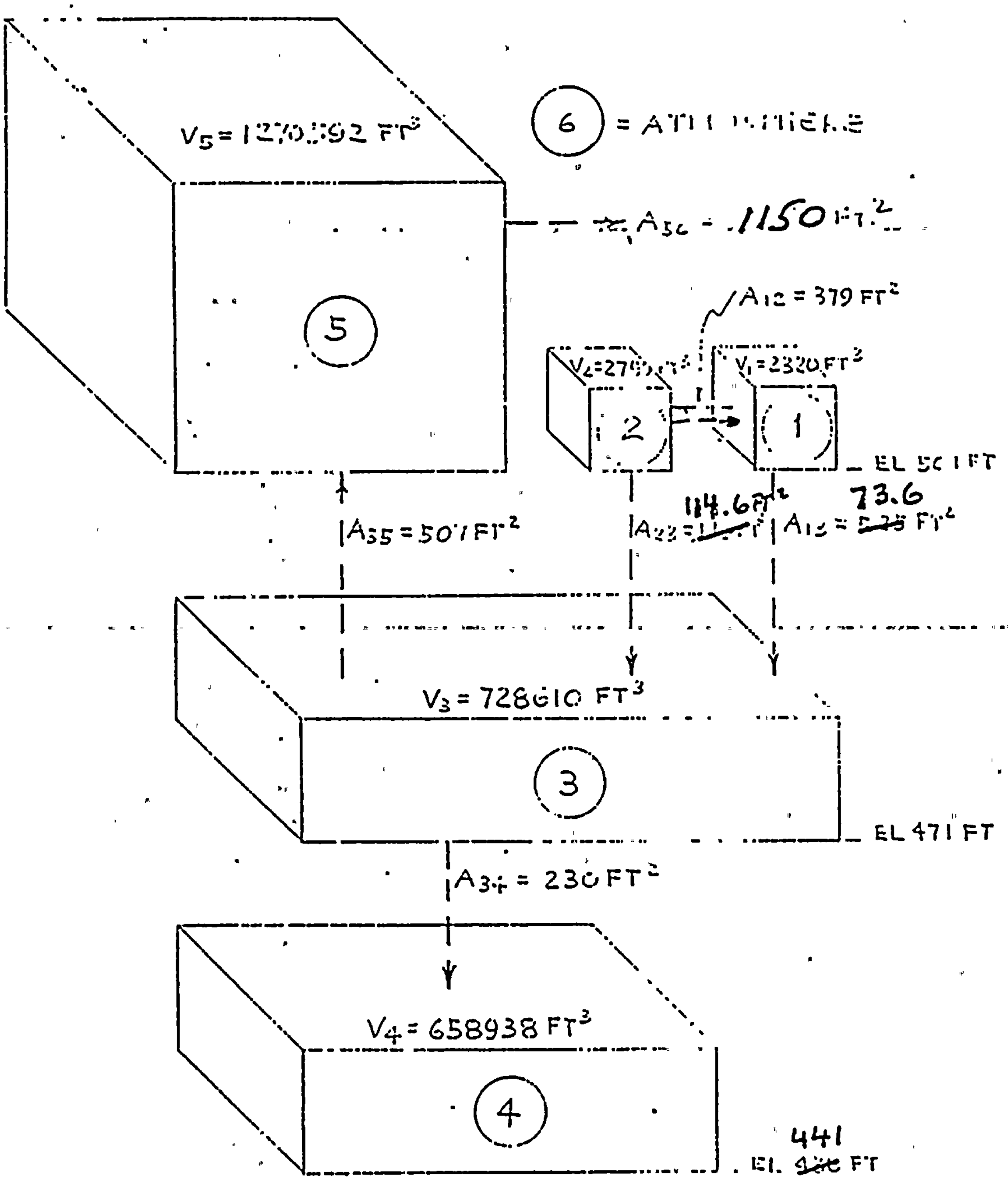
Drawing No. ____ Calc. No. 5.07.72 Sheet 41 Cont. on Sheet 42
By ____ Checked ____ Approved ____
Title ____



TURBINE BUILDING MODEL

FIGURE 3.6-138

Form BR 8002-2 (5/78)

W 00 MST SHORT TERM PRESSUR(OM MSLB IN TBN1R1RELAP4/C05 03/76 07

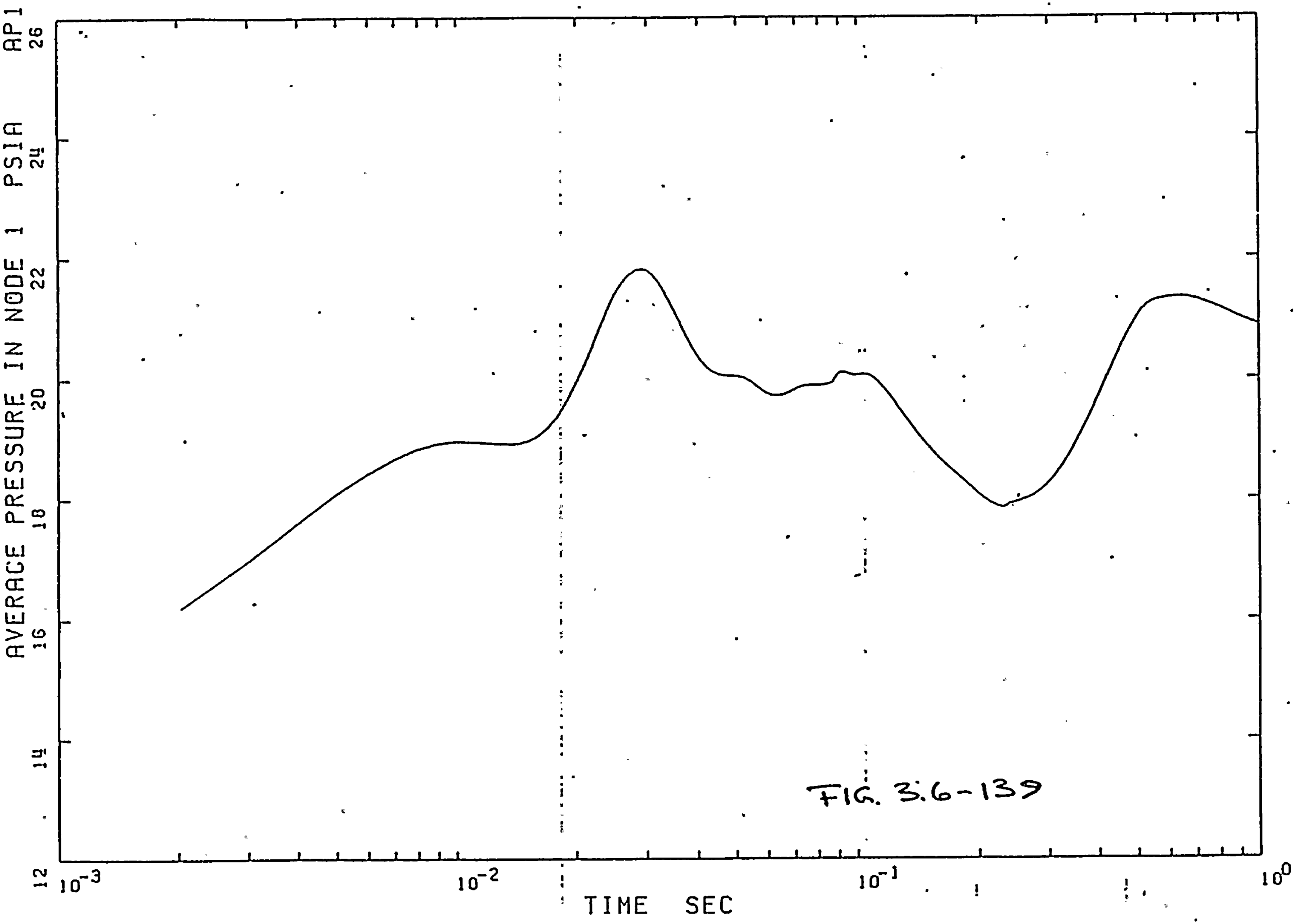


FIG. 3.6-139

( ) 3900 MST SHORT TERM PRESSUR ROM MSLB IN TBN1R1RELAP4/C05 03/76 C

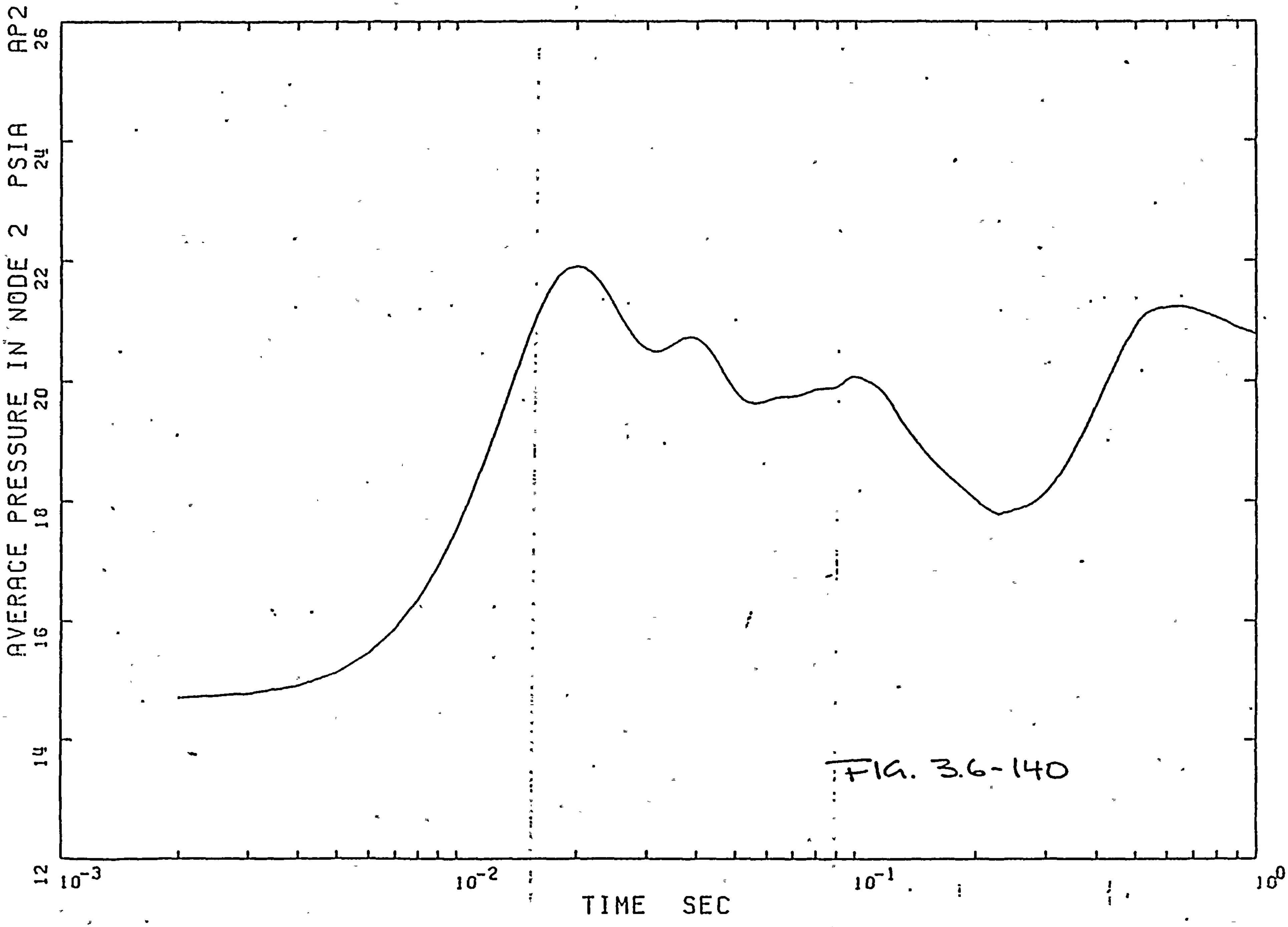


FIG. 3.6-140

W08900 MST SHORT TERM PRESSURE FROM MSLB IN TBN1R1RELAP4/C05 [illegible]/03/76 0

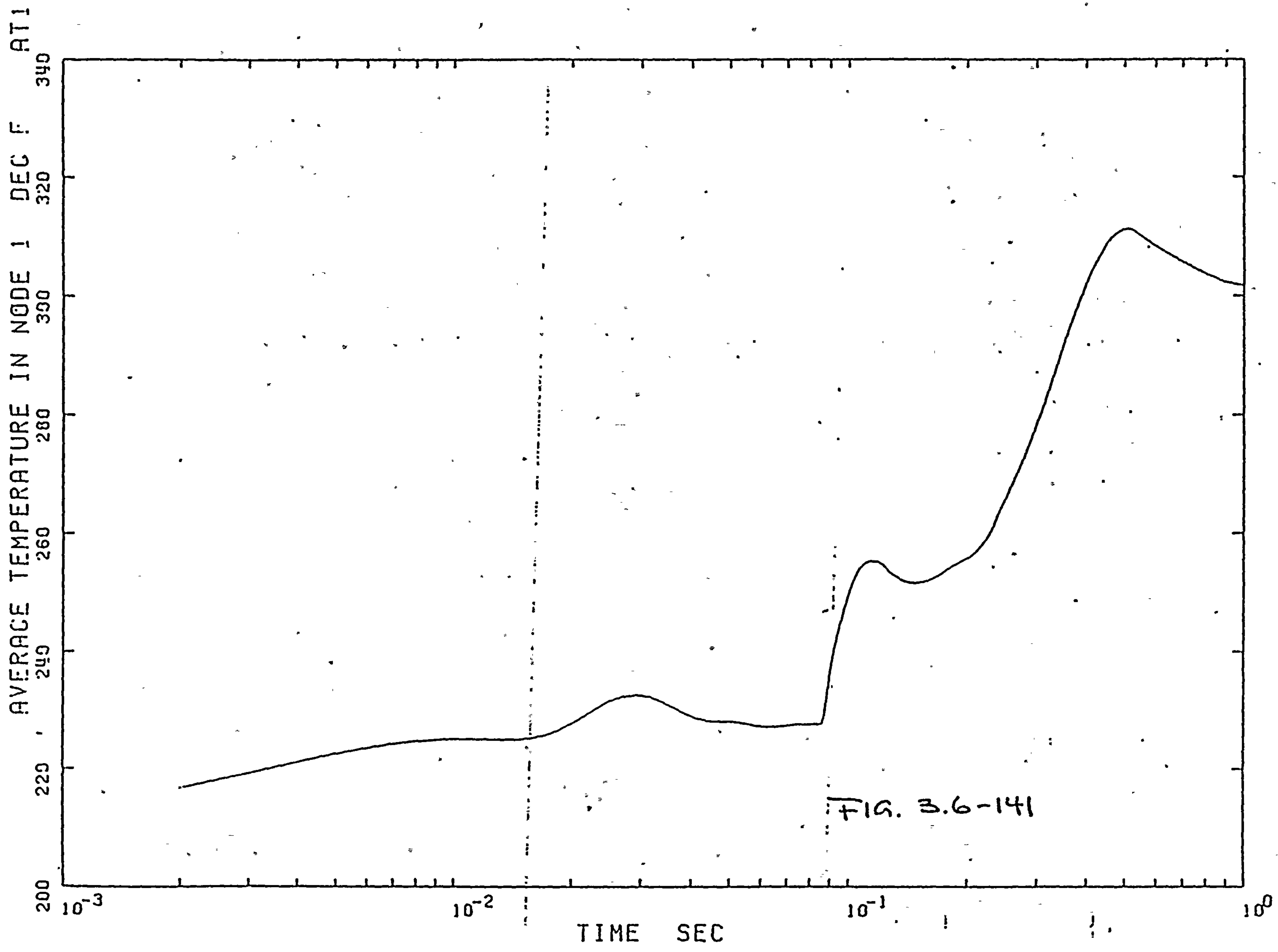


FIG. 3.6-141

W 900 MST SHORT TERM PRESSURE OM MSLB IN TBN1R1RELAP4/C05 03/76 .07

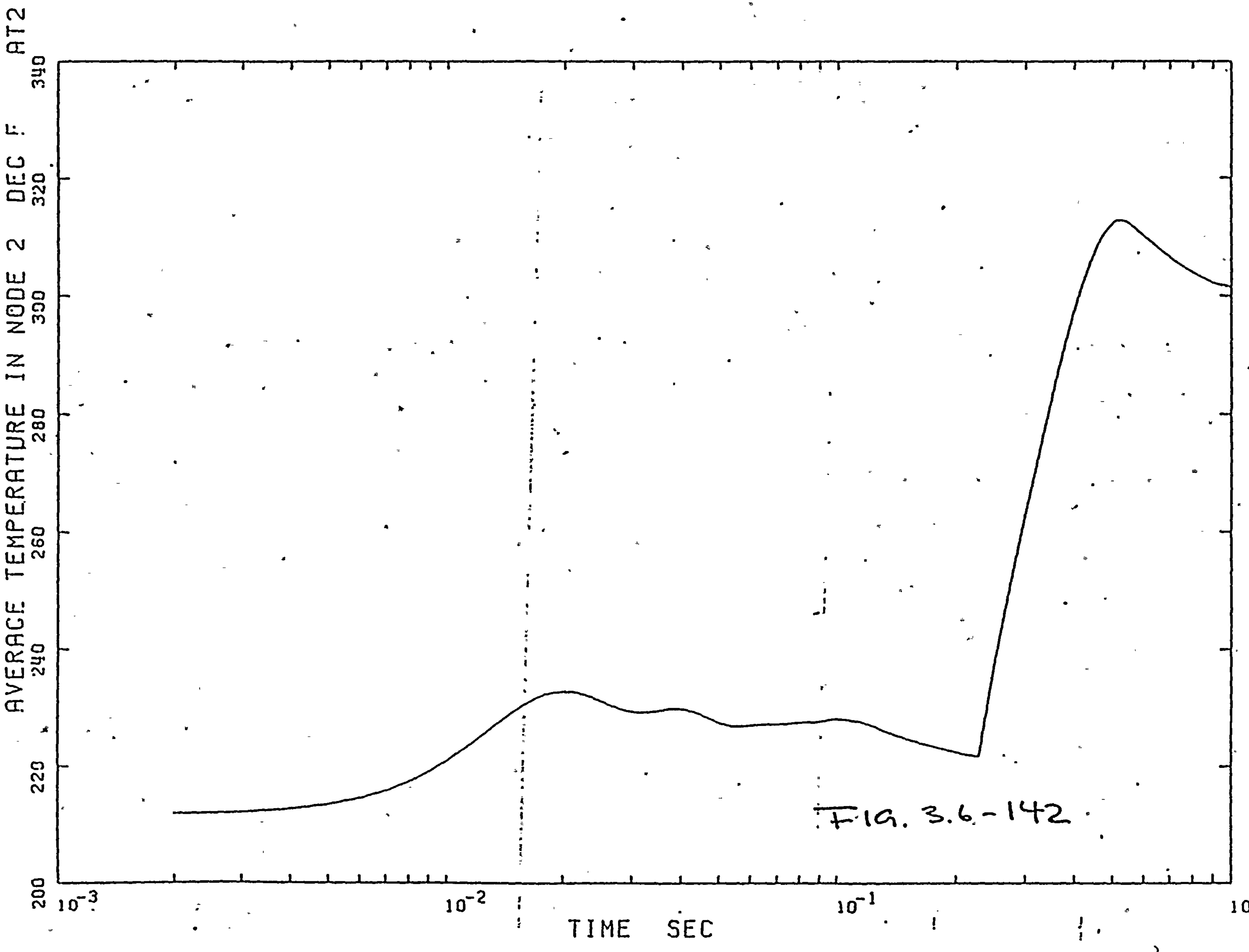


FIG. 3.6-142

W 300 MST SHORT TERM PRESSURE OM MSLB IN TBN2R1RELAP4/C05 03/76 07

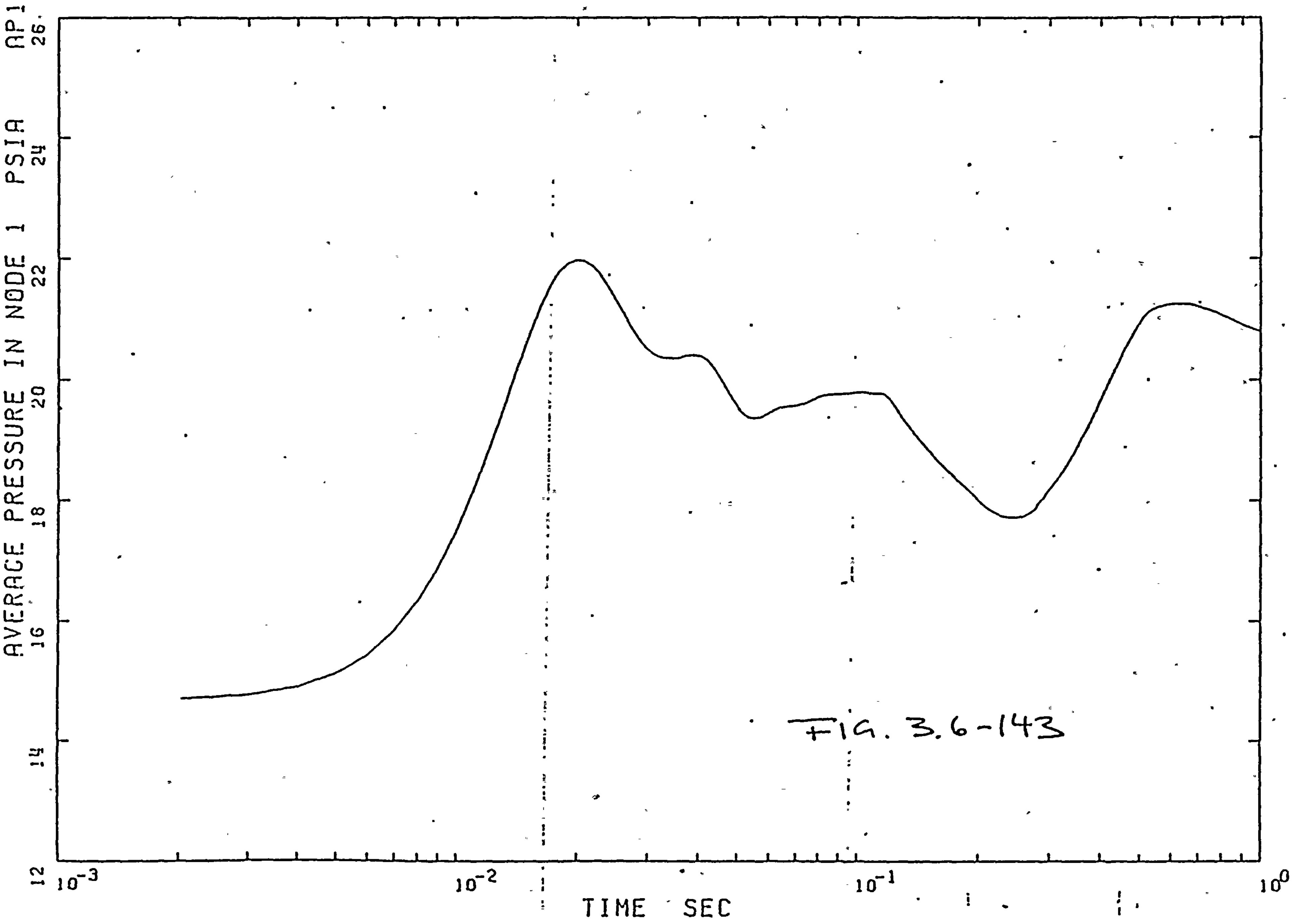


FIG. 3.6-143

(900 MST SHORT TERM PRESSUR ROM MSLB IN TBN2R1RELAP4/C05 /03/76 0

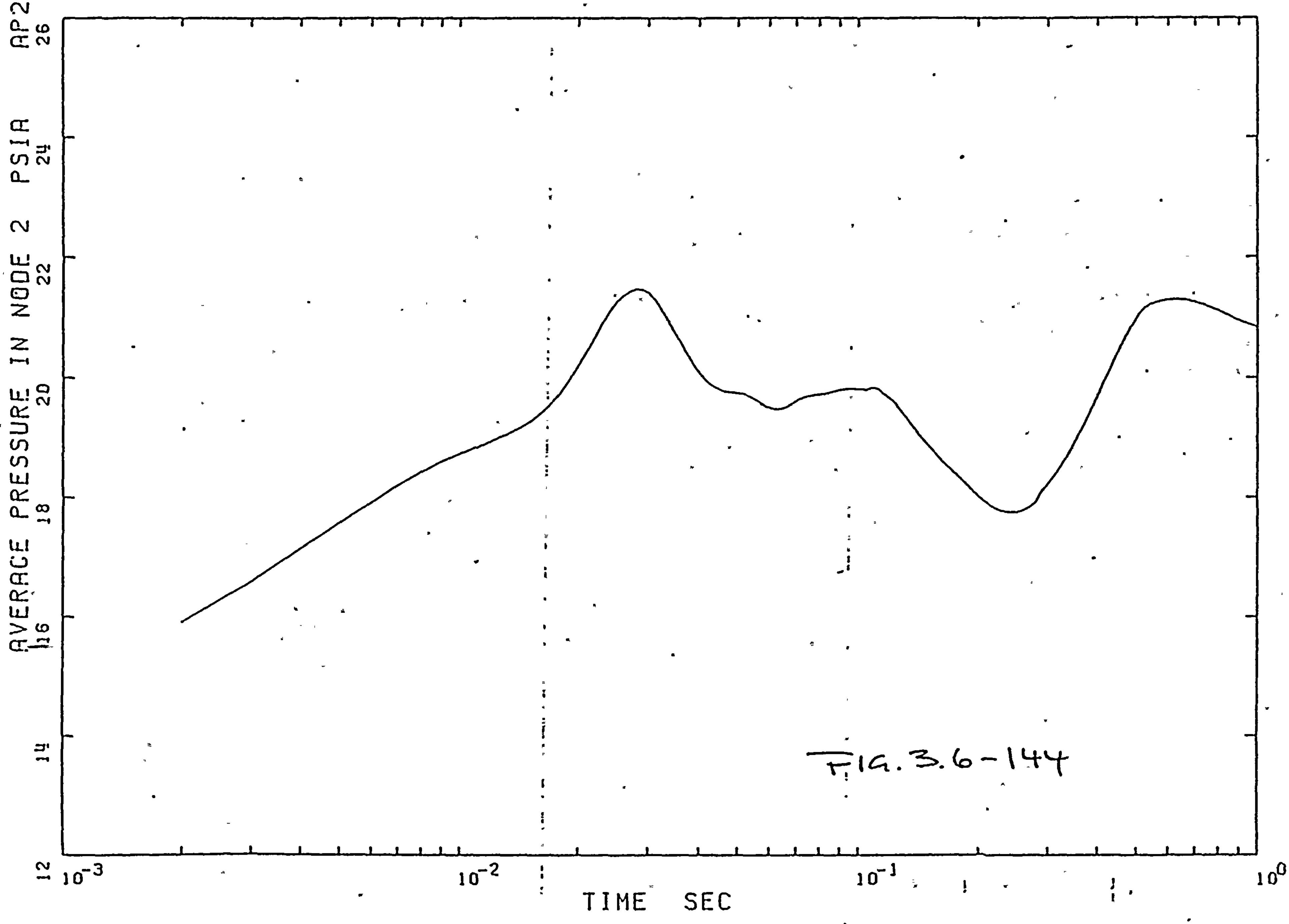


FIG. 3.6-144

(6900 MST SHORT TERM PRESSUR ROM MSLB IN TBN2R1RELAP4/C05 /03/76 0

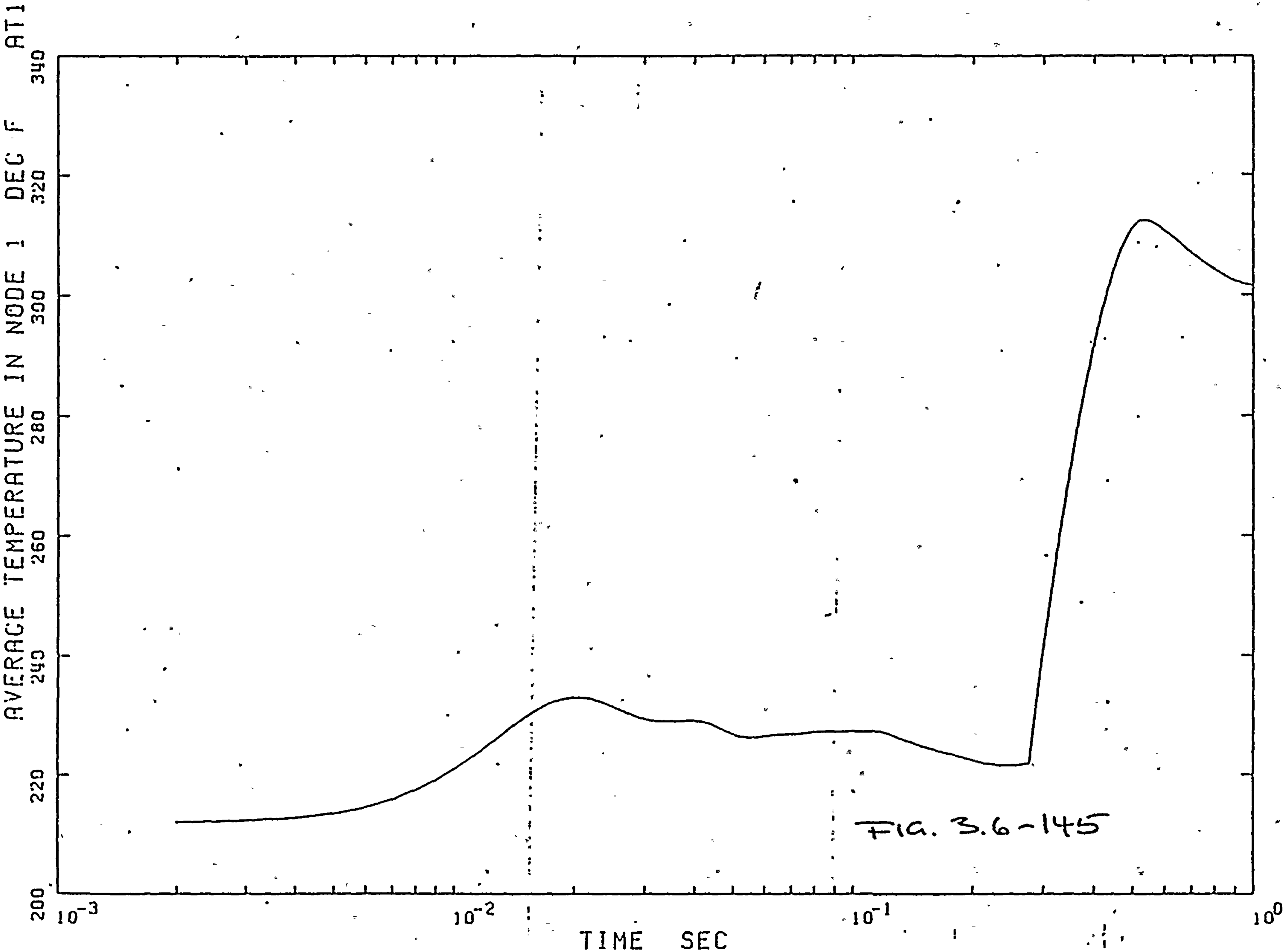


FIG. 3.6-145

(8900 MST SHORT TERM PRESSUR ROM MSLB IN TBN2R1RELAP4/C05 '03/76 0

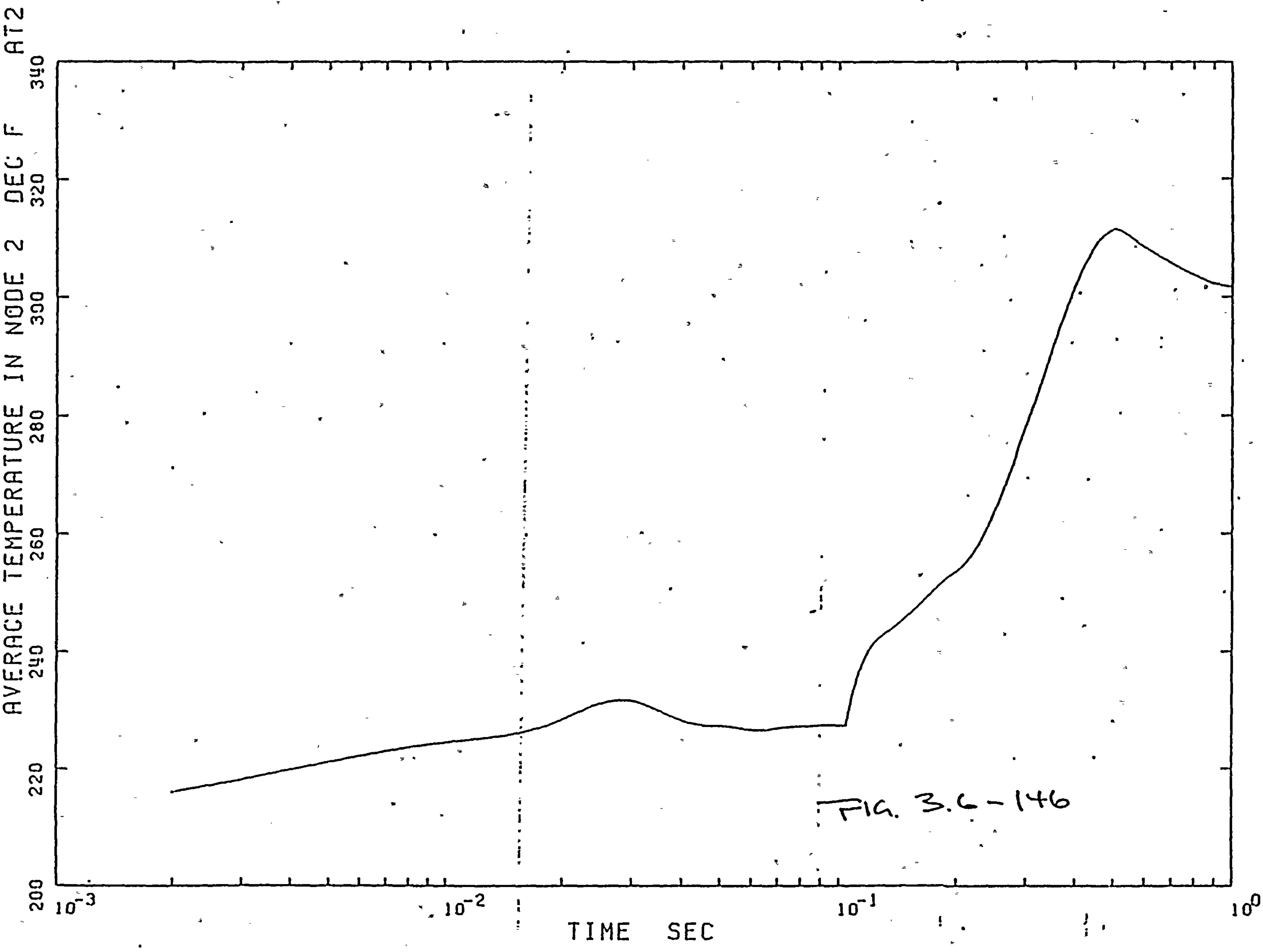


FIG. 3.6-146

The biological shield wall is a reinforced concrete structure varying in thickness from five feet to six feet.

The main function of the biological shield wall is to serve as a radiation shield around the primary containment vessel. it also functions as a major mechanical barrier for the protection of the containment and reactor system against missiles that may be generated external to the primary containment. In addition to the above functions, the biological shield supports the various reactor building floor elevations that frame into it; and, as part of the reactor building structure, it resists the earthquake - induced forces and the pipe rupture-induced forces acting on the reactor vessel and sacrificial shield wall and transferred to it through the stabilizer truss system discussed in 3.8.2.

3.8.4.1.1.4 Main Steam Tunnel, Ventway and Tunnel Extension

Refer to Figures 1.2-5, 1.2-6, 3.8-2, 3.8-30 through 3.8-33, 3.8-38, 3.8-39, 3.8-54 and 3.8-55.

Refer to 3.6.1.20 for the methods used to predict blowdown mass and energy release rates and pressure transient in the main steam tunnel in the reactor building, the main steam tunnel extension in the turbine-generator building and the ventway to the atmosphere attached external to the north exterior wall of the reactor building.

The reinforced concrete main steam tunnel and tunnel extension enclose the four main steam-to-turbine pipelines, the two feedwater-to-reactor vessel pipelines and a portion of the ~~line which supplies steam to the RCIC turbine~~. In the reactor building, the pipelines are thus enclosed in the main steam tunnel from the primary containment vessel to the north exterior wall of the reactor building. At the reactor building north exterior wall the main steam tunnel interfaces the turbine generator building and the pipelines continue into the turbine generator building through the main steam tunnel extension. The main steam lines in the turbine generator building are referred to in 3.8.4.1.3.

reactor water cleanup return line which connects to the reactor feedwater lines.

A separation gap is provided between the north end of main steam tunnel, which is part of the north exterior wall of the reactor building secondary containment, and the turbine generator building to permit differential movements.

Access to the main steam tunnel is provided from inside the reactor building secondary containment in the main steam tunnel west wall at floor elevation 501'-0". Removable reinforced concrete shield plugs are provided in the main steam tunnel roof at floor elevation 522'-0" for maintenance. The removable plugs are adequately anchored to the main steam tunnel walls and provide the same structural integrity as the main steam tunnel itself.

Overpressurization of the main steam tunnel and tunnel extension is prevented by venting the main steam tunnel and tunnel extension to the atmosphere (by way of the ventway) and to the turbine generator building as described in 3.6.1.20. Blow-out panels are placed at the north end and at the east side of the main steam tunnel. The panel at the north end of the main steam tunnel interfaces the tunnel extension in the turbine-generator building. The panel at the east side of the main steam tunnel interfaces the ventway; and the ventway interfaces the atmosphere by way of a blow-out panel which displaces away from the exterior wall of the ventway (above the main steam tunnel) out to the exterior of the ventway. The blow-out panels are designed to blow off and permit venting when the pressure generated in the main steam tunnel, ventway and tunnel extension by a postulated pipe break within the main steam tunnel or tunnel extension exceeds the value specified in 3.6.1.20.

Venting of the main steam tunnel, ventway and tunnel extension is ensured by controlled release type fasteners used in the fastening of the blow-out panels. The release of the panel into the turbine generator building does not affect the ability to shut down the reactor, integrity of the primary containment vessel and other Seismic Category I or safety related structures and the capability of the essential heat removal systems to perform their intended design functions. A structural steel framed structure is erected inside the tunnel extension. It is designed to support the pipes during normal operation and seismic disturbances and to provide backup support for pipe whip restraint in the event of a postulated pipe break. (See Figures 3.6-6g through 3.6-6k).

The main steam tunnel is designed as a rigid reinforced concrete structure supported on one end by the biological shield wall and on the other end by the north exterior wall of reactor building. The main steam tunnel and tunnel extension are designed to withstand the effects induced by a postulated pipe break inside the main steam or tunnel extension, such as jet forces, whipping pipes and missiles. The main steam tunnel also protects the piping within it from the effects induced by a pipe rupture, missile or other disturbance occuring outside the main steam tunnel in secondary containment. Refer to discussion in 3.6.1 on protection against pipe breaks outside primary containment.

The main steam tunnel and tunnel extension concrete also provide the shielding protection required in secondary containment against the sources of radiation from the piping within the main steam tunnel and tunnel extension.

3.8.4.1.1.5 Operating Floor, Steel Superstructure and Overhead Bridge Crane

The operating floor is the uppermost level in the reactor building and is the floor from which the reactor vessel is refueled. Refer to Figures 1.2-6 and 3.8-36. The floor slab varies in thickness from 1'-6" to 3'-0". It is supported monolithically by the walls of the refueling pools along its interior perimeter and monolithically by the exterior walls of the reactor building; along its exterior perimeter.

The structural steel superstructure enclosing the floor provides unobstructed access to the floor for the overhead bridge crane. The superstructure consists of a conventionally braced framed system with trusses spanning 130'-0" in the north-south direction, which is the full width of the superstructure.

f. Abnormal Loads

Abnormal loads are loads generated by the design basis accident under consideration.

P_a = Maximum differential pressure equivalent static load within or across a compartment generated by the postulated pipe break, and including an appropriate margin to account for uncertainty in the calculations. A small break case is also investigated.

P_a loads are due to a high-energy pipe break outside containment and are discussed in 3.6.1; this includes pipe break in the main steam tunnel, ventway and tunnel extension discussed in 3.6.1.20.

P' = Negative internal pressure or positive internal pressure (noted below) relative to the outside atmosphere and acting only within the reactor building secondary containment in conjunction with other loading including the design basis tornado or the safe shutdown earthquake (SSE).

(1) Positive internal pressure = (+) 0.25 psig

(2) Negative internal pressure = (−) 0.012 psig

T_a = Effects of thermal environment on the structure generated by a postulated pipe break. This includes T_o for all other areas not affected by the pipe break. (See 3.6.1).

R_a = Effects of thermal environment on the pipe reactions on the structure and equipment reactions on the structure generated by a postulated pipe break. This includes R_o for all other areas not affected by the pipe break. (See 3.6.1).

a. Exterior and interior walls: 39'-0" pressure head internally confined by the walls

b. Floor slab: 22'-0" pressure head acting upward on the slab

For lateral soil pressures on exterior surfaces of the subgrade walls, see 3.7.2.

The seismic shear forces on the exterior (shear) walls are obtained from the seismic analysis described in 3.7.2.

Loadings due to a high-energy line break outside the containment are discussed in 3.6.1.

Loadings on the spent fuel and dryer-separator pools include the effects of water set in motion by seismic accelerations and the thermal gradient resulting from the high temperature of the water in the pools.

The siding and roof deck on the reactor building superstructure are designed to blow off at a specified wind pressure, ensuring that only the stel frame need be designed for tornado loadings.

As noted in 3.8.4.1.1.4, overpressurization of the main steam tunnel in the secondary containment of the reactor building is prevented by means of venting the tunnel to the atmosphere and to the turbine generator building by means of blow out panels in the north end of the tunnel. The blow out panels are designed to blow off at a differential pressure specified in 3.6.1.20. The tunnel is designed to withstand the internal differential pressure arrived at on the basis of the pressure history in the tunnel following a steam line break. For a discussion of this analysis, see 3.6.1.20.

3.8.4.4 Design and Analysis Procedures

Conventional elastic techniques are used in the design and analysis of all structural components. All buildings are analyzed basically as shear wall structures, and all floors are checked for their ability to transmit shear forces through diaphragm action. Exterior walls are designed to resist a combination of vertical loads, bending moments and lateral shear and overturning moments associated with seismic forces (see 3.7.2) and tornado loads. Longitudinal and lateral shears are tranferred to the mat through shear friction reinforcement and keys. The floor slab or beam and column framing is modeled to most closely approximate the actual structural behavior, and all boundary conditions are

ADLPIPE computes the non-mass network force-moments sets for each mode. As seen previously, the network stiffness matrix formed is generated by the transfer matrix of a series of many individual members. This same accumulated transfer matrix is used to compute the force-moment sets at interior points of the piping system (including the mass points).

The cumulative effect of all the modes is estimated by taking the square root of the sum of squares of the force-moment sets at each position in the piping system. For closely spaced frequencies, an option exists which enables the addition of the absolute value of those modal moments and then forming the square of that sum in the square root of square summation.

This program is referred to in 3.9.1.2.2.

3.12.11 RELAP3

This program describes the behavior of water-cooled nuclear react ors during postulated accidents such as loss-of-coolant, pump failure, or power transients. The behavior of the primary cooling system and the reactor is emphasized. The program calculates flaws, mass inventories, energy inventories, pressures, temperatures, and qualities along with variables associated with reactor power, reactor heat transfer, or control systems.

RELAP3 is an NRC accepted computer program and is in the public domain. For a complete discussion of this program see Reference 3.12-18.

This program is referred to in 3.6.2.2.1b and 3.6.2.3.1.

3.12.11.1 RELAP4/MOD5

RELAP4 is a computer program written in FORTRAN IV for the digital computer analysis of nuclear reactors and related systems. It is primarily applied in the study of system transient response to postulated perturbations such as coolant. loop rupture, circulation pump failure, power excursions, etc. The program was written to be used for water-cooled (PWR and BWR) reactors and can be used for scale models such as LOFT and SEMISCALE. Additional versatility extends its usefulness to related applications, such as ice condenser and containment subcompartment analysis. Specific options are available for reflood (FLOOD) analysis and for the NRC Evaluation Model.

RELAP4 models system fluid conditions including flow, pressure, mass inventory, fluid quality, and heat transfer. A subroutine provides water property tables. Component thermal conditions and energy transfers are modeled. The reactor system is subdivided into discrete volumes which, with interconnecting junctions (flow paths), are treated as one dimensional homogeneous elements. RELAP4 solves an integral form of fluid conservation and state equations for each user defined volume and generates a time history of system conditions. Data are recorded for volume fluid, component heat, and junction flow characteristics. The output is in the form of printed tabular digital data. Available subroutines also allow output to be plotted as a function of time. Provision is made for selectively stopping the program at any point for data edits. The program can be restarted for problem continuation.

RELAP4/MOD5 was intended primarily as a blowdown code. It will calculate system phenomena from initial operating conditions at the time of pipe rupture through system decompression up to the beginning of core recovering with emergency core coolant . RELAP4/MOD5 is capable of calculating this core recovering within the limitations of the MOD5 models. These models will not adequately calculate all the reflood phenomena. RELAP4/MOD5 will be designed to address the PWR reflood problem.

The manual is comprised of three volumes. Volume I describes the models included in MOD5. Volume II is directed toward the use of the Code, including application and programming information, and a sample problem. These two volumes are cross-referenced to aid the program user. Volume III presents the results of eight computer runs used in checking out RELAP4/MOD5. These are furnished with interpretation of results for illustration purposes and represent the actual use of RELAP4/MOD5 to investigate real or hypothesized situations. These should not, however, be considered as an in depth study of the reactor plants modeled or of the postulated accidents. The input data for the checkout problems only generally relate to the identified plants.

RELAP4/MOD5 represents a current state of the art calculation method for estimating the transient thermal-hydraulic phenomena in light water reactors and reactor simulators.

RELAP4/MOD5 is an NRC accepted computer program and is in the public domain. For a complete discussion of this program see reference 3.12-25.

This program is referred to in 3.6.1.20.

3.12.12 CB&I PROGRAM 711 "GENOZZ"

The GENOZZ computer program is used by the General Electric Company to proportion barrel and double taper type nozzles of the reactor pressure vessel to comply with the specifications of the ASME Code, Section III and contract documents. The program either designs such a configuration or

3.12-18 Rettig, W. H., et al, "RELAP3: A Computer Program for Reactor Blowdown Analysis", IN-1321, U.S., Atomic Energy Commission Scientific and Technical Report, Reactor Technology TID-4500, Idaho Operations Office, June 1970.

3.12-19 Shen, D.T., "DAPS: A Program for Dynamic Analysis of Piping Systems", General Electric Company, Nuclear Engineering Division, Document No. 383 HA721, October 1972.

3.12-20 Moody, F. J., Time-Dependent Pipe Forces Caused by Blowdown and Flow Stoppage, ASME Paper No. 73-FE-23, presented before the Applied Mechanics and Fluids Engineering Conference, Atlanta, Ga., June 1973.

3.12-21 Hanson, G. H., Subcooled Blowdown Forces on Reactor System Components: Calculational Methods and Experimental Confirmation, Idaho Nuclear Corporation, Report No. IN-1354, June 1970.

3.12-22 Final Report Pipe Rupture Analysis of Recirculation System for 1969 Standard Plant Design, Nuclear Services Corporation, Report No. GEN-02-02, Campbell, California.

3.12-23 AISC, Specification for Design, Fabrication and Erection of Structural Steel for Buildings, American Institute of Steel Construction (1969).

3.12-24 ACI 318-71, "Building Code Requirements for Reinforced Concrete", American Concrete Institute (1971).

3.12-25 Idaho National Engineering Laboratory, RELAP4/MOD5, A Computer Program For Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, Volume I: RELAP4/MOD5 Description, Volume II: Program Implementation, Volume III: Checkout Application, National Technical Information Service, U.S. Department of Commerce, Springfield, Virginia, ANCR-NURGE-1335, September 1976.

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RESPONSES TO

MECHANICAL ENGINEERING BRANCH (MEB)

QUESTIONS 110.001 - 110.037

Q. 110.001
(3.6.1)
(3.6.2)

You indicate in the FSAR that the following portions will be provided at a later date: 3.6.1.6 through 3.6.1.10, 3.6.1.20, and 3.6.2.5.4.4b and 3.6.2.5.4.4c. Additionally, there are about 40 figures in 3.6 of the FSAR which are intended to be summaries of postulated pipe break locations. However, these figures have only a single entry; i.e., "later". Indicate when the missing sections and figures will be submitted.

Response:

Section 3.6.1.6 through 3.6.1.10, 3.6.2.5.4.4b and 3.6.2.5.4.4c are included in this amendment and may be referred to for the procedures used to evaluate the structural adequacy of Seismic Category I structures under pipe break effects outside containment.

The missing figures referred to in the question and in 3.6.2.5.4 are summary tabulations of postulated pipe break locations shown on the piping system isometrics in Figures 3.6-12a through 3.6-34a. It is intended that the missing information in Figures 3.6-16b, 3.6-17b, 3.6-19b, 3.6-26b, 3.6-32b and 3.6-34b will be provided when the final pipe break study is completed.

Section 3.6.1.20 is provided in response to Question 010.011. This response may be referred to for methods used to predict blowdown rates and compartment pressure due to postulated pipe breaks.

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LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

A total of 12 restraints are utilized for the main steam and feedwater lines in the main steam tunnel outside of primary containment. These restraints are illustrated in Figures 3.6-6g and 3.6-6h.

3.6.1.6 Procedures to Evaluate the Structural Adequacy of Seismic Category I Structures Under Pipe Break Effects Outside Containment

3.6.1.6.1 General Approach

Structures and structural components important to safety are designed with sufficient strength to resist the effects of postulated pipe breaks in high energy fluid piping systems, such as pipe whip and jet impingement. High energy fluid piping systems are defined in 3.6.1.1. Section 3.6.1.6 is concerned with the effects of postulated pipe breaks on structures and structural components. Environmental effects of postulated pipe breaks are addressed in 3.6.1.12, 3.6.1.13, 3.6.1.15 and 3.11.

The main component effects of a postulated pipe break include the following:

a. Pipe whip with its impacting energy

b. Jet impingement and accompanying jet reaction

c. Pressurization and temperature effects which accompany pipe break

Pipe whip effects from circumferential breaks are illustrated in Figure 3.6-116. Jet impingement effects from circumferential breaks and longitudinal splits are illustrated in Figure 3.6-117. Circumferential breaks and longitudinal splits are defined in 3.6.2.1.4.

In making a structural evaluation of the effects of pipe break accidents, the loads resulting from these pipe break accidents are used in combination with other prevailing loads that occur at the time of the break. For load information and combinations see 3.6.1.6.5 and 3.6.1.6.6.

In order to make a structural evaluation of the effects of a postulated pipe break, the local damage to the structural element is predicted and the overall structural response is assessed. Local damage is the damage done to a structural element in the immediate vicinity of the pipe whip impact or the jet impingement. Overall structural response concerns the overall response of the entire structural element to the effects of a postulated pipe break. In the following discussion, whipping pipe are described as missiles.

3.6.1.6.2 Local Damage Prediction

Local damage prediction due to whip or jet impingement in the immediate vicinity of the impacted area includes estimation of the depth of penetration and whether, in the case of concrete targets, secondary missiles might be generated by spalling. In general, a whipping pipe is a blunt missile and penetration and spalling are not appreciable for the structural component (e.g., walls) thickness of interest. Such a condition is illustrated in Figure 3.6-116. Jet impingement local damage is not considered significant because the fluid mass does not have the mass concentration of a solid and because of the divergence of a jet which spreads the load over a wide area (see Figure 3.6-117). Missile penetration is predicted for reinforced concrete targets and for steel targets.

3.6.1.6.2.1 Reinforced Concrete Targets

a. Penetration

The depth to which a rigid missile penetrates a reinforced concrete target of infinite thickness is estimated by the following "Modified Petry Formula" (Reference 3.6-15 and 3.6-17):

$$X = K_p A_p \log_{10} \frac{(1 + \cancel{V_s^2)}}{\cancel{215},000} \qquad \text{(Eq. 3.6.1.6.2.1-1)}$$

where: $\left(1 + \frac{V_s^2}{215{,}000}\right)$

X = Depth of missile penetration into concrete element of infinite thickness (feet)

K_p = Penetration coefficient for reinforced concrete (4.76×10^{-3} cubic feet per pound for normal reinforced concrete with a crushing strength of 3,200 psi and 1.4% of reinforcement. Reference 3.6-15.).

$$A_p = \frac{W}{A} = \frac{\text{Missile Weight (lbs)}}{\text{Projected Frontal Missile Area (ft}^2\text{)}}$$

V_s = Striking Velocity of Missile (ft./sec.)

When the element has a finite thickness, the depth of penetration is (Reference 3.6-15 and 3.6-17):

$$X_1 = \left[1 + e^{-4\left(\frac{t}{x} - 2\right)}\right] X, \quad (t \geq 2X) \qquad \text{(Eq. 3.6.1.6.2.1-2)}$$

where:

X_1 = Depth of penetration of missile into a concrete element of finite thickness (feet)

e = Base of Napierian Logarithms

t = Thickness of concrete element (feet)

b. Perforation

The thickness of a concrete element that will be just perforated by a missile is given as (Reference 3.6-15 and 3.6-17):

$$T = 2X \text{ (feet)} \qquad \text{(Eq. 3.6.1.6.2.1-3)}$$

c. Spalling

Spalling of concrete from the side opposite the impact surface of the structural element may occur even if the missile does not perforate the element. The estimate of the thickness that will just start spalling is given as (Reference 3.6-20):

$$T_s = 2.2X \text{ (feet)} \qquad \text{(Eq. 3.6.1.6.2.1-4)}$$

3.6.1.6.2.2 Steel Targets

The Ballistic Research Laboratories formula is used to determine perforation of a steel target. The thickness, T, of a steel target that will be just perforated by a missile is given as (Reference 3.6-17):

$$T^{3/2} = \frac{0.5MV^2}{17,400\ K^2D^{3/2}} \qquad \text{(Eq. 3.6.1.6.2.2-1)}$$

where:

T = Steel wall thickness to just perforate (inches)

M = Mass of the missile (weight/g in $lb\text{-}sec^2/ft$)

V = Velocity of missile (ft/sec)

K = Constant depending on grade of steel and is usually $\cong 1$

D = Diameter of missile (inches). For irregularly shaped missiles, an equivalent diameter is used, taken as the diameter of a circle with the same area as the projected frontal area of the irregularly shaped missile

The recommendation in Reference 3.6-13 to increase the perforation thickness, T, obtained by the Ballistic Research Laboratories Formula by 25% to prevent perforation is observed; that is:

$$t_p = 1.25T \qquad \text{(Eq. 3.6.1.6.2.2-2)}$$

where:

t_p = Thickness of steel barrier required to prevent penetration~~s~~ (inches)

3.6.1.6.3 Overall Structural Response

3.6.1.6.3.1 General

In general, pipe break loads are considered in combination with other loads (see 3.6.1.6.6). Dead loads, live loads, operating thermal loads and earthquake loads may or may not be significant compared to the pipe break load, depending on the severity of the pipe break load. Thermal loadings due to pipe break have only skin effect and are not considered.

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 Bechtal 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.

b. Velocity After Impact

The velocities of the missile and target after impact are calculated from the following relationships (Reference 3.6-19):

$$V_m = \frac{V_s (M_m - eM_e)}{M_m + M_e} \qquad \text{(Eq. 3.6.1.6.3.2-1)}$$

$$V_t = \frac{V_s M_m (1+e)}{M_m + M_e} \qquad \text{(Eq. 3.6.1.6.3.2-2)}$$

where:

V_m = Missile velocity after impact (ft./sec.)

V_T = Target velocity after impact (ft./sec.)

V_s = Missile striking velocity (obtained by using basic velocity formulas, knowing the initial thrust force, missile mass, and missile travel before impact) (ft./sec.)

M_m = Mass of missile (lbs.-sec.^{2}/ft.)

M_e = Effective mass of target during impact (lbs.-sec.^{2}/ft.)

e = Coefficient of restitution

c. Plastic Impact

In as plastic impact, the coefficient of restitution becomes zero, and the velocity of the missile and target masses become equal following impact. The strain energy, E_s, required to stop the missile/target combination is the summation of the missile mass kinetic energy and the target mass kinetic energy at the end of the impact duration, as follows:

$$E_s = \frac{M_m V_m^2}{2} + \frac{M_e V_T^2}{2} \qquad \text{(Eq. 3.6.1.6.3.2-3)}$$

From Equations 3.6.1.6.3.2-1 and 3.6.1.6.3.2-2:

$$V_m = V_T = \frac{M_m V_s}{M_m + M_e} \qquad \text{(Eq. 3.6.1.6.3.2-4)}$$

Substituting the value for V_m and V_T from Equation 3.6.1.6.3.2-4 into Equation 3.6.1.6.3.2-3, the required target strain energy is:

$$E_s = \frac{M_m{}^2\, V_s{}^2}{2(M_m + M_e)} \qquad \text{(Eq. 3.6.1.6.3.2-5)}$$

d. Target Effective Mass

Due to the complexity of missile-target impact, a determination of an effective coupled mass on a continuous time basis by means of a general analytical solution is not available. However, an estimate of the average effective mass can be approximated from the results of impact tests on reinforced concrete beams (Reference 3.6-10) in which the measured structural response is used to back-calculate the average mass during impact. Based on these data, the following formulae are used for estimating the target effective mass.

For concrete beams:

$$M_e = (D_x + 2T)\, \frac{B\gamma_c T}{g}, \quad \text{if } B \leq (D_y + 2T)$$

(Eq. 3.6.1.6.3.2-6a)

$$M_e = (D_x + 2T)\,(D_y + 2T)\, \frac{\gamma_c T}{g}, \quad \text{if } B \geq (D_y + 2T)$$

(Eq. 3.6.1.6.3.2-6b)

For concrete slabs:

$$M_e = (D_x + T)\,(D_y + T)\, \frac{\gamma_c T}{g} \qquad \text{(Eq. 3.6.1.6.3.2-7)}$$

For steel beams:

$$M_e = (D_x + 2d)\, M_x \qquad \text{(Eq. 3.6.1.6.3.2-8)}$$

For steel plates:

$$M_e = D_x D_y\, \frac{\gamma_s t}{g} \qquad \text{(Eq. 3.6.1.6.3.2-9)}$$

where:

M_e = Average effective mass of target during impact $\frac{(lbs.-sec.^2)}{ft.}$

M_x = Mass per unit length of steel beam $\frac{(lbs.-sec.^2)}{ft.^2}$

D_x = Maximum missile contact dimension in the x direction (longitudinal axis for beams or slabs) (ft.)

D_y = Maximum missile contact dimension in the y direction (transverse to longitudinal axis for beams or slabs) (ft.)

T = Thickness or depth of concrete element (ft.)

t = Thickness of steel plate (ft.)

d = Depth of steel beam (ft.)

B = Width of concrete beam (not to exceed $D_y + 2T$)(ft.)

γ_c = Weight per unit volume of concrete (lbs./cu. ft.)

γ_s = Weight per unit volume of steel (lbs./cu. ft.)

g = Acceleration of gravity (32.2 ft./sec.^{2})

c. Structural Response by Energy Balance Method

(1) General Procedures

The strain energy, E_s, required to stop the target (or missile-target combination) is determined from the relationships in 3.6.1.6.3.2.

The resistance-displacement function, R(x), for a concentrated load at the area of impact is determined from the target structure physical configuration and material properties.

The estimated maximum target response is determined by equating the available target strain energy to the required strain energy and solving for the maximum displacement, x_m, (see Figure 3.6-120.)

(2) Elasto-Plastic Target Response

For elasto-plastic target response with no other concurrent loads acting:

$$R(x) = Kx, \quad (0 < x \leq x_e)$$

$$R(x) = Kx_e = R_m, \quad (x_e < x \leq x_m)$$

where:

R = Resisting force of target (lbs.)

x = Displacement of target (ft.)

k = Elastic Spring constant for target (lbs./ft.)

x_e = Yield displacement (ft.) (Reference Tables 3.6-9, 3.6-10, Figure 3.6-120)

R_m = Plastic resistance (lbs.) (Reference Tables 3.6-9, 3.6-10, Figure 3.6-120)

x_m = Maximum displacement of target (ft.)

then:

$$E_s = R_m \left(x_m - \frac{x_e}{2}\right)$$

or:

$$x_m = \frac{E_s}{R_m} + \frac{x_e}{2} \qquad \text{(Eq. 3.6.1.6.3.2-10)}$$

The required ductility ratio, μ_r, is obtained from Equation 3.6.1.6.3.2-10 by dividing both sides of the equation by x_e.

$$\mu_r = \frac{x_m}{x_e}$$

$$\mu_r = \frac{E_s}{x_e R_m} + 1/2 \qquad \text{(Eq. 3.6.1.6.3.2-11)}$$

If other loads are present on the target structure which act concurrent with missile impact loads, (see 3.6.1.6.5, 3.6.1.6.6 and Table 3.6-11), the maximum combined displacement is determined as follows:

Let:

$x' = x_e - x_o$ (see Figure 3.6-120) (ft.)

x_o = Displacement due to other loads (ft.)

x_e = Yield displacement (ft.)

x_m = Maximum combined displacement (ft.)

R_m = Plastic resisting force (lbs.)

k = Elastic spring constant (lbs./ft.)

Then:

$$E_s = \frac{k(x')^2}{2} + kx' (x_m - x_e)$$

(See Figure 3.6-120)

or:

$$x_m = \frac{E_s}{kx'} - \frac{x'}{2} + x_e$$

Substituting $x' = x_e - x_o$ in the above equation gives:

$$x_m = \frac{E_s}{k (x_e - x_o)} + \frac{x_e + x_o}{2}$$

(Eq. 3.6.1.6.3.2-12)

The required ductility ratio, μ_r, is obtained by dividing both sides of Equation 3.6.1.6.3.2-12 by x_e.

$$\mu_r = \frac{Es}{R_m (x_e - x_o)} + \frac{1 + x_o/x_e}{2}$$

(Eq. 3.6.1.6.3.2-13)

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

with the strength design methods described in ACI 318-71 (Reference 3.6-12) and in the general practices of Part 2 of the AISC specifications (Reference 3.6-11), modified by the dynamic strength increase factors of Table 3.6-8.

The local damage is considered acceptable if the pipe whip impact does not cause spalling and excessive penetration in concrete, or perforation in steel, as determined by the procedures set forth in 3.6.1.6.2.

3.6.1.6.4.2 Pipe Break Loads (Excluding Pipe Whip) With or Without Other Loads

Pipe break loads (excluding pipe whip) with or without other loads are considered acceptable if the loading from the loading combinations in 3.6.1.6.6 does not result in stresses that exceed the allowable limits on section strength as given in Tables 3.8-12 and 3.8-17, modified by the dynamic strength increase factors in Table 3.6-8.

3.6.1.6.5 Loads, Definition of Terms and Nomenclature

For loads, definition of terms and nomenclature, see 3.8.4.3.

3.6.1.6.6 Load Combinations

3.6.1.6.6.1 Seismic Category I Concrete Structures

For load combinations for Seismic Category I concrete structures, see Table 3.8-15, load combinations 6, 7, and 8.

3.6.1.6.6.2 Seismic Category I Steel Structures

For load combinations for Seismic Category I steel structures, see Table 3.8-16, load combinations 6, 7 and 8.

3.6.1.7 Structural Design Loads

Structural elements are designed to withstand the loads generated by piping failures outside of primary containment in combination with other loads given in 3.6.1.6.6. Table 3.6-11 furnishes the design loads considered in the areas where piping failures occur.

3.6.1.8 Analysis of Load Reversal

Structural elements such as floors, interior walls, exterior walls and the building as a whole are analyzed for the effects of reversal of load due to the postulated pipe failure accident. They are also analyzed for rebound loads that accompany pipe break accidents. The analysis approach for rebound is set forth in Figure 3.6-122.

3.6.1.9 Modified Structures

New Openings or other modifications are not planned to be provided in existing structures, and, therefore, the capabilities of structures to carry the design loads due to modification need not be demonstrated.

3.6.1.10 Verification That Failure of Any Structure Does Not Preclude Safe Reactor Shutdown

Structures subjected to pipe whip and/or jet impingement loads are investigated and found not to fail under these loads in conjunction with the applicable load combinations, so that there are not cases of structural barriers failing and causing additional structural failures which would adversely affect the mitigation of the consequences of accidents and the capability to bring the plant to a cold shutdown condition.

3.6.1.11 Verification That Adequate Redundancy Exists for All Postulated Fluid Piping System Ruptures

3.6.1.11.1 Approach

The purpose of the study is to ensure that for all postulated ruptures of fluid piping systems, safe reactor operation and shutdown is not precluded. The basis of this approach is that adequate separation of redundant systems or components, required to shutdown and maintain the reactor in a cold condition, provides the level of protection required to ensure safe reactor operation and shutdown.

The input used for this study includes the routing of all cables, cable trays and conduit necessary to shutdown and maintain the reactor in a cold condition. The locations of all motor control centers, instrument racks, sensors and heating ventilation and air conditioning (HVAC) equipment necessary to shutdown and maintain the reactor in a cold condition are also included in the input of this study.

The locations of all postulated high and moderate energy fluid piping system ruptures dictate where this study is to be performed.

The input described above is coded to indicate: the location of the system or component by elevations and grid[a]; the

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(a) The reactor building is subdivided into 42 grids each measuring approximately 20' x 20'. This permits rapid location of any component on the floor plan.

Figure 3.6-37 illustrates the locations of the grids.

3.6.2.1.1.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Runs

a. The terminal ends of the pressurized portions of the run.

b. Intermediate locations of postulated pipe breaks are selected by applications 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 for valves and flanges.

(2) At each location where the stresses under the loadings resulting from upset plant conditions, including an OBE event as calculated by the summation of Equations (9) and (10) of ASME Code Section III Subsection NC, Paragraph NC 3652, exceed 0.8 $(1.2S_h + S_A)$ where S_h and S_A), are as defined in Paragraph NC 36.11.2.

(3) If there are not at least two intermediate locations, where the above noted stresses exceed 0.8 $(1.2S_h + S_A)$, a minimum of two separate locations are chosen based upon stress, except if the piping run has only one change of direction, a minimum of one intermediate break is postulated.

(4) Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, valves, or flanges.

3.6.2.1.1.3 Break Locations in Other Piping Runs

Postulated pipe break locations for piping other than ASME Code Section III Class 1, 2 and 3, are postulated in accordance with pipe whip criteria which generally conforms to the criteria set forth for ASME Code Section III Class 2 and 3 piping.

3.6.2.1.2 Postulated Pipe ~~Brak~~ Break Locations in High Energy Fluid System Piping Between Primary Containment Isolation Valves

Pipe breaks (not including leakage cracks) are postulated in locations as indicated below:

3.6.2.1.2.1 Postulated Pipe Break Locations in ASME Section III Class I Piping Between Primary Containment Isolation Valves

No pipe breaks are postulated in the portion of piping between primary containment isolation valves, if any of the following apply:

(1) S_n does not exceed $2.4S_m$.

(2) S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U) does not exceed 0.1.

(3) S_n exceeds $3S_m$, but S_e and S_r are each less than $2.4S_m$, and U does not exceed 0.1.

The stress levels in the ASME Section III Class I containment penetration high energy piping are maintained at or below these limits and therefore, breaks are not postulated. See 3.6.2.1.2.3 for further discussion of containment penetration piping.

3.6.2.1.2.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Between Primary Containment Isolation Valves

See 3.6.2.1.1.2 b. (2) for stress criteria applicable to ASME Section III Class 2 and 3 piping between containment isolation valves.

The stress levels are maintained at or below these limits and therefore breaks are not postulated. See 3.6.2.1.2.3 for further discussion of containment penetration piping.

3.6.2.1.2.3 Primary Containment Penetration Piping

Primary containment penetrations, in order to maintain containment integrity, are designed with the following characteristics:

a. They are capable of withstanding the forces caused by impingement of the fluid from the rupture of the largest local pipe without failure.

b. They are capable of withstanding the maximum reactions that the pipes to which they are attached are capable of exerting.

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.

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.

For systems or parts thereof, that are necessary for a safe shutdown but cannot be protected by redundancy, a detailed analysis is performed to determine jet impingement effects on the operability of these systems. Barriers are provided where necessary.

3.6.2.3.2.3 Postulated Pipe Rupture Locations Inside Containment

The criteria used to define pipe rupture locations is described in 3.6.2.1 and is shown in Figures 3.6-12a through 3.6-17a, 3.6-18a, 3.6-18b, 3.6-19a through 3.6-34a and 3.6-35.

3.6.2.3.2.4 Signals from Primary Containment

For instrumentation located inside primary containment, sufficient redundancy is provided such that all signals necessary to cause actuation of essential systems, remain functional. Each system, that is required to bring the plant to a safe shutdown condition, is furnished with two or more sets of redundant instrumentation lines.

In this review, it is conservatively assumed that a jet stream or whipping pipe may damage one of these sets. The redundant system is shown to remain operational by physical separation and barriers, such as the RPV. An example of the above is the location of Sets "A" and "B" instrumentation lines for the HPCS. Set "A" and its redundant Set "B" are located at opposite sides of the RPV. Therefore, a jet stream or whipping pipe cannot damage both sets of instrumentation. Function of instrumentation inside primary containment necessary to result in the actuation of the HPCS system is thereby assured. These conditions, as discussed for the HPCS instrumentation lines, are typical for all instrumentation lines that support essential systems. The capabilities of redundant instrumentation is discussed in 7.3.

3.6.2.3.2.5 Signals to the Primary Containment

No instrumentation signal is necessary to return inside primary containment to operate any of the essential systems. Signals to the ADS valves are provided through their power supply as described in the following section.

3.6.2.3.2.6 Power Requirement Inside Primary Containment

The only essential system that required power, inside primary containment, is the automatic depressurization system (ADS).

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):

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.

Figures 3.6-12a through 3.6-35 show the piping configurations for each high energy system inside primary containment and include numerical indentification 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 3.6-34a and as noted on Figure 3.6-35.

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 none control rod drive (CRD) withdrawal lines may be completely crimped (totally blocked). Complete serverence of withdrawal lines does not affect the rod insert function. Protection of

the CRD insert lines is not required, since a reactor pressure of 450 psig or higher, can adequately insert the control rods.

ment with mirror image symmetry about the 0° and 180° north-south azimuth. The lines exit the reactor pressure vessel on opposite sides of primary containment and drop down vertically in two parallel pairs to the main steam relief valve platform at elevation 541 ft. where they are routed horizontally, in parallel, in the northeast and northwest quadrants to the 0° north azimuth. At this point, the four lines drop vertically in parallel, to an elevation just above the diaphragm floor. The main steam isolation valves are located here. The four lines exit the containment nearest the north azimuth at elevation 500 ft. (approx.). The two feedwater piping loops are described in 3.6.2.5.4.2 and are routed near the main steam lines.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the four main steam lines, are shown in Figures 3.6-12a through 3.6-15a. Where pipe breaks are postulated inside primary containment, the main steam lines are restrained to prevent the unacceptable motion of these pipes. These restraints are mounted on the side of the sacrificial shield wall structure, as well as on radial beams which extend from the sacrificial shield wall to the primary containment vessel wall. A sliding beam seat at the primary containment wall, permits the beam to grow axially and also permits the primary containment wall to move relative to the sacrificial shield wall.

A structural steel frame (see Figures 3.6-36a, 3.6-36b, and 3.6-36c) between the drywall diaphragm floor and the containment vessel, in the area of the main steam isolation valves, is provided for mounting of pipe whip restraints. The structure is designed with vertically sliding connections at the containment vessel, to allow for differential thermal expansion between the containment vessel and the diaphragm floor.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam system to assure safety as defined

in 3.6.2.5.2. The pipe whip restraints limit the pipe whip motion of the main steam lines to prevent impact and rupture of the adjacent feedwater lines which would otherwise result in a break area in excess of the ECCS capability.

Impact with the control rod drive piping is prevented by pipe whip restraints at the main steam relief valve platform and separation. The control rod drive piping bundles are routed below the elevation 541 ft. main steam relief valve platform, a considerable distance away from where the main steam lines drop down to the diaphragm floor.

3.6.2.5.4.2 Reactor Feedwater System (Inside Primary Containment)

a. System Arrangement

The reactor feedwater system inside primary containment, consists of two piping loops symmetrically arranged with respect to 0° and 180° north-south azimuth. The two piping loops emerge from each side of the reactor as three 12 inch vertical risers which drop down and join a header at the main steam relief valve platform. The header is routed paralled to, and outside of, the main steam lines, increasing in diameter from 12 inches, to 18 inches and to 24 inches as it approaches the 0° north azimuth. At this location, the two 24-inch feedwater pipes drop down to 12'-6" above the diaphragm floor. The pipe is furnished with a check valve in each line in the short horizontal run near the primary containment vessel penetration.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for both reactor feedwater loops are shown in Figures 3.6-16a and 3.6-17a. The feedwater lines are restrained to provide protection from the results of all postulated pipe breaks. Specifically, protection is provided where the resulting pipe motion would otherwise impact

equipment necessary to mitigate the consequences of the break, causing unacceptable damage to that equipment. The restraints are mounted on the side of the sacrificial shield wall, on radial beams at the elevation 541 ft. main steam relief valve platform and on a specially designed structure between the containment and diaphragm floor, as shown in Figures 3.6-36a, 3.6-36b and 3.6-36c. Special features of these structures are described in 3.6.2.5.4.1(b).

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the reactor feedwater system to assure safety as defined in 3.6.2.5.2.

In all cases the pipe is sufficiently restrained to prevent impact with containment or impact with other piping systems that would result in violation of pipe break area or pipe break combination limitations. Impact with the control rod drive piping is prevented by pipe whip restraints at the main steam relief valve platform and separation. The control rod drive piping bundles are routed below the elevation 541 ft. main steam relief valve platform, at a considerable distance away from the 0° north azimuth where the 24 inch feedwater lines drop to 12'-6" above the diaphragm floor. In two cases, portions of the 12 inch vertical risers are restrained in only one direction, allowing impact with one main steam relief valve. This constitutes acceptable damage because depressurization can be accomplished with one valve not functioning. Furthermore, the HPCS is available as a redundant system.

3.6.2.5.4.3 Reactor Water Cleanup System (RWCU)

a. System Arrangement

The RWCU system consists of two 4-inch lines which branch from the two reactor recirculation cooling (RRC) pump suction lines located near the 0° and 180° north and south azimuths. The two lines are routed along the diaphragm floor at approximate elevation 500 ft. to azimuth 67° where they join into one, 6-inch pipe. This 6-inch pipe branches off into two segments. One branch rises to elevation 538 ft. just below the main steam relief valve platform. It is then routed to azimuth 150°, where it exits primary containment. An isolation valve is located inside primary containment near the penetrations. The other 6-inch segment reduces to a 4-inch pipe and then rises to elevation 514 ft. and terminates at the 2-inch RPV drain system.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the RWCU system are shown in Figures 3.6-18a and 3.6-18b. At all locations where pipe breaks are postulated inside primary containment, the RWCU system is restrained to prevent unacceptable motion of the pipe. Where the pipe is routed along the diaphragm floor, restraints are mounted on special structures built up from the floor. Where the pipe is routed below main steam relief valve platform, restraints are mounted on intermediate structures between radial beams.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RWCU system to assure safety as defined in 3.6.2.5.2. Pipe whip restraints located above the diaphragm floor are designed to prevent impact with the floor, which might impair steam quenching capability of the suppression pool. Pipe whip restraints located directly below the main steam relief valve platform prevent impact with CRD piping and also primary containment. Equipment necessary to mitigate RWCU pipe breaks, such as ADS system, core spray, low pressure core injection, is protected by separation.

3.6.2.5.4.4 Standby Liquid Control (SLC) Piping

a. System Arrangement

The SLC system consists of 1-1/2-inch piping that originates at the bottom of the reactor pressure vessel and is routed through the reactor pedestal. Immediately outside the pedestal, there is a normally closed check valve which limits the high energy portion of this system to the area inside the reactor pedestal.

b. Pipe Whip Protection

The postulated pipe breaks for the SLC system are shown in Figure 3.6-19a. Pipe whip restraints are not required for this system.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the SLC system to assure safety as defined in 3.6.2.5.2. In the event of a pipe whip resulting from a pipe rupture at any postulated location, the piping system neither impacts the primary containment vessel nor damages equipment or systems required for safe shutdown of the reactor. Therefore, pipe restraints are not required for this system.

3.6.2.5.4.5 Residual Heat Removal System (RHR) - Shutdown Cooling Supply and Return Piping

a. System Arrangement

The RHR shutdown cooling supply and return piping consists of two, 12-inch piping loops and one, 20-inch loop, with all three branching from the RRC piping at the elevation 512 ft. platform. All three loops are routed primarily in a horizontal plane, below the 512 ft. platform, from the RRC pipe to its primary containment penetration. There is a normally closed valve in each loop located as close as possible to the high energy source, thereby limiting the portion of each loop considered high energy on the basis defined in 3.6.2.1.

b. Pipe Whip Protection

The pipe whip restraints for the RHR shutdown cooling supply and return system are shown in Figures 3.6-23a, 3.6-24a and 3.6-25a. Where pipe breaks are postulated inside primary containment, the lines are restrained to prevent the unacceptable motion of these pipes. For the two, 12-inch shutdown cooling return loops, restraints are mounted on intermediate structures between the radial beams in the elevation 512 ft. platform which radial beams extend from the reactor pedestal to the primary containment wall. A sliding beam seat, at the primary containment wall, permits differential thermal expansion between the containment vessel and reactor pedestal. Restraints

for the 20-inch cooling return loop are mounted on a specially designed structure between the diaphragm floor and radial beams in the elevation 512 ft. platform, as shown in Figure 3.6-10.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR shutdown cooling supply and return system to assure safety as defined in 3.6.2.5.2.

For the two, 12-inch shutdown cooling return loops, pipe whip restraints are provided to prevent impact with primary containment wall and the diaphragm floor. The pipe whip restraints also prevent impact with the CRD piping bundles located above the elevation 512 ft. platform. The ECCS system and the ADS systems are protected by separation, being located at higher elevations.

For unrestrained sections of this system, analysis shows a plastic hinge does not develop at the recirculation pipe, and pipe whip does not occur.

For the 20-inch shutdown cooling supply loop, pipe whip restraints are provided to prevent impact with primary containment and the diaphragm floor. Impact with the CRD piping is precluded by a 90° separation from both CRD piping bundles.

3.6.2.5.4.6 RCIC RPV Head Spray System

a. System Arrangement

The RPV head spray system is a 6-inch line that originates at the top of the RPV dome. After a 2 ft. vertical riser and a 2 ft. horizontal run, there is a normally closed valve that limits the high energy portion of this system to a total length of 4 feet.

b. Pipe Whip Protection

The postulated pipe breaks for this system are shown in Figure 3.6-26a. Pipe whip restraints are not required for this system.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided to assure safety as defined in 3.6.2.5.2. The location of the normally closed valve limits the high energy section of this system such that the unrestrained motion of the pipe resulting from postulated breaks can only impact the reactor vessel head.

3.6.2.5.4.7 Low Pressure and High Pressure Core Spray (LPCS, HPCS) Piping

a. System Arrangement

The LPCS and HPCS are 12-inch piping systems with similar arrangements inside primary containment. They originate at elevation 561 ft. from the reactor at azimuths 120° and 240° respectively, and drop vertically to an elevation just below the main steam relief valve platform where, there is an expansion loop in a horizontal plane leading to a penetration through primary containment. In the vertical section, there is a normally closed check valve located as close as possible to the reactor, thereby limiting the portion of piping in both systems considered high energy under the definition given in 3.6.2.1.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the LPCS and HPCS systems are shown in Figures 3.6-27a and 3.6-28a. Where pipe breaks are postulated inside primary containment the two lines are restrained to prevent the unacceptable motion of these pipes. These restraints are mounted directly onto the sacrificial shield wall.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the LPCS and HPCS systems to assure safety as defind in 3.6.2.5.2. Pipe whip restraints are

provided to limit pipe movement resulting from postulated pipe breaks to prevent impact with primary containment and adjacent RHR/LPCI piping. Impact on safety relief valves, resulting from postulated pipe breaks in the HPCS system, is precluded by sufficient separation between these two redundant depressurization methods. The CRD piping bundles are separated by sufficient distance from the high energy sections of the LPCS and HPCS systems.

3.6.2.5.4.8 RHR Condensing Mode and RCIC Turbine Steam Supply System

a. System Arrangement

The RHR condensing mode system consists of a 10-inch piping loop which branches off a main steam line at elevation 551'-2 1/4" and azimuth 105°. An expansion loop in the horizontal plane leads to a penetration through primary containment at elevation 550 ft and azimuth 120°.

The RCIC turbine steam supply system consists of a 4-inch line which branches off the 10-inch RHR condensing mode line at approximately azimuth 125° and drops down to elevation 532 ft below the main steam relief valve platform. The line is then routed horizontally to a penetration through primary containment at azimuth 35°.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the RHR condensing mode and RCIC turbine steam supply systems are shown in Figure 3.6-29a. Where pipe breaks are postulated inside primary containment, this piping is restrained to prevent unacceptable motion of the piping. The restraints for these two systems are mounted on specially designed structures which tie into the sacrificial shield wall and/or radial beams of the main steam relief valve platform.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR condensing mode and RCIC turbine steam supply systems to assure safety as defined in 3.6.2.5.2. For the 10-inch RHR condensing mode system, pipe whip restraints are provided at all locations where pipe breaks are postulated. The pipe whip restraints limit pipe motion resulting from postulated break to prevent impact with primary containment vessel wall, the HPCS system and the main steam safety relief valves. Protection is required, since either the ADS or the HPCS are required to depressurize the reactor subsequent to a pipe break in a line with cross-section area less than 0.7 $ft.^2$ (See 3.6.2.5.3).

For the 4-inch RCIC turbine steam supply, restraints are provided for the portion above the main steam relief valve platform to protect containment and the HPCS and safety relief valves. For the section of this system below the main steam relief valve platform, the pipe movement resulting from postulated breaks will move radially inward impacting the sacrificial shield or vertically down impacting the elevation 512 ft. platform. The CRD piping bundle in this area is located above this line precluding impact.

3.6.2.5.4.9 Main Steam Valve Drainage Piping

a. System Arrangement

The main steam valve drainage piping consists of four, 2-inch pipe lines, each originating at the bottom of the four main steam isolation valves inside primary containment. The four lines are routed above the diaphragm floor joining into one, 3-inch line which then exits containment. Isolation valves are located just inside and just outside of the primary containment protection.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for this system are shown in Figure

3.6-30a. Where pipe breaks are postulated the system is restrained to prevent unacceptable motion of the main steam valve drainage piping. A number of the pipe whip restraints for this system are mounted on specially designed structures built up from the diaphragm floor. The remaining restraints are attached to the structure between primary containment and the diaphragm floor (See 3.6.2.5.4.1b), which has been designed to support the main steam and reactor feedwater pipe whip restraints.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam valve drainage piping to assure safety as defined in 3.6.2.5.2. Pipe whip restraints are provided for this system to protect primary containment structure and the diaphragm floor. Other required safety systems are protected by separation, by being located at considerably higher elevations.

3.6.2.5.4.10 Main Steam RPV Head Vent System

a. System Arrangement

The RPV head vent system consists of a 2-inch line which originates at the top of the RPV dome and is routed through the primary containment bulkhead plate at azimuth 237°. The line is then routed below the bulkhead plate, to azimuth 70° where it drops down to elevation 570 ft. and joins a 26-inch main steam line.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for this system are shown in Figure 3.6-31a. For the piping section above the bulkhead plate, the pipe whip restraints are mounted onto a removeable lattice framework. For the portion of this line below the primary containment bulkhead plate, the restraints are mounted on structures, which tie into the stiffening beams for the bulkhead plate.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RPV head vent piping to assure safety as defined in 3.6.2.5.2. There are no safety related systems in the vicinity of the RPV head vent piping and pipe whip restraints are provided to protect the primary containment structure.

3.6.2.5.4.11 Main Steam and Reactor Feedwater Piping Inside Main Steam Tunnel

a. System Arrangement

The four, 26-inch main steam and two, 24-inch reactor feedwater lines inside the main steam tunnel originate at the primary containment penetrations and run horizontally to the end of the tunnel. At this point, the six lines drop vertically and are then routed horizontally within the turbine generator building. An isolation valve is located in each line just beyond the penetration.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the main steam and reactor feedwater lines inside main steam tunnel, are shown in Figures 3.6-33a and 3.6-34a. Where breaks are postulated, the six lines are restrained to prevent unacceptable motion. The restraints are mounted on steel structures which then tie into the concrete walls and floors.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the main steam and reactor feedwater lines inside the main steam tunnel to assure safety as defined in 3.6.2.5.2.

The basis for providing protection in this area is to prevent pipe whip impact with adjacent isolation valves and to prevent pipe break damage escalation. The six lines and the six isolation valves in this area are located in close proximity to each other. A pipe break in one of

the six lines, if unrestrained, may result in pipe whip impact with adjacent isolation valves, possibly rendering them inoperative. Futhermore, unrestrained motion may cause impact with other lines, which may result in escalation of pipe breaks. Such a condition may unacceptably increase the severity of the initial pipe break.

3.6.2.5.4.12 Residual Heat Removal System (RHR) - Low Pressure Core Injection

a. System Arrangement

The RHR/LPCI piping consists of three, 14-inch loops whose arrangement is the same for two loops with the third loop being the mirror image of the other two. The piping originates at the reactor vessel at elevation 552 ft., rises vertically to elevation 563 ft. where there is a horizontal section with a check valve. This valve is normally closed, limiting the high energy portion of each loop. After the valve, the normally unpressurized section of piping drops to an elevation just below the main steam relief valve platform where it is routed to a penetration through primary containment at elevation 534 ft.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraints for the three RHR/LPCI mode piping loops are shown in Figures 3.6-20a, 3.6-21a and 3.6-22a. Where pipe breaks are postulated, the three piping loops are restrained to prevent unacceptable motion. The restraints for this system are mounted onto the sacrificial shield wall and also on structures which tie back to the sacrificial shield wall.

c. Verification of Pipe Whip Protection Adequacy

Sufficient pipe whip protection is provided for the RHR/LPCI mode piping to assure safety as defined in 3.6.2.5.2. The pipe whip restraints

Limit pipe motion resulting from postulated breaks to preclude impact with primary containment and adjacent feedwater or core spray piping. Impact with adjacent feedwater or core spray piping may result in pipe break escalation that can exceed limitations of pipe break area and pipe break combination. The CRD piping bundles are separated by a considerable distance from high energy sections of the RHR/LPCI mode piping.

3.6.2.5.4.13 RPV Drain System

a. System Arrangement

The RPV drain system is a 2-inch line that originates at the bottom of the reactor pressure vessel and is routed inside the peedestal to a sleeve which leads through the pedestal. Outside the reactor pedestal; the line then joins the RWCU system.

b. Pipe Whip Protection

The postulated pipe breaks and pipe whip restraint for the RPV drain system are shown in Figure 3.6-32a. At postulated pipe break locations inside primary containment, the RPV drain system is restrained to prevent unacceptable motion of the pipe. This system contains only one pipe whip restraint. Where the pipe is routed along the platform at elevation 512'-8", the pipe whip restraint is mounted on a transverse beam which is welded to the top of the radial platform beams.

c. Verification of Pipe Whip Protecton Adequacy

Sufficient pipe whip protection is provided for the RPV drain system to assure safety as defined in 3.6.2.5.2. The single pipe whip support for the RPV drain system serves the dual purpose of providing pipe whip protection and seismic restraint. The pipe whip restraint is located above the platform at elevation 512'-8", and is designed to prevent impact with the Quality Class I electrical conduits in the immediate vicinity of the RPV drain line. Since an annular clearance of only 1/16-inch is maintained between the pipe and the pipe whip support, the pipe whip

support is also utilized as a rigid three-way support.

3.6.2.5.4.14 Reactor Recirculation Cooling System

a. System Arrangement

The recirculation piping consists of the pump discharge and suction piping systems. The recirculation pump "A" and "B" discharge lines are arranged with mirror image symmetry, in the northern and southern segments of primary containment. 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 elevation 536'-1 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 nozzles of the recirculation pumps.

The recirculation pump "A" and "B" suction lines consits of two mirror image systems oriented along the 0° and 180° azimuths with respect to the reactor pressure vessel. Each loop 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 nozzles of the recirculation pumps.

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 3.6-35. 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 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.

3.6-11 AISC, "Specification for Design, Fabarication and Erection of Structural Steel for Buildings", American Institute of Steel Construction, New York, N.Y., February 12, 1969.

3.6-12 ACI 318-71, "Building Code Requirements for Reinforced Concrete", American Concrete Institute, Detroit, Michigan, 1971.

3.6-13 Linderman, R. B., Rotz, J. V., Yeh, G. C. K., Design of Structures for Missile Impact, Bechtel Power Corporation, Topical Report BC-TOP-9A, Revision 2, San Francisco, California, September 1974.

3.6-14 Roark, R. J., Formulas for Stress and Strain, Fourth Edition, McGraw-Hill Book Company, Inc., New York, 1965.

3.6-15 Amarikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, Report NP-3726, August 1950.

3.6-16 Moody, F. J., Maximum Flow Rate of a Single Component, Two-Phase Mixture, American Society of Mechanical Engineers Journal of Heat Transfer, pp. 134-142, February 1965.

3.6-17 Gwaltney, R. C., Missile Generation and Protection in Light-Water Cooled Power Reactor Plants, ORNL NSIC-22, Oak Ridge National Laboratory, Oak Ridge Tennessee, for the U. S. Atomic Energy Commission, September, 1968.

3.6-18 Deleted. Replaced by 3.6-20 below.

3.6-19 Harris and Crede, Shock and Vibration Handbook, McGraw Hill Book Company, Inc., 1961.

3.6-20 Kennedy, R. P., A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects, Nuclear System Sciences Group, Holmes & Narver Inc., Anaheim, California, September 1975.

TABLE 3.6-1

MAXIMUM DUCTILITY RATIOS

STEEL STRUCTURAL COMPONENTS

Steel Beams (Lateral Load) (Note: To develop this ductility, the flanges must be thick enough to prevent local plastic buckling).	26
Steel Beams (Lateral and Axial Load)	8
Welded Portal Frames (Vertical Load)	6-16

REINFORCED CONCRETE STRUCTURAL COMPONENTS

Tension reinforced concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p^*} \leq 10$
Doubly reinforced concrete beams and slabs, (flexure controls design)	$\frac{0.10}{p-p'^{**}} \leq 10$
Reinforced concrete columns, walls and other elements exhibiting brittle fracture, (compression controls design)	1.3

*p is the ratio of tensile reinforcement and must satisfy the limitations:

$$0.0025 \leq p = \frac{As}{bd} \leq 0.015$$

**p' is the ratio of compression reinforcement and must satisfy the limitations:

$$p' = \frac{A's}{bd} \geq 0.0025$$

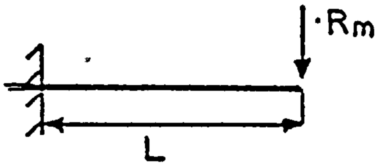
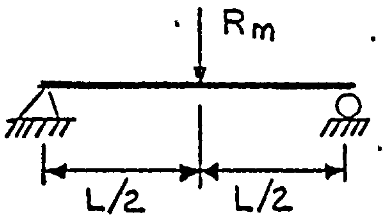
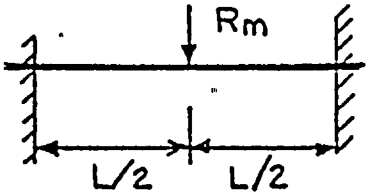
TABLE 3.6-8

DYNAMIC STRENGTH OF MATERIALS

	Dynamic Increase Factor (DIF)
1. Reinforced Concrete	
Concrete	
Compression, axial or flexural	1.25
Shear as a measure of diagonal tension and punching shear	1.00
Bond	1.00
Reinforcing Steel	
Tension	1.10
Compression	1.10
Shear reinforcement to resist shear as a measure of diagonal tension and punching shear	1.00
2. Structural Steel	
Flexure and tension	1.10
Compression	1.10
Shear	1.00

TABLE 3.6-9

RESISTANCE-YIELD DISPLACEMENT VALUES FOR BEAMS

DESCRIPTION	RESISTANCE	YIELD DISPLACEMENT
(1) CANTILEVER 	$R_m = \frac{M_u}{L}$	$X_e = \frac{R_m L^3}{3EI}$
(2) SIMPLY SUPPORTED 	$R_m = \frac{4M_u}{L}$	$X_e = \frac{R_m L^3}{48EI}$
(3) FIXED SUPPORTS 	$R_m = \frac{4(M_u^+ + M_u^-)}{L}$	$X_e = \frac{R_m L^3}{192EI}$

WHERE:

M_u^+ = ULTIMATE POSITIVE MOMENT CAPACITY. (FT.-LBS)

M_u^- = ULTIMATE NEGATIVE MOMENT CAPACITY. (FT.-LBS.)

I = MOMENT OF INERTIA ($IN.^4$)

FOR REINFORED CONCRETE $I = I_a$

SEE NOTES ACCOMPANYING THIS TABLE

TABLE 3.6-9 (Continued) Page 2 of 3

NOTES:

The resistance of typical structural elements, whose flexural strength defines the minimum capacity, and their yield displacement approximations are presented in Tables 3.6-9 and 3.6-10. It is preferable that the limiting capacity of an element be in the flexural mode, not in shear. In evaluating the yield displacement with the usual elastic analysis, the moment of inertia must account for cracking of concrete sections. The empirical relation for this type of loading is an average moment of inertia, I_a, calculated as follows:

$$I_a = 1/2\ (I_g + I_c) = 1/2 \left(\frac{bt^3}{12} + Fbd^3\right)$$

Where:

I_g = Moment of inertia of gross concrete cross section of thickness t about its centroid (neglecting steel areas) (inches^4)

I_c = Moment of inertia of the cracked concrete section (inches^4)

b = Width of concrete sections (inches)

F = Coefficient for moment of inertia of cracked section with tension reinforcing only (see Figure 3.6-121)

t = Concrete thickness (inches)

d = Distance from extreme compression fiber to centroid of tension reinforcing (inches)

The moment of inertia, I_a, as calculated by the above equation must be used in the displacement equation in Tables 3.6-9 and 3.6-10 for all reinforeced concrete members. The ultimate moment capacity of a concrete section is considered as the moment strength:

$$M_u = 0.9A_s\ f_{dy}\ (d - a/2)\ \text{(inch-lbs)}$$

TABLE 3.6-9 (Continued) Page 3 of 3

NOTES: (Continued)

Where:

A_s = area of tensile reinforcing steel (in^2)

f_{dy} = allowable dynamic yield stress for reinforcing steel ($lbs./in^2$)

d = distance from extreme compression fiber to centroid of tension reinforcing (inches)

a = depth of equivalent rectangular stress block (inches)

If the element has compression steel, the appropriate equation for compression steel applies.

The amount of reinforcing steel in concrete members satisfied the following criteria:

For members with tension steel only:

$$\frac{1.4\sqrt{f'_c}}{f_y}\left(\frac{t}{d}\right)^2 \leq \frac{A_s}{bd} \leq \frac{0.25\ f'_c}{f_y}$$

For members with tension and compression steel:

$$\frac{1.4\sqrt{f'_c}}{f_y}\left(\frac{t}{d}\right)^2 \leq \frac{A_s}{bd}$$

$$\frac{A_s - A'_s}{bd}\left(\frac{t}{d}\right)^2 \leq \frac{0.25\ f'_c}{f_y}$$

Where:

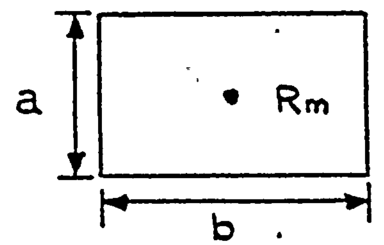
f'_c = compression strength of concrete (lbs/in^2)

A'_s = area of compressive reinforcement of concrete ($inch^2$)

TABLE 3.6-10

RESISTANCE-YIELD DISPLACEMENT VALUES FOR SLABS AND PLATES

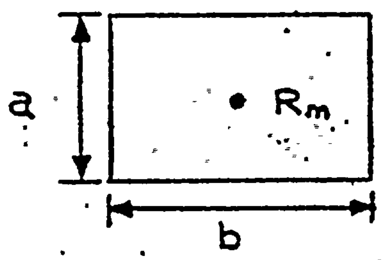
DESCRIPTION	RESISTANCE	YIELD DISPLACEMENT
(1) SIMPLY SUPPORTED ON ALL 4 SIDES WITH LOAD AT CENTER.	$R_m = 2\pi M_u$	$X_e = \dfrac{\alpha R_m a^2}{12EI}(1-\nu^2)$



b/a	1.0	1.1	1.2	1.4	1.6	1.8	2.0	3.0	∞
α	0.1390	0.1518	0.1624	0.1781	0.1884	0.1944	0.1981	0.2029	0.2031

(2) FIXED SUPPORTS ON ALL 4 SIDES WITH LOAD AT CENTER.

ν = POISSON'S RATIO
t = THICKNESS (IN.)
E = MODULUS OF ELASTICITY (lb/IN.^{2})
I = MOMENT OF INERTIA PER UNIT WIDTH (IN.^{4}/IN.), FOR REINFORCED CONCRETE SECTION $I = I_a$ SEE NOTES ACCOMPANYING TABLE 3.6-9.
M_u^+ = ULTIMATE POSITIVE MOMENT CAPACITY (IN. lb./IN.)
M_u^- = ULTIMATE NEGATIVE MOMENT CAPACITY (IN. lb./IN.)



$$R_m = 2\pi(M_u^+ + M_u^-)\ ;\ X_e = \frac{\alpha R_m a^2}{12EI}(1-\nu^2)$$

b/a	1.0	1.2	1.4	1.6	1.8	2.0	∞
α	0.0671	0.0776	0.0830	0.0854	0.0864	0.0866	0.0871

TABLE 3.6-11

DESIGN LOADS 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)
16 to 18	R 15	422	0.51	0°	40°	–	–	59	1.4k Pump
19 to 20	R 113	441	0.33	0°	40°	250	59	68	None
21 to 23	R 112	441	0.51	0°	40°	250	59	68	None
24 to 26	R 206	471	0.05	0°	40°	250	32	34	None
27 to 31	R 313	510'-6"	0.48	0°	40°	250	40	30	None
32 to 33	R 305	501	0.00	0°	40°	1000	34	85	None
34 to 35	R 408	522	1.0	0°	–	250	41	88	None
36 to 45	R 406 & 407	522	15.0	0°	–	250	126	0	1.5k Pump
46 to 55	R 409	535	11.0	0°	–	250	40	80	None
56 to 57	R 511	548	4.4	20°	–	400	80	55	None
58 to 68	R 510	548	1.8	20°	–	400	65	51	Heat Exchs. 16.2 & 29.5
69 to 70	R 509	548	2.1	20°	–	400	88	50	None
71 to 72	R 106	444	0.09	0°	40°	250	74	84	None
73 to 82	R 206	471	0.05	0°	40°	250	84	34	Vaporizer
83 to 86	R 604	572	0.03	0°	40°	250	15	36	None
87 to 93	R 206	471	0.05	0°	40°	250	75	45	Vaporizer
94 to 97	R 504	548	0.00	0°	40°	400	59	15	None
98 to 106	R 604	572	0.03	0°	40°	250	15	36	Heat and Vent Unit 51k
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 thru 3.6-62
2. For vertical and horizontal seismic factors, see 3.7
3. For pipe break thrust loads at pipe break number locations, see Table 3.6-6

NOT USED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-11

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 223	
Node 224	
Node 226	
Node 227	
Node 229	
Node 230	
Node 232	
Node 291	
Node 628	
Node 630	
Node 632	
Node 634	

DELETED

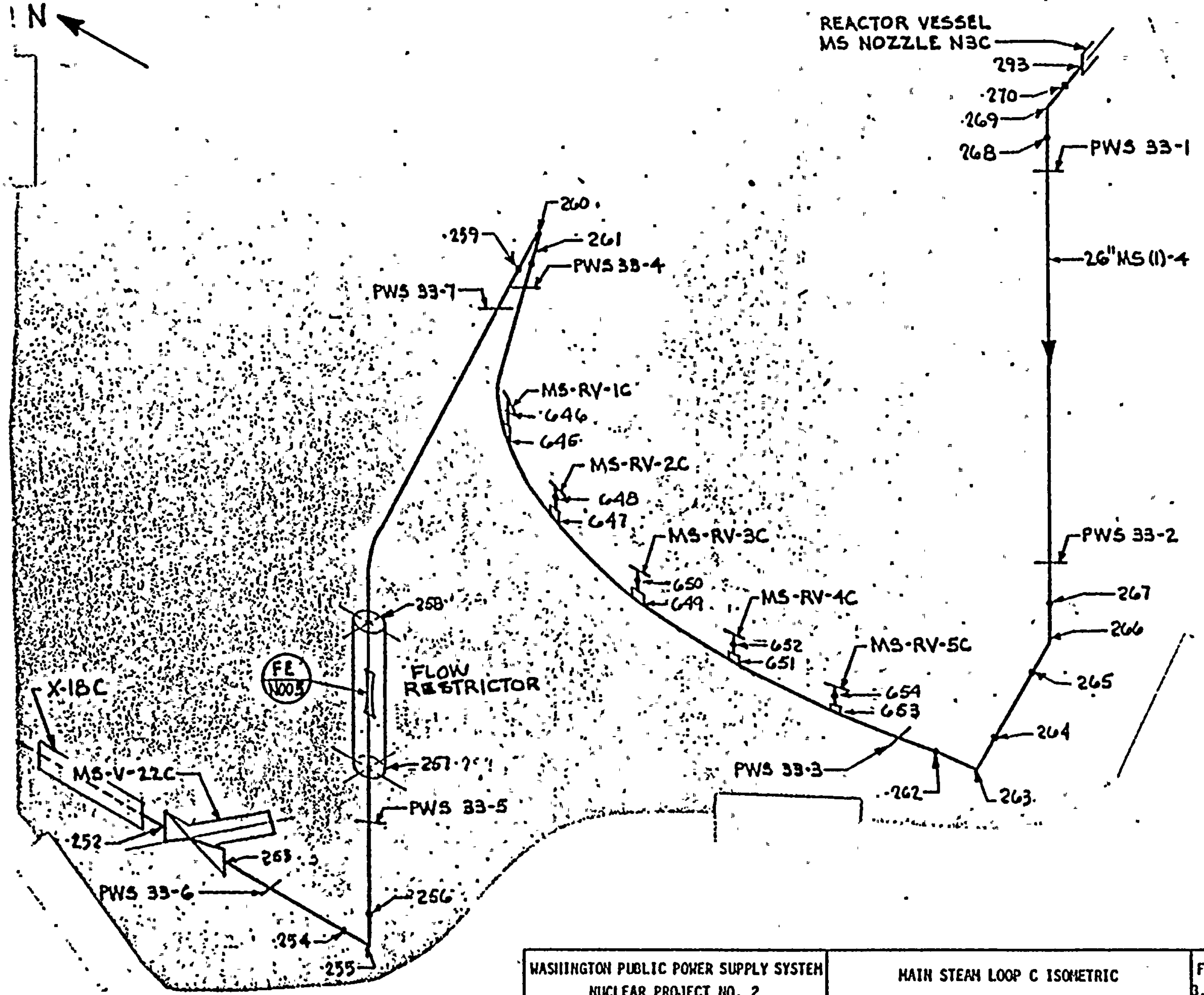
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-12c

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 233	Node 236
Node 234	Node 241
Node 235	Node 244
Node 237	Node 247
Node 238	Node 250
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	

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-13c



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 648
Node 650
Node 652
Node 654

LONGITUDINAL BREAKS

Node 255
Node 260
Node 263
Node 266
Node 269

DELETED

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 271
Node 272
Node 273
Node 275
Node 277
Node 278
Node 279
Node 280
Node 282
Node 283
Node 285
Node 286
Node 288
Node 290
Node 656
Node 658
Node 660
Node 662

LONGITUDINAL BREAKS

Node 274
Node 276
Node 281
Node 284
Node 287

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM LOOP D	FIGURE 3.6-15b

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-15c

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER LINE A	FIGURE 3.6-16b

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-16c

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-16d

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-16e

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER LINE B	FIGURE 3.6-17b

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-17c

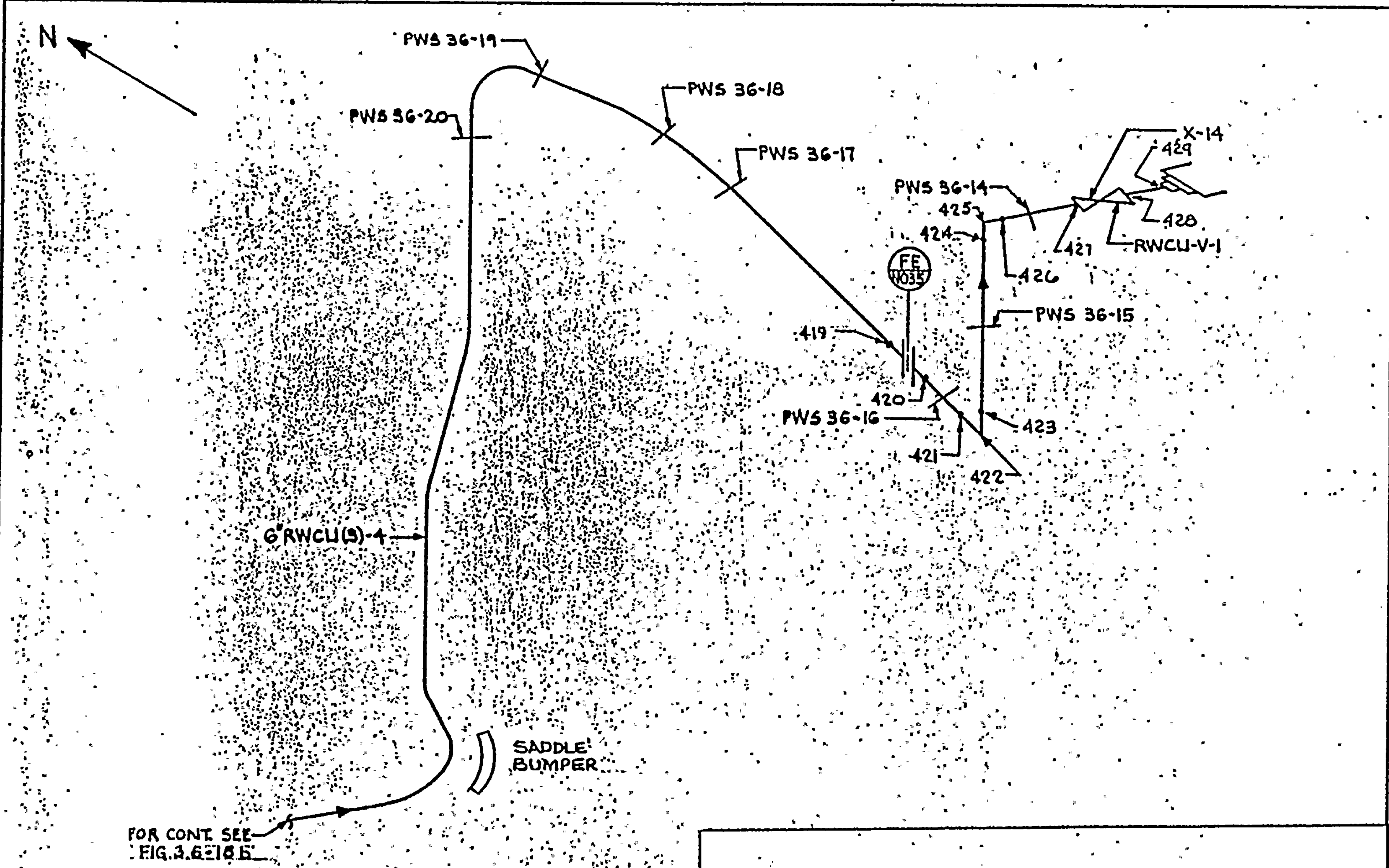
DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-17d

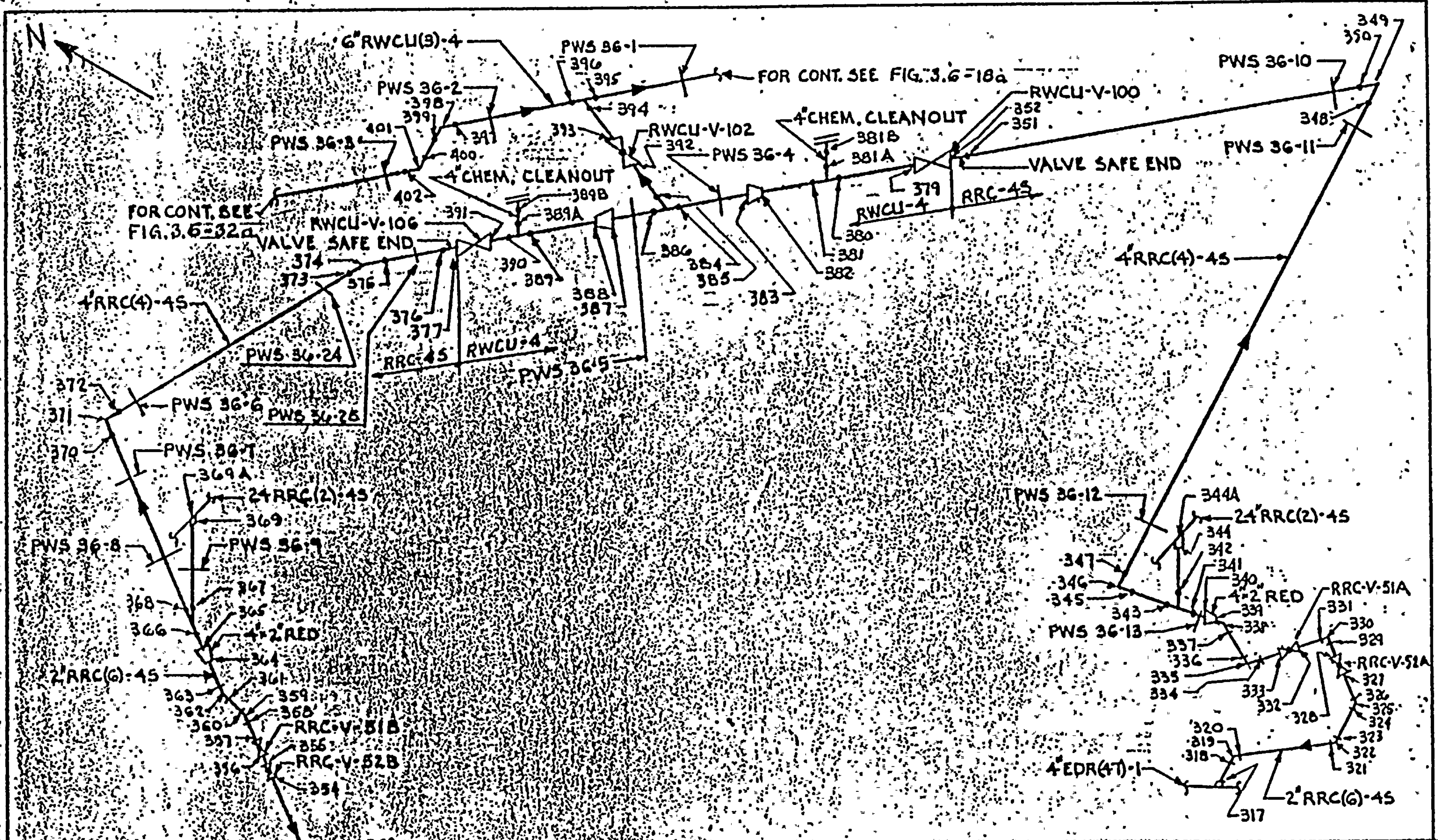
DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

FIGURE
3.6-17e



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR WATER CLEANUP ISOMETRIC	FIGURE 3.6-18a



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	LONGITUDINAL BREAKS
Node 419	Node 422
Node 420	Node 425
Node 421	
Node 423	
Node 424	
Node 426	
Node 427	
Node 428	
Node 429	

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 340	Node 372	Node 388
Node 341	Node 373	Node 389
Node 342	Node 375	Node 389A
Node 343	Node 377	Node 389B
Node 344	Node 379	Node 390
Node 345	Node 380	Node 391
Node 347	Node 381	Node 392
Node 348	Node 381A	Node 393
Node 350	Node 381B	Node 394
Node 352	Node 382	Node 395
Node 365	Node 383	Node 396
Node 366	Node 384	Node 397
Node 367	Node 385	Node 399
Node 368	Node 386	Node 400
Node 369	Node 387	Node 402
Node 370		

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)

DELETED

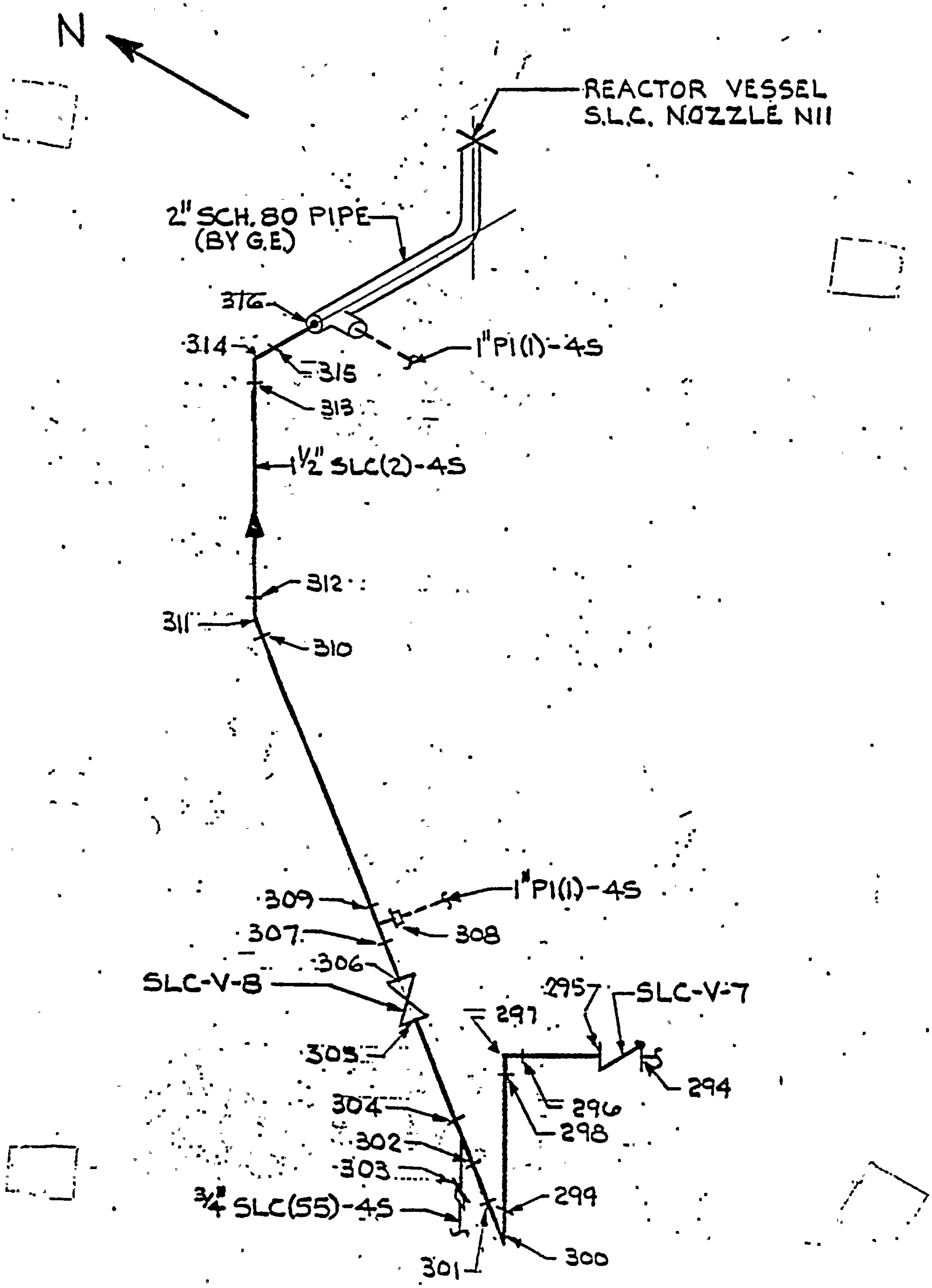
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-18e

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-18f

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-18g



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL ISOMETRIC	FIGURE 3.6-19a

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL	FIGURE 3.6-19b

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-19c

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL LPCI MODE LOOP A	FIGURE 3.6-20b

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 33
Node 34
Node 36
Node 37
Node 38
Node 39
Node 41
Node 42
Node 44
Node 45
Node 46
Node 47

LONGITUDINAL BREAKS

Node 35
Node 40
Node 43

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL LPCI MODE LOOP B	FIGURE 3.6-21b

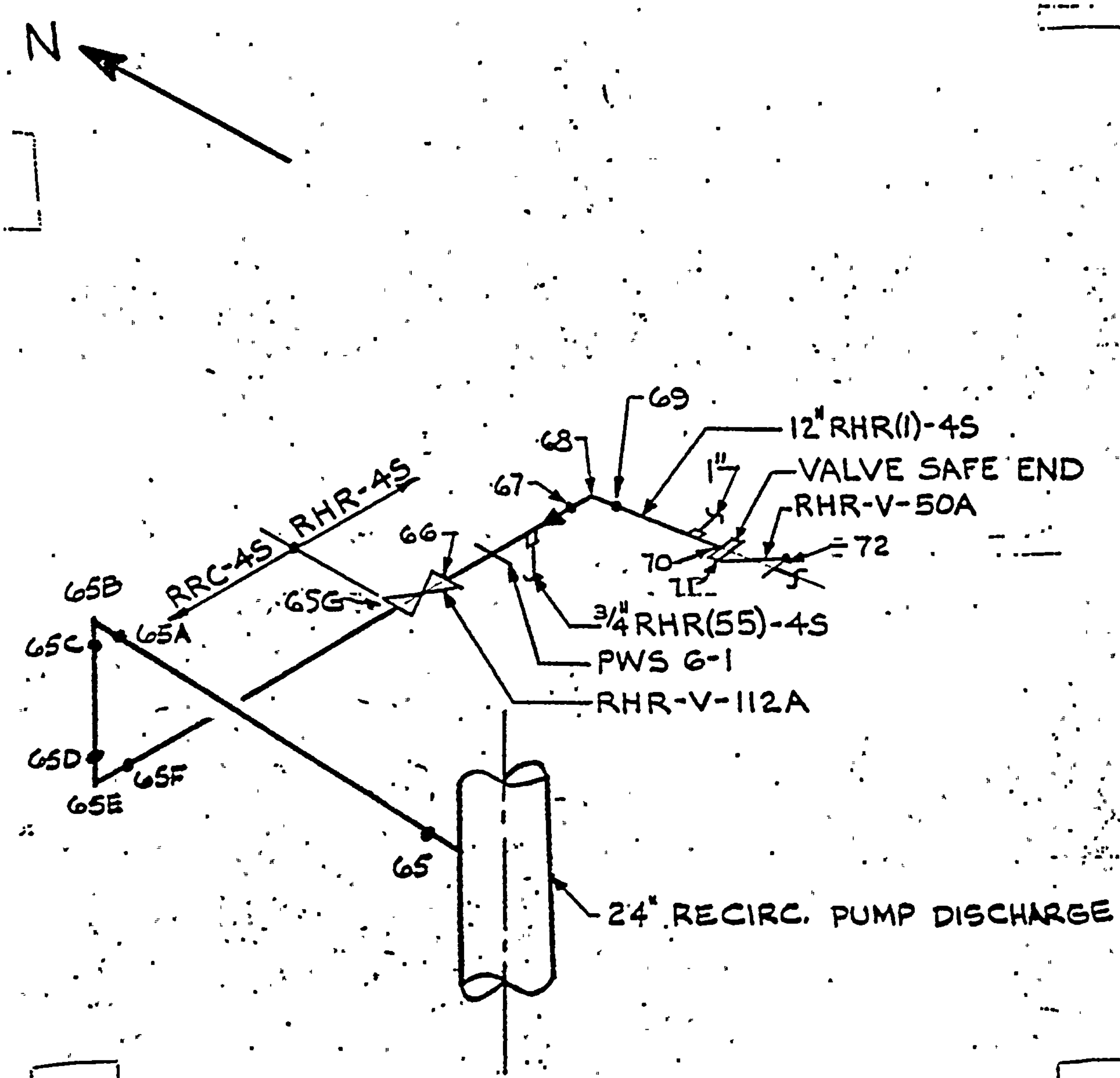
SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 49
Node 50
Node 52
Node 53
Node 54
Node 55
Node 57
Node 58
Node 60
Node 61
Node 62
Node 63

LONGITUDINAL BREAKS

Node 51
Node 56
Node 59



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

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

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

LONGITUDINAL BREAKS

Node 72C
Node 72F
Node 76

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

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC RPV HEAD SPRAY	FIGURE 3.6-26b

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 1
Node 2
Node 4
Node 5
Node 6
Node 7

LONGITUDINAL BREAKS

Node 3

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LOW PRESSURE CORE SPRAY	FIGURE 3.6-27b

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 9	Node 11
Node 10	
Node 12	
Node 13	
Node 14	
Node 15	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY	FIGURE 3.6-28b

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS	LONGITUDINAL BREAKS
Node 194	Node 196
Node 195	Node 199
Node 197	Node 201A
Node 198	Node 204
Node 200	Node 207
Node 201	Node 210
Node 202	Node 213
Node 203	
Node 205	
Node 206	
Node 208	
Node 209	
Node 211	
Node 211A	
Node 212	
Node 212A	
Node 214	
Node 214A	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR CONDENSING MODE RCIC TURBINE STEAM	FIGURE 3.6-29b

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-29c

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

Node 541	Node 557	Node 574
Node 542	Node 558	Node 576
Node 543	Node 559	Node 577
Node 544	Node 560	Node 578
Node 545	Node 562	Node 579
Node 546	Node 563	Node 580
Node 547	Node 565	Node 582
Node 549	Node 566	Node 583
Node 550	Node 568	Node 585
Node 551	Node 569	Node 586
Node 553	Node 570	Node 588
Node 554	Node 571	Node 589
Node 556	Node 573	

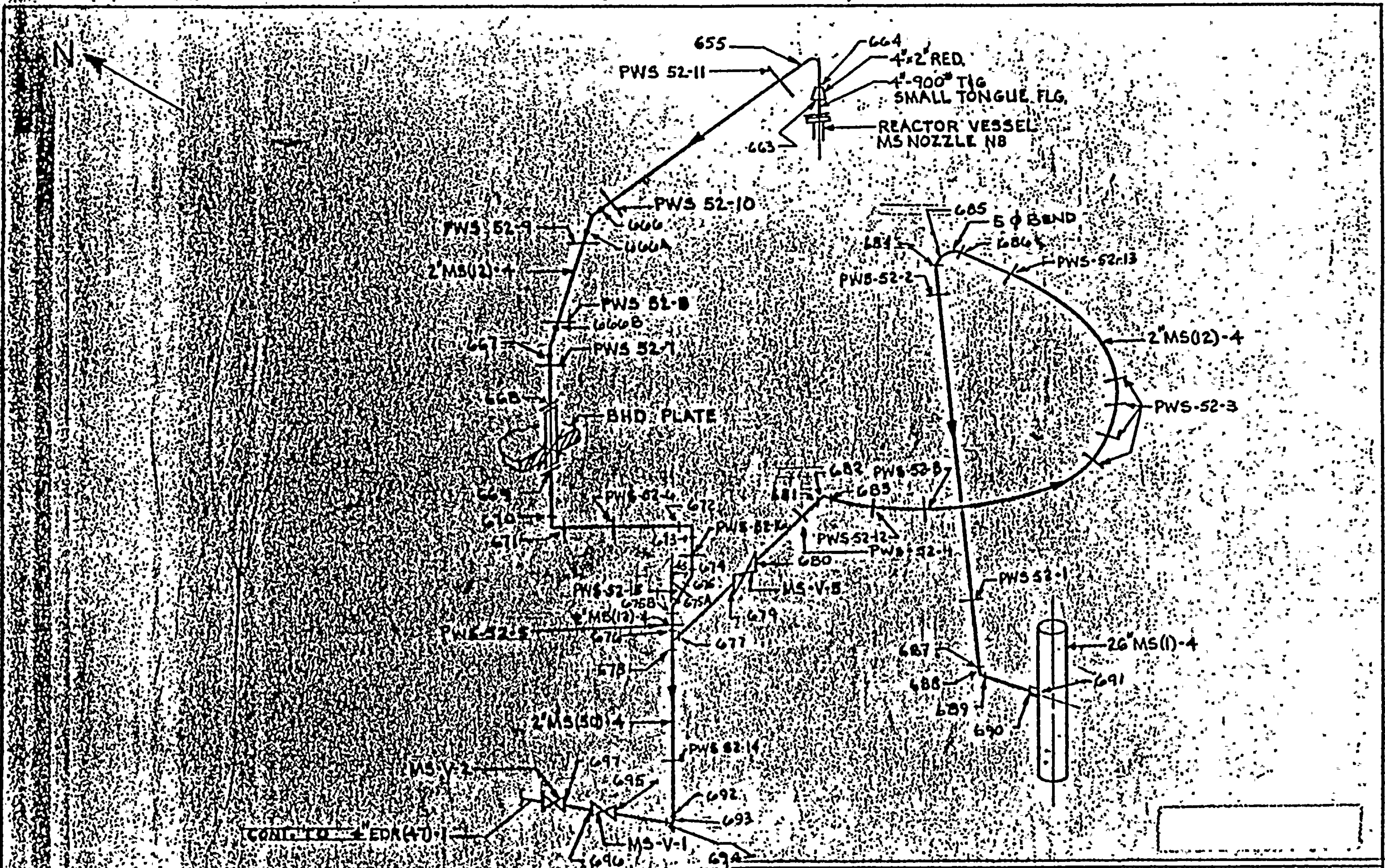
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM VALVES DRAINAGE PIPING	FIGURE 3.6-30b

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-30c

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-30d



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM RPV HEAD VENT ISOMETRIC	FIGURE 3.6-31a

SUMMARY OF POSTULATED PIPE BREAK LOCATIONS

CIRCUMFERENTIAL BREAKS

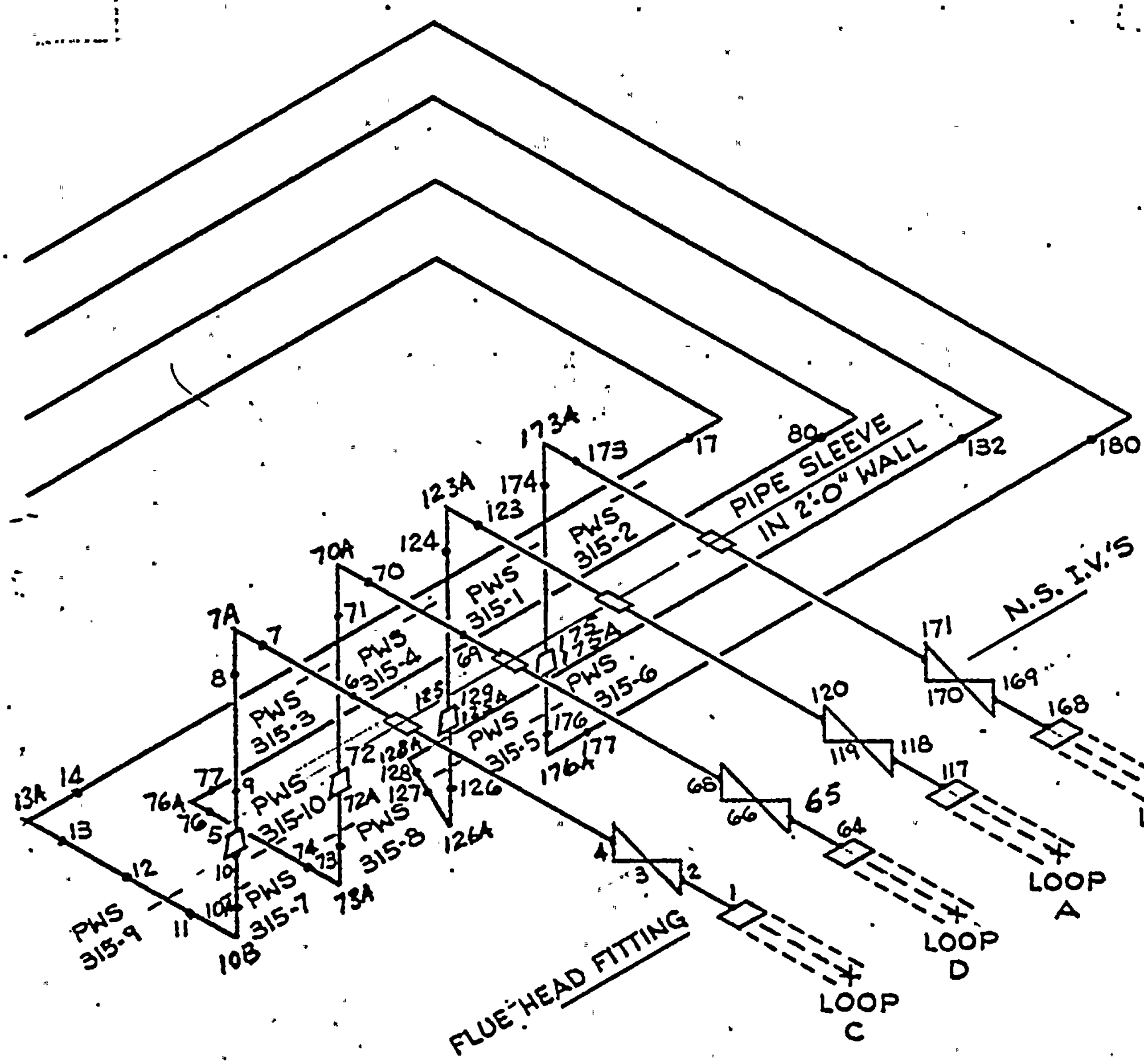
Node 663	Node 676
Node 664	Node 677
Node 666	Node 678
Node 666A	Node 679
Node 666B	Node 680
Node 667	Node 681
Node 668	Node 683
Node 669	Node 687
Node 670	Node 689
Node 671	Node 690
Node 672	Node 692
Node 673	Node 694
Node 674	Node 695
Node 675	Node 696
Node 675A	Node 697
Node 675B	Node 698

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-31c

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RRC REACTOR PRESSURE VESSEL DRAIN	FIGURE 3.6-32b



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		LONGITUDINAL BREAKS
Node 1	Node 80	Node 7A
Node 2	Node 117	Node 10B
Node 4	Node 118	Node 13A
Node 5	Node 120	Node 70A
Node 7	Node 123	Node 73A
Node 8	Node 124	Node 76A
Node 10	Node 125	Node 123A
Node 10A	Node 125A	Node 126A
Node 11	Node 126	Node 128A
Node 13	Node 127	Node 173A
Node 14	Node 128	Node 176A
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	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM LOOP A, B, C, & D INSIDE MAIN STEAM TUNNEL	FIGURE 3.6-33b

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-33c

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-33d

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-33e

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER PIPING LOOP A INSIDE MAIN STEAM TUNNEL	FIGURE 3.6-34b

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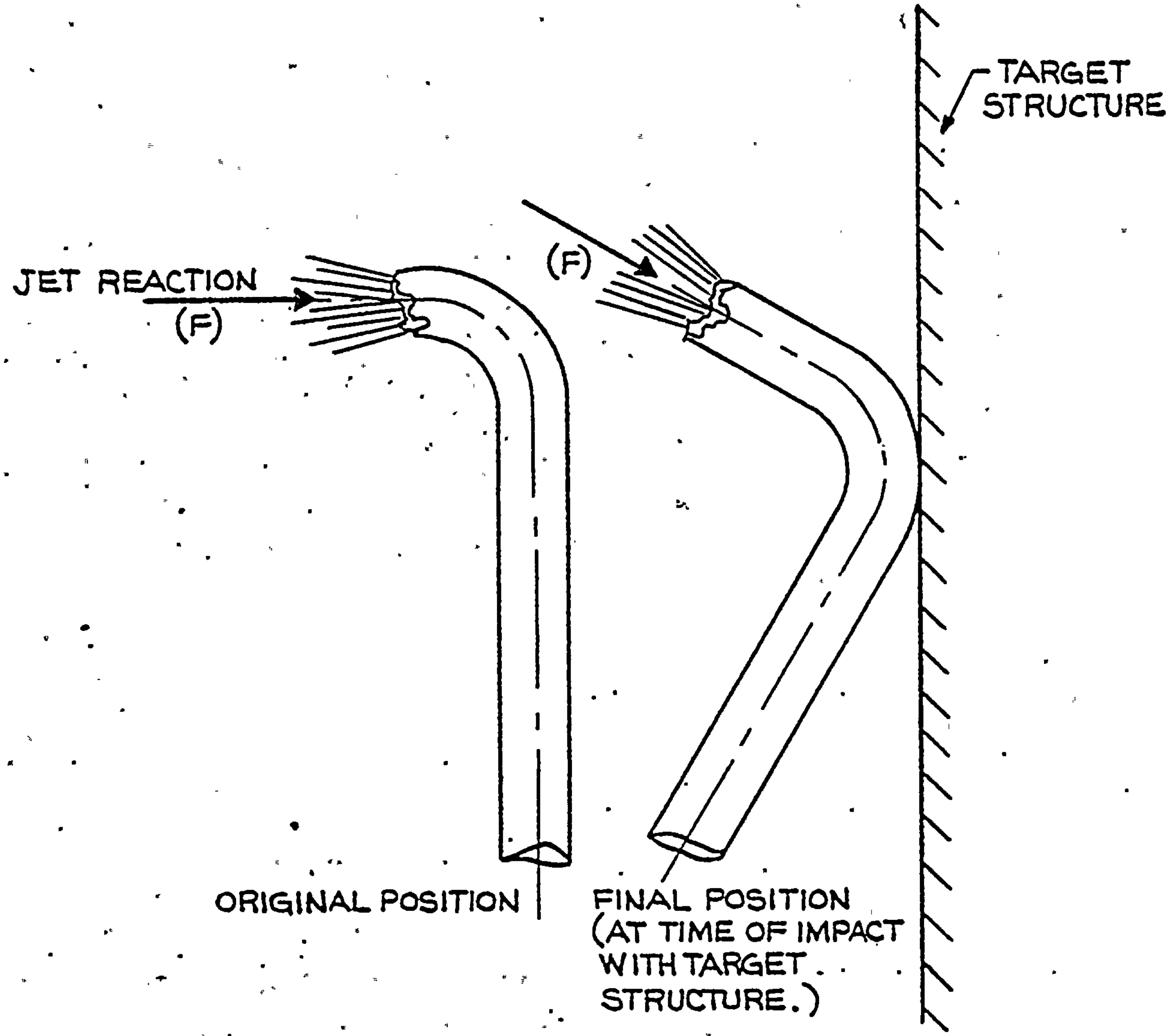
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-34c

AMENDMENT NO.



NOTE: EFFECTS ON TARGET STRUCTURE ARE:

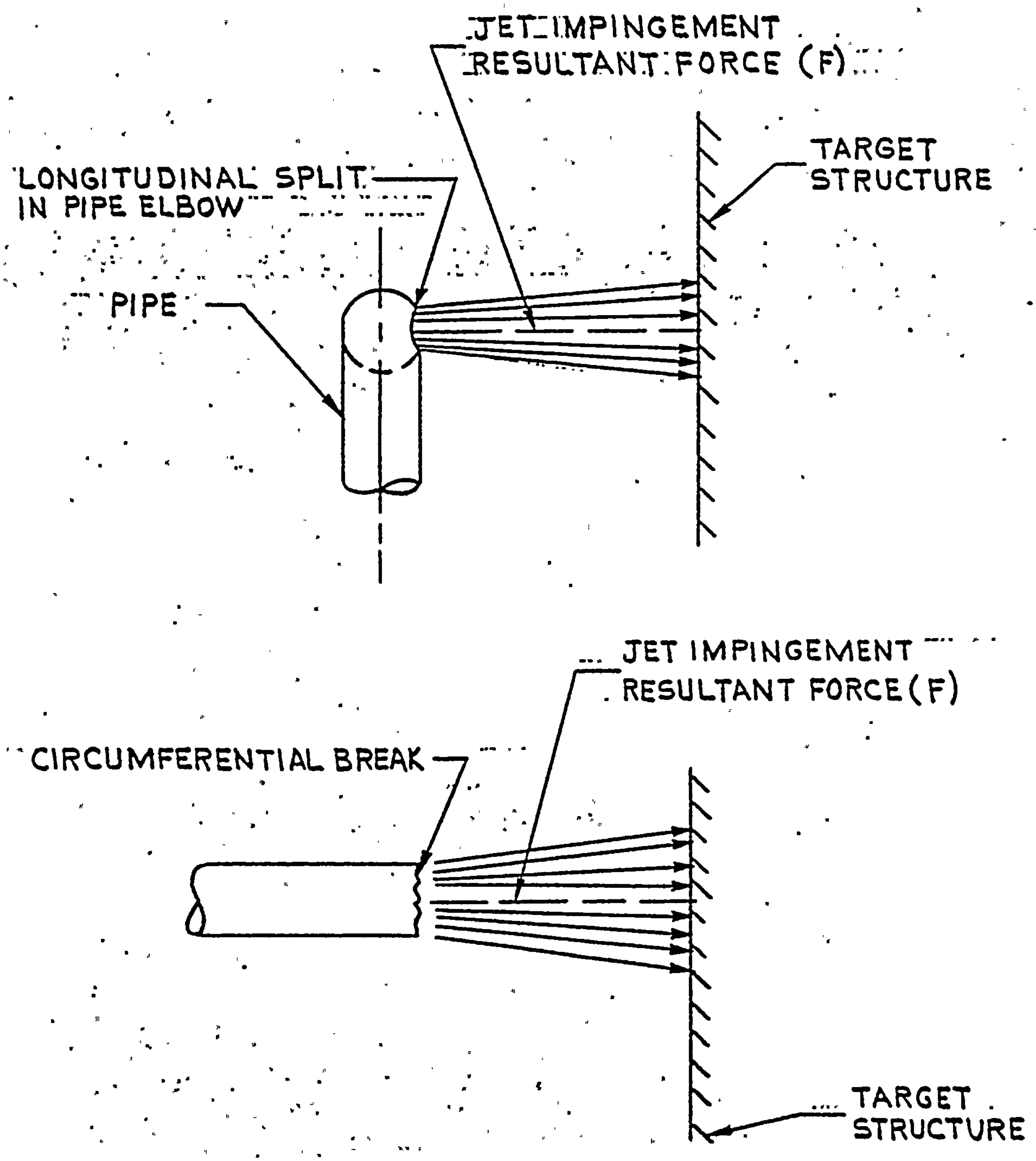
(1) A JET REACTION FORCE, F (FOR TIME HISTORY DESCRIPTION, SEE FIG. 3.6-118), AND

(2) IMPACT DUE TO ENERGY ACCUMULATED BY PIPE WHILE BEING ACCELERATED FROM ORIGINAL TO FINAL POSITION.

(3) CIRCUMFERENTIAL BREAK IS SHOWN

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PIPE WHIP IMPACT ON TARGET	FIGURE 3.6-116

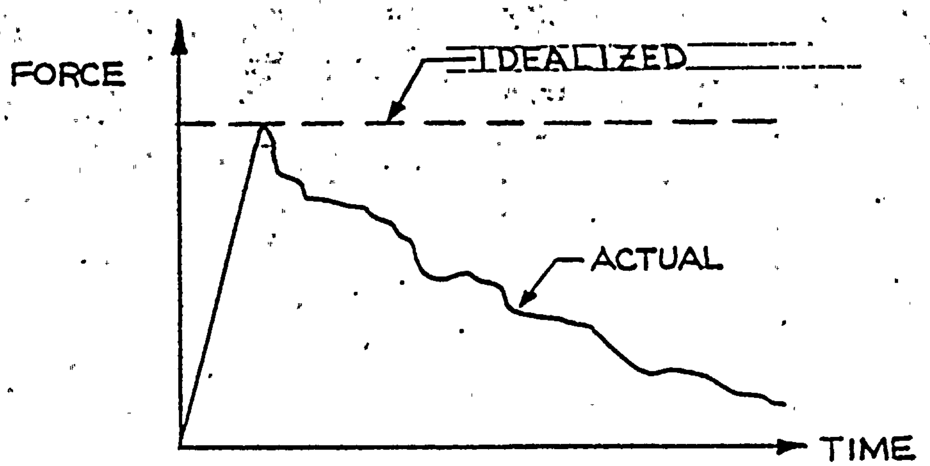
AMENDMENT NO.



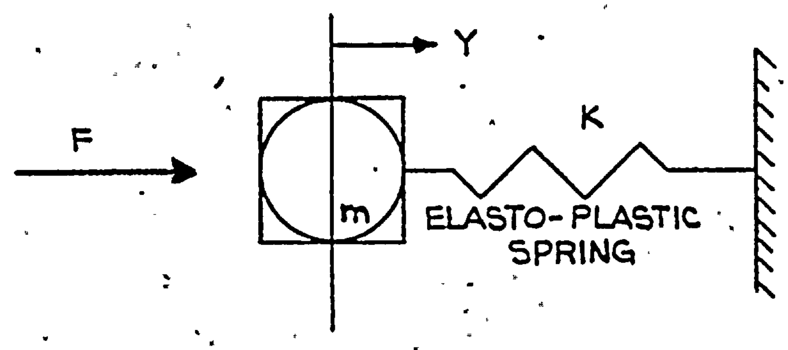
NOTE: FOR TIME HISTORY DESCRIPTIONS SEE FIG. 3.6-118

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	JET IMPINGEMENT ON TARGET	FIGURE 3.6-117

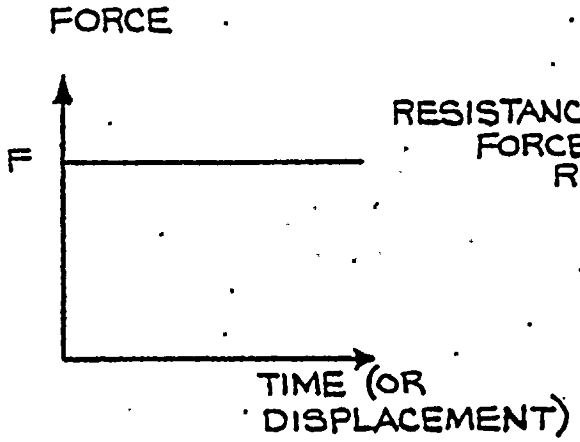
AMENDMENT NO.

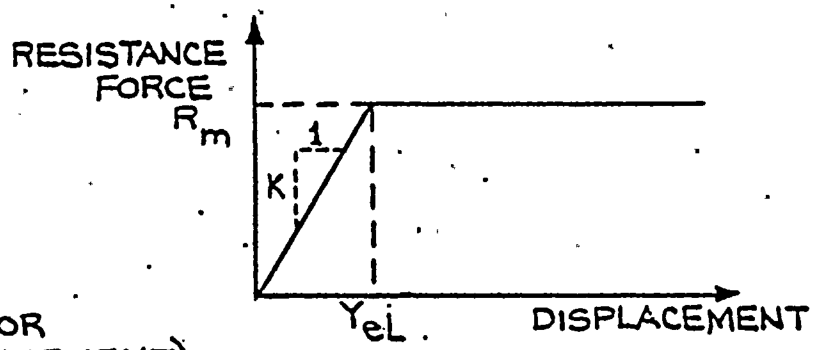


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TIME HISTORY OF JET IMPINGEMENT AND REACTION FORCE	FIGURE 3.6-118



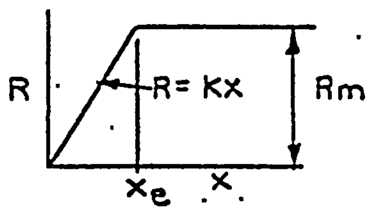
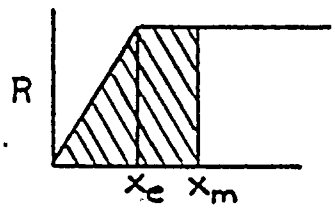
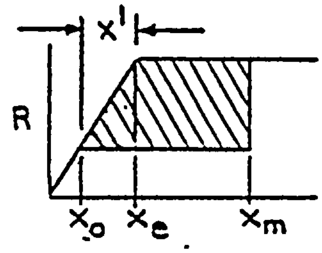
(A) SINGLE DEGREE OF FREEDOM MATHEMATICAL IDEALIZATION FOR A STRUCTURE.





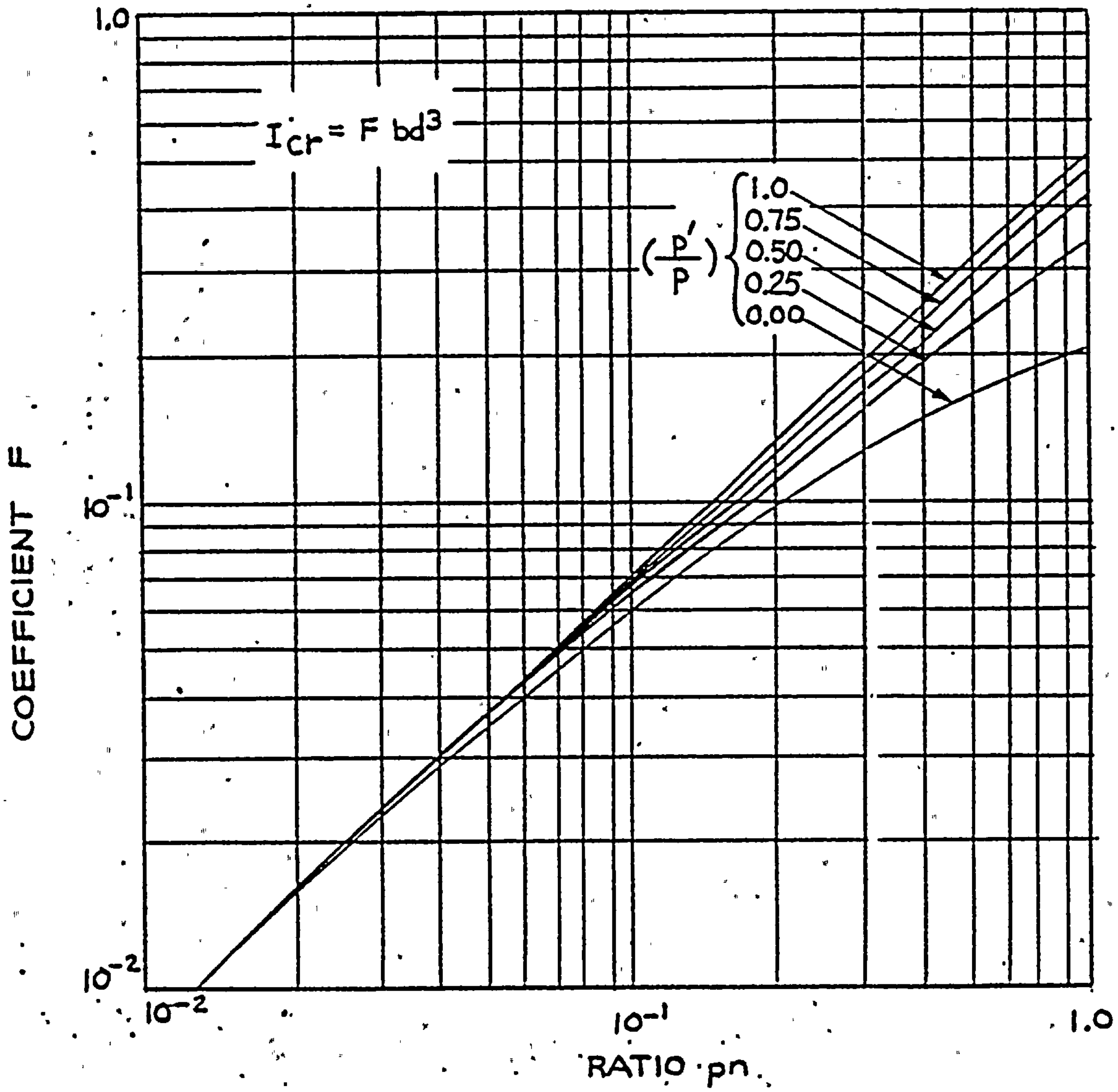
(B) STEP FUNCTION LOAD (C) RESISTANCE FUNCTION

AMENDMENT NO.

RESPONSE	RESISTANCE-DISPLACEMENT FUNCTION	AVAILABLE STRAIN ENERGY WITHOUT OTHER LOADING	AVAILABLE STRAIN ENERGY WITH OTHER LOADING
ELASTO-PLASTIC			

NOTE: SHADED AREA (STRAIN ENERGY) MUST EQUAL E_S (FROM 3.6.1.6.3.2.c)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESISTANCE-DISPLACEMENT FUNCTIONS WITH ASSOCIATED STRUCTURAL RESPONSE WITH & WITHOUT EFFECT OF OTHER LOADS	FIGURE 3.6-120



$$p = \frac{A_s}{bd}, \quad p' = \frac{A'_s}{bd}, \quad n = \frac{E_s}{E_c}$$

$$F = \frac{K^3}{3} + pn(1-K)^2 + \left(\frac{2n-1}{n}\right)(pn)\frac{p'}{p}\left(K - \frac{d'}{d}\right)^2$$

$$\frac{2n-1}{n} \cong 1.9, \quad \frac{d'}{d} \cong 0.10, \quad K = -m + (m^2 + 2q)^{\frac{1}{2}}$$

$$m = pn\left(1 + 1.9\frac{p'}{p}\right), \quad q = pn\left(1 + 0.19\frac{p'}{p}\right)$$

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	COEFFICIENTS FOR MOMENT OF INERTIA OF CRACKED SECTIONS	FIGURE 3.6-121

NOTE:
THE WALL REBOUND FORCE IS USED IN A DIRECTION OPPOSITE TO THE PIPE OR PIPE JET IMPACT LOAD.

TO DETERMINE R_m:

$$\frac{Y_{eL_2}}{R_{m_2}} = \frac{Y_{eL_1}}{R_{m_1}}$$

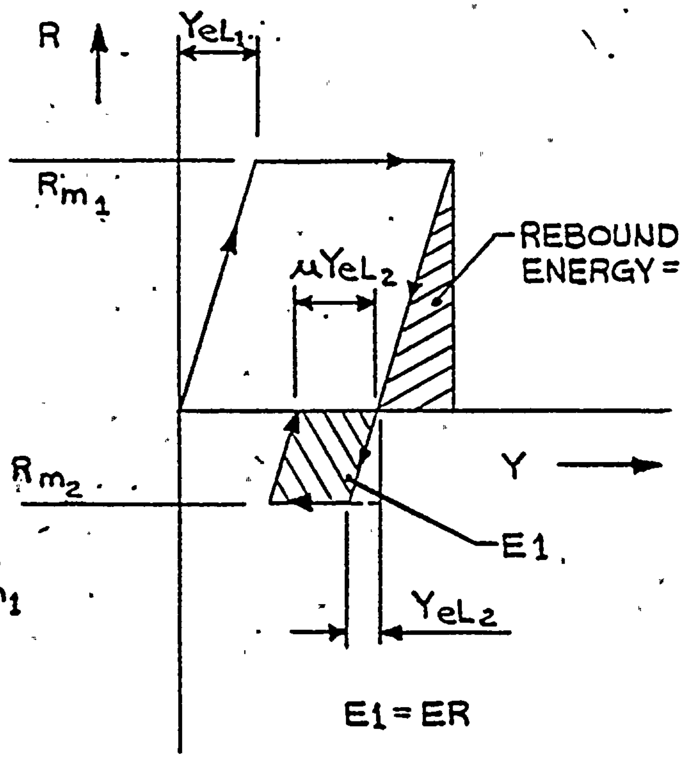
$$Y_{eL_2} = Y_{eL_1}\left(\frac{R_{m_2}}{R_{m_1}}\right)$$

$$E1 = \mu Y_{eL_2} R_{m_2} = ER = \tfrac{1}{2} Y_{eL_1} R_{m_1}$$

OR $\mu Y_{eL_1}\left(\frac{R_{m_2}}{R_{m_1}}\right) R_{m_2} = \frac{1}{2} Y_{eL_1} R_{m_1}$

THEREFORE, $R_{m_2}^{\,2} = \frac{1}{2\mu}\left(R_{m_1}^{\,2}\right)$

OR $R_{m_2} = \frac{1}{\sqrt{2\mu}} R_{m_1}$



f. Abnormal Loads

Abnormal loads are loads generated by the design basis accident under consideration.

P_a = Maximum differential pressure equivalent static load within or across a compartment generated by the postulated pipe break, and including an appropriate margin to account for uncertainty in the calculations. A small break case is also investigated.

P_a loads are due to a high energy pipe break outside containment and are discussed in 3.6.1.6; this includes pipe break in the main steam tunnel.

P' = Negative internal pressure or positive internal pressure (noted below) relative to the outside atmosphere and acting only within the reactor building secondary containment in conjunction with other loading including the design basis tornado or the safe shutdown earthquake (SSE).

(1) Positive internal pressure = (+) 0.25 psig

(2) Negative internal pressure = (-) 0.012 psig

T_a = Effects of thermal environment on the structure generated by a postulated pipe break. This includes T_o for all other areas not affected by the pipe break. (See 3.6.1.6).

R_a = Effects of thermal environment on the pipe reactions on the structure and equipment reactions on the structure generated by a postulated pipe break. This includes R_o for all other areas not affected by the pipe break. (See 3.6.1.6).

R_r = Local effects on the structure (e.g., walls and barriers) generated by a postulated pipe break. (See 3.6.1.6) These effects include:

(1) Reactions from broken pipes, Y_r

(2) Jet impingement, Y_j

(3) Missile impact due to a postulated ruptured pipe, Y_m

Y_r = Equivalent static load on the structure generated by the reaction on the broken high - energy pipe during the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_j = Jet impingement equivalent static load on a structure generated by the postulated break, and including an appropriate dynamic load factor to account for the dynamic nature of the load.

Y_m = Missile impact load on structure generated by or during the postulated pipe break, as from pipe whipping, arrived at by an energy approach to account for its dynamic nature (See 3.6.1.6.3). Elasto-plastic behavior is assumed with appropriate ductility ratios (from Table 3.6-1), provided excessive deflection does not result in loss of function of any safety-related system.

In addition to their own dead loads, including the weight of equipment, piping, cable trays, etc., floors are designed for conservative live loads resulting from the movements of the largest possible pieces of equipment. These live loads are patterned to produce the most critical loading effects for the slabs and beams. Floors and roofs are checked for their ability to transmit shear loads through diaphragm action. The live load on subgrade walls includes a minimum surcharge load of 300 psf resulting from normal live loads.

a. Exterior and interior walls: 39'-0" pressure head internally confined by the walls

b. Floor slab: 22'-0" pressure head acting upward on the slab

For lateral soil pressures on exterior surfaces of the subgrade walls, see 3.7.2.

The Seismic shear forces on the exterior (shear) walls are obtained from the seismic analysis described in 3.7.2.

Loadings due to a high-energy line break outside the containment are discussed in 3.6.1 and 3.6.1.6.

Loadings on the spent fuel and dryer-separator pools include the effects of water set in motion by seismic accelerations and the thermal gradient resulting from the high temperature of the water in the pools.

The siding and roof deck on the reactor building superstructure are designed to blow off at a specified wind pressure, ensuring that only the steel frame need be designed for tornado loadings.

As noted in 3.8.4.1.1.4, overpressurization of the main steam tunnel in the secondary containment of the reactor building is prevented by means of venting the tunnel to the atmosphere and to the turbine generator building by means of blow out panels in the north end of the tunnel. The blow out panels are designed to blow off at 1/2 psi differential pressure. The tunnel is designed to withstand the internal differential pressure arrived at on the basis of the pressure history in the tunnel following a steam line break. For a discussion of this analysis, see 3.6.1.20.

3.8.4.4 Design and Analysis Procedures

Conventional elastic techniques are used in the design and analysis of all structural components, subject to qualifications presented in 3.5.3 and 3.6.1.6. All buildings are analyzed basically as shear wall structures, and all floors are checked for their ability to transmit shear forces through diaphragm action. Exterior walls are designed to resist a combination of vertical loads, bending moments and lateral shear and overturning moments associated with seismic forces (see 3.7.2) and tornado loads. Longitudinal and lateral

shears are transferred to the mat through shear friction reinforcement and keys. The floor slab or beam and column framing is modeled to most closely approximate the actual structural behavior, and all boundary conditions are determined by stiffness evaluation of the actual intersecting structural members at the points of interest.

The design and analysis procedures utilized comply with ACI 318-71 Code (Reference 3.8-10) for concrete structures and with the AISC Specification (Reference 3.8-11) for steel structures, except as qualified in 3.5.3 and 3.6.1.6.

All concrete structures, for both operating and design basis loadings, have concrete strains limited to 0.003 with the exception of structures analyzed for the effects of a high-energy line pipe break outside the containment, where the elasto-plastic method of design is used (see 3.6.1.6). For steel structures under operating and design basis loadings, strains are limited to within the elastic range, with the exception of structures analyzed for the effects of a pipe break where an elasto-plastic method of design is used together with appropriate ductility ratios (see 3.6.1.6).

Typical arrangements of reinforcing steel are shown for the reactor building exterior walls and floor slabs in Figures 3.8-30, 3.8-31, 3.8-36, 3.8-38 and 3.8-39.

All interfacing structures are separated by a gap. The gap is of sufficient horizontal dimension to preclude disturbance of Seismic Category I structures, including non-Seismic Category I safety related structures, and Seismic Category II structures, during the SSE and the Operating Basis Earthquake. The combined deflections of adjacent structures during the SSE and the Operating Basis Earthquake are less than the gap.

Seismic Category II structures are arranged and designed in such a manner that adjoining Seismic Category I structures, including non-Seismic Category I safety related structures, will not be damaged by Seismic Category II structures, because the stresses under the SSE and the Operating Basis Earthquake conditions are either within the elastic range or, if not, the plastic deformations are tolerable.

3.8.4.4.1 Reactor Building

The distribution of horizontal seismic shears and moments between the biological shield wall and the exterior walls, for the normal operating condition and the containment vessel flooded condition, is determined by using the capabilities of STRUDL II, as discussed in 3.12.

The steel superstructure of the reactor building is analyzed and designed using elastic methods in Reference 3.8-11. The steel roof trusses and the vertical and horizontal bracing are analyzed for seismic and tornado loads using the STRUDL II computer program.

For each of the loading combinations delineated in Table 3.8-15, the required sectional strength of concrete (U) is calculated using the strength design method of ACI 318-71 with the applicable capacity reduction factor, modified by a dynamic increase factor (from Table 3.6-8) in load combinations 6, 7 and 8 for the abnormal load categories in Table 3.8-15.

The symbol "U" denotes the section strength required to resist design loads or their related internal moments and forces based on the strength design methods described in ACI 318-71.

For the strength design method load combinations, the margins of safety are contained in the capacity reduction (ϕ) factors specified in the ACI 318-71 code.

3.8.4.5.2 Structural Steel

See Table 3.8-17 for the criteria used for:

a. Required limits of section strength, S and Y

b. Section moduli

The symbol "S" denotes the required section strength based on the elastic design methods and the allowable stressed defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings", February 12, 1969.

The 33% increase in allowable stresses due to seismic loading is not permitted.

The symbol "Y" denotes the section strength required to resist design loads based on plastic design methods described in Part 2 of the AISC "Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings", February 12, 1969, modified by a dynamic increase factor (from Table 3.6-8 in load combinations 6, 7 and 8 for the abnormal load categories in Table 3.8-16, under Plastic Design Method).

For steel structures, the elastic working stress design method of Part 1 of the AISC specification (See Table 3.8-9) is used. All the loads considered in the loading combinations are factored loads.

The plastic design method of Part 2 of the AISC specification (See Table 3.8-9) is used as may be required for such structures as pipe restraint supports and pipe whip impacted components such as steel plate barriers.

TABLE 3.8-15

LOAD COMBINATIONS AND LOAD FACTORS
SEISMIC CATEGORY I AND NON-SEISMIC CATEGORY I SAFETY RELATED
CONCRETE STRUCTURES OUTSIDE PRIMARY METAL CONTAINMENT

Method	Conditions	LOAD CATEGORY	LOAD COMBINATION No.	NORMAL									SEVERE ENVIRONMENTAL					ABNORMAL					EXTREME ENVIRON.		
				D	L	R_o	T_o	P_o	P_t	T_t	F	Q	E	W	H	F*	Q*	P_a	T_a	R_a	R_r	P'	E'	W'	H'
ACI 318-71 STRENGTH DESIGN METHOD	SERVICE LOAD CONDITIONS	CONSTRUCTION	U1	1.4																					
			U2	0.9	1.3	0.9	1.3							1.3											
		NORMAL	1	1.4	1.7			1.7																	
			1b	1.4	1.7	1.4	1.4	1.7																	
			U3	1.4	1.7			1.7			1.4	1.7													
			U4	1.4	1.7		1.4	1.7			1.4	1.7													
			U5	0.9							1.4	1.7													
			U6	0.9			1.4				1.4	1.7													
		SEVERE	2	1.4	1.7			1.7					1.9												
			2b	1.4	1.4	1.4	1.4	1.4					1.4												
			2b'	0.9									1.4												
			3	1.1	1.3			1.3						1.3											
			3b	1.1	1.3	1.1	1.1	1.3						1.3											
			3b'	0.9										1.3											
			U7	1.1	1.3			1.3			1.1	1.3		1.3											
			U8	1.1	1.3		1.1	1.3			1.1	1.3		1.3											
			U9	1.4	1.7			1.7					1.9			1.4	1.7								
			U10	1.4	1.4		1.4	1.4					1.4			1.4	1.4								
	FACTORED LOAD CONDITIONS SEE NOTES 1 & 2	EXTREME ENVIRONMENTAL	4	1.0	1.0	1.0	1.0																1.0		
			5	——	——	——	——																	——	
		ABNORMAL	6	1.0	1.0													1.5	1.0	1.0					
		ABNORMAL / SEVERE ENVIRONMENTAL	7	1.0	1.0								1.25					1.25	1.0	1.0	1.0				
		ABNORMAL / EXTREME ENVIRONMENTAL	8	1.0	1.0													1.0	1.0	1.0	1.0		1.0		
			U11	1.0	1.0	1.0	1.0															1.0	1.0		
				1.0	1.0	1.0	1.0															1.0		1.0	

SEE NOTES ON FOLLOWING PAGE.

TABLE 3.8-15 (Continued)

NOTES:

1. In combinations 6, 7 and 8, the maximum values of P_a, T_a, R_a, Y_r, and Y_j, including an appropriate dynamic load factor, are used. The value of Y_m is arrived at by an energy balance method of structural action (3.6.1.6.3.2), to account for the dynamic nature of the load.

   In combinations 7 and 8, local stresses due to concentrated loads Y_r, Y_j and Y_m may be permitted to exceed the allowable stresses, provided there is no loss of function of any safety-related system as a result thereof.

2. In considering the concentrated tornado missile load in Combination U11, local section strength capacities may be exceeded under these concentrated loads provided there is no loss of function of any safety-related system as a result thereof.

3. All the loads listed are not necessarily applicable to all concrete structures. Loads not applicable to a particular structure are deleted.

4. If, for any combination, the effect of any load other than dead loads reduces the stress it is deleted from the combination.

5. Combinations 1 through 8, 1b, 2b, 2b', 3b and 3b' correspond to those in the NRC Standard Review Plan for 3.8.4. Combinations U1 through U11 are not in the review plan for 3.8.4, and are used in addition to those of the review plan combinations.

6. Dashed lines indicate that the load or load combination is not used.

7. For load definitions see 3.8.4.3.

8. This table applies to 3.8.4 and 3.8.5. This table is displayed, in part, in 3.4.2.

9. Deleted. Replaced by Note 11.

10. Deleted. Replaced by Note 11.

11. Combinations 6, 7 and 8 are used only when abnormal loads generated by a postulated pipe break are included.

TABLE 3.8-16

LOAD COMBINATIONS AND LOAD FACTORS

SEISMIC CATEGORY I AND NON-SEISMIC CATEGORY I SAFETY RELATED

STEEL STRUCTURES OUTSIDE PRIMARY METAL CONTAINMENT*

		LOAD CATEGORY	LOAD COMBINATION No.	NORMAL					SEVERE ENVIRON.		ABNORMAL					EXTREME ENVIRON.	
				D	L	R_o	T_o	P_o	E	W	P_a	T_a	R_a	R_r	P'	E'	W'
ELASTIC WORKING STRESS DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	1.0	1.0			1.0									
			1a	--	--	--	--										
		SEVERE ENVIRONMENTAL	2	1.0	--			1.0	1.0								
			2a	--	--	--	--		--								
			3	1.0	--			1.0		1.0							
			3a	--	--	--	--			--							
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	4	--	--	--	--									--	
			5	--	--	--	--										--
		ABNORMAL	6	--	--						--	--	--				
			U12	1.0	1.0										1.0		
		ABNORMAL/SEVERE ENVIRONMENTAL	7	--	--				--		--	--	--	--			
		ABNORMAL/EXTREME ENVIRONMENTAL	8	--	--						--	--	--	--		--	
			U13	1.0											1.0	1.0	
			U14	1.0											1.0		1.0
PLASTIC DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	1.0	1.0			1.0									
			1b	--	--	--	--										
		SEVERE ENVIRONMENTAL	2	1.0	--			1.0	1.0								
			2b	--	--	--	--		--								
			3	1.0	--			1.0		1.0							
			3b	--	--	--	--			--							
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	4	--	--	--	--									--	
			5	--	--	--	--										--
		ABNORMAL	6	1.0	1.0						1.5	1.0	1.0				
			U12	1.0	1.0										1.0		
		ABNORMAL/SEVERE ENVIRONMENTAL	7	1.0	1.0				1.25		1.25	1.0	1.0	1.0			
		ABNORMAL/EXTREME ENVIRONMENTAL	8	1.0	1.0						1.0	1.0	1.0	1.0		1.0	
			U13	1.0											1.0	1.0	
			U14	1.0											1.0		1.0

* SEE NOTES APPLICABLE TO TABLE ON FOLLOWING PAGE.

TABLE 3.8-16 (Continued) Page 2 of 2

NOTES:

1. In combinations 6, 7 and 8, (factored load conditions, Plastic Design Method), the maximum values of P_a, T_a, R_a, Y_j, and Y_r, including an appropriate dyanmic load factor, are used; and the value of Y_m is arrived at by an energy balance method of structural action (3.6.1.6.3.2), to account for the dynamic nature of the load.

   In combinations 7 and 8, (factored load conditions, Plastic Design Method), local stresses due to concentrated loads Y_r, Y_j and Y_m may be permitted to exceed the allowable stresses, provided there is no loss of function of any safety-related system as a result thereof.

   In considereing the concentrated tornado missile load in combination U14, local section strength capacities may be exceeded under these concentrated loads provided there is no loss of function of any safety-related system as a result thereof.

2. Thermal loads for factored load conditions are neglected when it can be shown that they are secondary and self-limiting in nature.

3. All the loads listed are not necsessarly applicable to all concrete structures. Loads not applicable to a particular structure are deleted.

4. If, for any load combination, the effect of any load other than D reduces the stress, it is deleted from the combination.

5. Combinations 1 through 8, 1a, 2a, 3a, 1b, 2b, and 3b correspond to those in the NRC Standard Review Plan for 3.8.4. Combinations U12, U13 and U14 are not in the review plan for 3.8.4, and are used in addition to those of the review plan combinations.

6. Dashed lines indicate that the load or load combination is not used.

7. For load definitions, see 3.8.4.3.

8. This table applies to 3.8.4.

9. Combinations 6, 7 and 8 in the Plastic Design Method are used only when abnormal loads generated by a postulated pipe break are included.

Q. 110.002
(3.6.1)

Provide in Section 3.6.1.11.2.1 of the FSAR a definition of what is meant by the term "contiguous grid". Indicate clearly whether it includes the corner grids (i.e., which are diagonally adjacent). Discuss the vertical extent of a contiguous grid. Additionally, provide justification for not assuming the simultaneous destruction of equipment in more than one contiguous grid.

Response:

In the revised pipe break and missile evaluation the "grid" approach is not used. Results of this evaluation and the methodology used will be provided in a future amendment after the evaluation has been completed.

Q. 110.003
(3.6.1)

Describe in 3.6.1.11.2 of the FSAR, how you evaluate the environmental effects of leakage cracks in high energy fluid systems postulated in accordance with the criteria contained in 3.6.2.1.3 and 3.6.2.1.4.2.

Response:

Leakage cracks in high energy piping are not postulated, since the environmental and flooding effects of high energy pipe breaks are considered as described in 3.6.1.11, and are bounding. The FSAR will be revised to reflect the results after the current pipe break and missile evaluation has been completed.

Q. 110.004
(3.6.1)

Expand Section 3.6.1.11.3.1 of the FSAR to: (1) provide justification for not assuming the simultaneous malfunction of equipment in one or more contiguous grids; (2) describe your procedures to evaluate the effects of flooding which are discussed in 3.6.2.1.4.2.c of the FSAR.

Response:

Environmental and flooding effects resulting from moderate energy piping failures are not assumed to be confined by grid boundaries. Upon completion of the current pipe break and missile evaluation, the FSAR text will be revised to reflect this, and to respond to part (2) of your question. ~~See the response to Question 110.033~~.

Q. 110.005
(3.6.2)

State in 3.6.2.1.1 of the FSAR, the criteria for postulating break locations in high energy piping not designed to Seismic Category I criteria.

Response:

The criteria for other piping runs is defined in 3.6.2.1.1.3.*

*See ~~the response to Question 110.008~~.
draft revised FSAR page change.

Q. 110.006
RSP
(3.6.2)

It is our position that a branch pipe connection to a main run of pipe need not be considered as a terminal end when all the following conditions are met: (1) the branch and main runs are of comparable size and degree of fixity (i.e., the nominal size of the branch is at least one half that of the main); (2) the intersection is not rigidly constrained by the building structure; and (3) the branch and main runs of pipe are modeled as a common piping system in the stress analysis of these pipes. Revise Note (a) of 3.6.2.1.1.1.a to correspond with this definition of terminal ends.

Response:

For the response see revised 3.6.2.1.1.1.a.*

*See attached draft page changes.

3.6.2.1.1.1 Postulated Pipe Break Locations in ASME Section III Class 1 Piping

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 ~~is~~ 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.

Q. 110.007
(3.6.2)

Indicate in Note (f) of 3.6.2.1.1.1.b(2)(c) of the FSAR the range of plant operating conditions considered in your evaluation of Equation 13 of Subsection NE-3653.6(b) of the ASME Code.

Response:

In this evaluation we are using the load combinations applicable to the normal plant conditions defined in footnote (a) to FSAR 3.6.2.1.

Q. 110.007
(3.6.2)

Indicate in Note (f) of 3.6.2.1.1.1.b(2)(c) of the FSAR, the range of plant operating conditions considered in your evaluation of Equation 13 of subsection NE-3653.6(b) of the ASME Code.

Response:

In this evaluation we are using the load combinations applicable to the normal plant conditions defined in footnote a) to 3.6.2.1.

Q. 110.008
(3.6.2)

Indicate in 3.6.2.1.1.3 of the FSAR how the criteria for piping, which is not designed to comply with the ASME Boiler and Pressure Vessel Code, differs from that for piping which is designed to this code.

Response:

The criteria for postulating pipe break locations in piping not designed to comply with the ASME Boiler and Pressure Vessel Code does not differ from that for ASME Section III, Class 2 and 3 piping.

See revised Section 3.6.2.1.1.3.*

*See attached draft page change.

3.6.2.1.1.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Runs

a. The terminal ends 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 for valves and flanges.

(2) At each location where the stresses under the loadings resulting from upset plant conditions, including an OBE event as calculated by the summation of Equations (9) and (10) of ASME Code Section III Subsection NC, Paragraph NC 3652, exceed 0.8 ($1.2S_h + S_A$) where S_h and S_A are as defined in Paragraph NC 3611.2.

(3) If there are not at least two intermediate locations, where the above noted stresses exceed 0.8 ($1.2S_h + S_A$), a minimum of two separate locations are chosen based upon stress, except if the piping run has only one change of direction, a minimum of one intermediate break is postulated.

(4) Intermediate breaks are not postulated in sections of straight pipe where there are no pipe fittings, valves, or flanges.

3.6.2.1.1.3 Break Locations in Other Piping Runs

Postulated pipe break locations for piping other than ASME Code Section III Class 1, 2 and 3, are postulated in accordance with pipe whip criteria which ~~generally~~ conforms to the criteria set forth for ASME Code Section III Class 2 and 3 piping.

3.6.2.1.2 Postulated Pipe Break Locations in High Energy Fluid System Piping Between Containment Isolation Valves.

Pipe breaks (not including leakage cracks) are postulated in locations as indicated below:

Q. 110.009
(3.6.2)

Provide in Section 3.6.2.1.2.3 of the FSAR a definition of the phrase "through-wall leakage crack" for which the tunnel structures are designed. We note that this section cross-references 3.6.1.20 for further discussion. However, as noted in Question 110.001, this latter section is not in the FSAR.

Response:

The "through-wall leakage crack" is defined in 3.6.2.1.4.2. Section 3.6.1.20 has been provided with the response to Question 010.011.

Q. 110.010
RSP
(3.6.2)

It is our position that the piping which is between the containment isolation valves and for which no breaks are postulated, will receive a one hundred percent volumetric examination of all welds, including the circumferential, the longitudinal, and the branch to main run welds, during each inspection interval. (Refer to Subsection IWA-2400 of the ASME Code.) Accordingly, revise Section 3.6.2.1.2 of the FSAR to provide a commitment to such an augmented inservice inspection program.

Response:

As discussed in Section 3.8.6, the WNP-2 design incorporates integrally forged (one piece) Type 1 flued head fittings on all high energy piping containment penetrations with the exception of the main steam penetrations. The main steam penetrations are classified as Type 1, but are not integrally forged. They are constructed from a flued head forging welded to a section of process pipe. That penetration weld is classified as an integral attachment weld and not a pressure boundary weld. Neither the integrally forged nor the main steam flued heads contain longitudinal, circumferential, or branch to main run pressure boundary welds. All circumferential, longitudinal, and branch to main run welds between containment isolation valves in ASME Section III, Class 1 high energy piping systems will be examined prior to service according to the WNP-2 Preservice Inspection Program Plan. That plan requires one hundred percent volumetric examination of essentially all pressure boundary welds in piping exceeding 1" nominal diameter per the requirements of the ASME Section XI Code. In addition, one hundred percent volumetric examination of the main steam flued head integral attachment weld is required. The WNP-2 Inservice Inspection Program Plan will include provisions to repeat the examinations performed preservice during each inspection interval on pressure boundary welds in all high energy ASME Section III, Class 1 piping which is between the containment isolation valves for which no breaks are postulated. Section 3.6.2.1.2 has been revised accordingly.*

WNP-2

WNP-2 does not contain any ASME Section III, Class 2 high energy piping between containment isolation valves. All such piping is classified as moderate energy per the definitions provided in Section 3.6.2.1. Therefore, a commitment for an augmented ISI program for Class 2 piping is not required.

*See attached draft page change.

3.6.2.1.2.1 Postulated Pipe Break Locations in AMSE Section III Class I Piping

No pipe breaks are postulated in the portion of piping between primary containment isolation valves, if any of the following apply:

(1) S_n does not exceed $2.4S_m$.

(2) S_n exceeds $2.4S_m$ but does not exceed $3S_m$, and the Cumulative Usage Factor (U) does not exceed 0.1.

(3) S_n exceeds $3S_m$, but S_e and S_r are each less than $2.4S_m$, and U does not exceed 0.1.

The stress levels in the ASME Section III Class I containment penetration high energy piping are maintained at or below these limits and therefore, breaks are not postulated.(a) See 3.6.2.1.2.3 for further discussion of containment penetration piping.

3.6.2.1.2.2 Postulated Pipe Break Locations in ASME Section III Class 2 and 3 Piping Runs

See 3.6.2.1.1.2 b.(2) for stress criteria applicable to ASME Section III Class 2 and 3 piping between containment isolation valves.

The stress levels are maintained at or below these limits and therefore breaks are not postulated. See 3.6.2.1.2.3 for further discussion of containment penetration piping.

3.6.2.1.2.3 Primary Containment Penetration Piping

Primary containment penetrations, in order to maintain containment integrity, are designed with the following characteristics:

a. They are capable of withstanding the forces caused by impingement of the fluid from the rupture of the largest local pipe without failure.

b. They are capable of withstanding the maximum reactions that the pipes to which they are attached are capable of exerting.

(a) A program for augmented inservice inspection will be included in the WNP-2 Inservice Inspection Program Plans to provide one hundred percent volumetric examination each inspection interval of all pressure boundary welds ~~greater than one inch nominal pipe size~~ in Class I high energy piping between containment isolation valves for which no breaks are postulated.

exceeding one inch nominal diameter

3.8.6.1 Description

3.8.6.1.1 Piping Penetrations - Type 1

Process lines traverse the boundary between the inside of the steel primary containment vessel and the outside of the biological shield wall by means of piping penetration assemblies made up of several elements. Two general types of piping penetration assemblies are provided: Type 1 (also referred to as "hot" type piping penetration assemblies) and Type 2 (also referred to as "cold" type piping penetration assemblies).

Figure 3.8-5 shows a Type 1 piping penetration assembly. The Type 2 penetration assembly is described in 3.8.6.1.2.

All piping is generally attached directly to the penetration nozzle. However, hot piping and multiple piping penetrations pass through the nozzle as Type 1 and Type 5 penetrations, respectively. The Type 5 penetration is described in 3.8.6.1.5.

Type 1 penetration assemblies which penetrate the primary containment vessel consist of:

a. Penetration nozzle

b. Flued head fitting

c. Process pipe

d. Guard pipe, when required

In all Type 1 penetrations, containment closure is accomplished by means of the flued head fitting. The flued head fitting is located at the outer end of the nozzle, guard pipe and process pipe, at a suitable distance external to the primary containment vessel and biological shield wall. At this location, the flued head fitting is welded to the nozzle. With the exception of the main steam penetrations, The portion of the process pipe, which is within the penetration assembly, is an integral part of the forged flued head fitting thus eliminating pipe welds which would be inaccessible for in-service inspection. (see Figure 3.8-5.) [The main steam flued heads are constructed from a flued head forging welded to a length of process pipe such that the welded assembly becomes a Type 1 penetration. (See Figure 3.8-10, Note 7.) The flued head to process pipe weld is accessible for inservice inspection.]

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 ~~The~~ inservice inspection of welds ~~at~~ in high energy (hot type) containment penetration assemblies ~~piping~~ is described in 3.8.6.1.1.2. All required inservice inspection locations are accessible.

3.6.2.1.3 Postulated 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.

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

Q. 110.011
(3.6.2)

Indicate in 3.6.2.1.3 of the FSAR, the criteria for postulating cracks in: (1) ASME Class 1 piping designated as moderate energy lines; and (2) piping not designated to Seismic Category I criteria but which are designated as moderate energy lines.

Response:

Please see revised 3.6.2.1.3.*

*See attached draft page changes.

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.

The inservice inspection of welds at containment penetration piping is described in 3.8.6.1.1. All required inservice inspection locations are accessible.

3.6.2.1.3 Postulated 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.

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

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:

(1) If the result of a detailed stress analysis indicates that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated.

(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.

Q. 110.012
(3.6.2)

Indicate in 3.6.2.1.4.1.e (1) and (2) of the FSAR, how a consideration of the maximum stress range is used to exempt certain break orientations when the postulated break is due to a usage factor in excess of 0.1.

Response:

The rules set forth in 3.6.2.1.4.1.e (1) and (2) exempt certain break orientations based solely on stress and are independent of calculated cumulative usage factor. This is in accordance with Branch Technical Position MEB 3-1.

Q. 110.013
(3.6.2)

Indicate in 3.6.2.1.4.1.f of the FSAR where the postulated breaks are located with respect to the fittings.

Response:

Please see revised 3.6.2.1.4.1.f and 3.6.2.1.4.1.i.*

*See attached draft page changes.

(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. ~~unless analytical methods can conservatively demonstrate justification for the use of a smaller break area based on a mechanistic approach.~~

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

Q. 110.014
(3.6.2)

Describe in 3.6.2.1.4.1.g of the FSAR the "mechanistic approach" which you propose to justify longitudinal breaks having a break area less than the flow areas of the pipe.

Response:

The analysis performed to date assumes the longitudinal break area to be equal to the cross-sectional flow area of the pipe, and the use of a smaller break area based on a mechanistic approach is not presently contemplated.

See revised 3.6.2.1.4.1.g.*

*See draft page change with the response to Question 110.013.

Q. 110.015
(3.6.2)

Describe in 3.6.2.1.5 of the FSAR, the criteria for providing protection for safety-related structures, systems and components which might be subject to jet impingement from postulated cracks.

Response:

The protection criteria relating to the jet impingement effects of high energy system failures are presented in 3.6.2.3.2. See revised 3.6.2.1.5.*

*Draft FSAR page change attached.

c. Piping for which the internal energy level associated with the whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the presssure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for, in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe will be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size, and of equal or heavier wall thickness.

For further discussion of pipe whip protection, see 3.6.2.3.3.

3.6.2.2 Analytic Methods to Define Blowdown Forcing Functions and Response Models

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces which can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors. The methods used to calculate the reaction forces for various piping systems are presented in the following sections.

A rise time, not exceeding one millisecond, is assumed for the initial pulse of the fluid blowdown forcing function, unless longer crack propagation times or rupture opening times are substantiated by experimental data or analytical theory.

Blowdown forcing functions are determined by either of two methods given below:

a. The predicted blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 3.6-3. These may be simply described as follows:

Insert

Insert as new paragraph:

Protection of essential systems from the effects of jet impingement is provided where necessary to ensure reactor shutdown to a safe cold condition and to limit the release of radioactivity to within 10CFR Part 100 limits. For further discussion of criteria for protection against jet impingement, see 3.6.2.3.2.

[illegible]

Q. 110.015
(3.6.2)

Describe in 3.6.2.1.5 of the FSAR, the criteria for providing protection for safety-related structures, systems and components which might be subject to jet impingement from postulated cracks.

Response:

The protection criteria relating to the jet impingement effects of high energy system failures are presented in 3.6.2.3.2. See revised 3.6.2.1.5.*

*Draft FSAR page change attached.

c. Piping for which the internal energy level associated with the whipping is insufficient to impair the safety function of any structure, system, or component to an unacceptable level. Any line restrictions (e.g., flow limiters) between the presssure source and break location, and the effects of either a single-ended or double-ended flow condition are accounted for, in the determination of the internal fluid energy level associated with the postulated pipe break reaction. The energy level in a whipping pipe will be considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size, and of equal or heavier wall thickness.

For further discussion of pipe whip protection, see 3.6.2.3.3.

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Blowdown forcing functions are determined by either of two methods given below:

a. The predicted blowdown forces on pipes fed by a pressure vessel can be described by transient and steady-state forcing functions. The forcing functions used are based on methods described in Reference 3.6-3. These may be simply described as follows:

Insert

Insert as new paragraph:

Protection of essential systems from the effects of jet impingement is provided where necessary to ensure reactor shutdown to a safe cold condition and to limit the release of radioactivity to within 10CFR Part 100 limits. For further discussion of criteria for protection against jet impingement, see 3.6.2.3.2.

Q. 110.016
(3.6.2)

Provide assurance in 3.6.2.2.3.3 of the FSAR, that the mechanical strain in the energy absorbing, flexural support members does not exceed half of the ultimate uniform strain of the materials under the design loads using your design procedures.

Response:

Limiting mechanical strain to 50% of the ultimate uniform strain is appropriate for pure tension members, as discussed in 3.6.2.2.3.2, but not for flexural support members. Deformation of energy absorbing flexural supports is limited by design criteria to 50% of μ_c (5% for members in direct contact with the primary containment) as discussed in 3.6.2.2.2.2.C.(2). Limiting the flexural deformation to 50% of that deformation which corresponds to flexural collapse, provides a realistic factor of safety for these members, which includes consideration of ultimate uniform strain, as well as local member instability due to crimping and buckling.

Q. 110.017

Indicate in Sections 3.6.2.3.2 and 3.6.2.5 of the FSAR, whether: (1) the environmental effects of postulated pipe breaks (i.e., pressure, temperature, humidity, wetting of exposed equipment and flooding), have been considered in the design of the WNP-2 facility; and (2) these environmental effects are at least as severe as those associated with a postulated crack of the same size as the postulated break.

Response:

The environmental effects of postulated high and moderate energy fluid piping systems failures are being addressed in the redundancy studies currently undergoing updating. A conclusion of the study will be provided at a later date. ~~See the response to Questions 110.033 and 110.003~~.

Q. 110.018
(RSP)

Expand Table 3.6-4 to include the control rod drive hydraulic piping system and the condensate piping system (i.e., the piping which runs between the condensate pump discharge and the condensate demineralizers).

Response:

Neither the CRD nor the condensate piping systems should be included in FSAR Table 3.6-4 entitled "High Energy Fluid Systems Outside Primary Containment" as neither system is high energy as indicated in the reply to NRC Question 010.014.

The CRD and condensate piping systems are included as moderate energy systems in FSAR Table 3.6-5.

Q. 110.019
RSP
(3.9.2)

Previous analyses of other nuclear plants have shown that certain reactor system components and their supports may be subjected to asymmetric loads which are higher, on a conservative basis, than the loads estimated in the original analyses. These asymmetric loads could result from postulated ruptures of the reactor coolant piping at various locations. Accordingly, it is our position that you assess the capability of these reactor system components of the WNP-2 facility, including their supports, to provide assurance that the calculated dynamic, asymmetric loads resulting from these postulated pipe ruptures will be adequately conservative (i.e., provide assurance that the reactor can be brought safely to a cold shutdown condition). The reactor system components that require this reassessment are: (1) the reactor pressure vessel; (2) the core supports and other reactor internals; (3) the control rod drives; (4) the emergency core cooling system (ECCS) piping which is attached to the primary coolant piping; (5) the primary coolant piping; and (6) the reactor vessel supports. The effects of postulated asymmetric LOCA loads on these reactor system components and the various cavity structures should be submitted, including the following information:

a. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show: (1) the geometry of all principal elements; and (2) the materials of construction.

b. If you choose to reference a generic analysis in your response to this item rather than submitting a unique analysis for the WNP-2 facility, demonstrate that the analysis of the referenced generic plant adequately bounds the postulated accidents in the WNP-2 facility. Additionally, provide a comparison of the geometric, structural, mechanical and thermal-hydraulic similarities between the WNP-2 facility and the referenced analysis. Discuss the effects of any differences between your facility and the generic plant.

c. Consider all postulated breaks in the reactor coolant piping system, including the following locations: (1) the steam line nozzles at the piping terminal ends;

(2) the feedwater nozzles at the piping terminal ends; and (3) the recirculation inlet and outlet nozzles at the recirculation piping terminal ends.

Additionally, provide an assessment of the effects of asymmetric pressure differentials on the systems and components listed above in combination with all external loadings, including the safe shutdown earthquake (SSE) loads and other faulted condition loads, for the postulated pipe breaks described in Item (c) above. These pressure differentials are the blowdown jet forces at the location of the rupture (i.e., the reaction forces), transient differential pressures in the annular region between any affected component and its cavity wall, and transient differential pressures across the core barrel within the reactor pressure vessel. This assessment may utilize the following mechanistic effects as applicable: (1) the limited displacement break areas; (2) the effect of fluid-structure interaction; (3) the actual time-dependent forcing function; (4) the reactor support stiffness; and (5) the break opening times.

The following information should also be submitted:

d. If the assessment in Item (c) above indicates that loads could be imposed on the affected reactor system components which would exceed the elastic limit of the materials in these components and their supports or which could cause displacements that exceed previous design limits, provide: (1) an evaluation of the inelastic behavior of the affected material, including a consideration of the strain hardening of the material; and (2) the consequent effect on the loads transmitted to the backup structures to which these systems are attached.

e. For all the analyses performed in responding to this request, indicate the method of analysis and provide: (1) the structural and hydraulic computer codes employed; (2) schematic drawings of the models; (3) comparisons of the calculated stresses, strains and deflections with the allowable values; and (4) the basis for selecting the allowable stresses and strains.

f. Demonstrate that all safety-related active components will function properly when subjected to the loads resulting from a postulated LOCA in combination with the SSE loads.

g. Demonstrate the functional capability of the safety-related piping when subjected to the loads resulting from a postulated LOCA in combination with the SSE loads.

Response:

a. Installation arrangement of the reactor pressure vessel bearing plate is shown in Figure 3.8-19. The design and analysis procedures applicable to the reactor pressure vessel bearing plate as a vessel support are described in Reference 3.8-6. The reference was submitted to NRC by WPPSS letter G02-75-37, dated February 11, 1975, and approved by NRC by letter dated August 13, 1975, Docket No. 50-397. Arrangement drawings of the reactor vessel support system except the RPV bearing plate are listed below and will be provided upon request after NRC review of the information submittals committed to in (c) through (g).

Drawing	VPF #
Vessel Outline	3133-1
Skirt Knuckle Details	3133-19
Skirt and Base Plate	3133-20
Bottom Head Assembly	3133-23
#1 Shell Ring	3133-30
#2 Shell Ring	3133-31
#3 Shell Ring	3133-32
#4 Shell Ring	3133-33
Shell Flange	3133-35
Top Head Flange	3133-39
Top Head	3133-41
Stabilizer Bracket	3133-104

b. A unique analysis for the WNP-2 facility will be provided for the reactor system components as described below.

c-g. Responses to these parts of this question are being developed under the NSSS New Loads Design Adequacy Evaluations for the WNP-2 facility, and will be submitted in future revisions to the WNP-2 "Plant Design Assessment for SRV and LOCA Loads".

Q. 110.020
(3.9.1)

Provide your basis for placing the Operating Basis Earthquake (OBE) in the "Emergency" category in Section 3.9.1.1.3 of the FSAR. Since continued operation of the plant after the OBE is required by Section III(d) of Appendix A to 10CFR100, provide justification for the exclusion of fatigue considerations in the design of essential components (e.g., hydraulic control unit).

Response:

The Operating Basis Earthquake (OBE) was analyzed as normal to upset condition and 10 cycles associated with this event were considered for the fatigue evaluation of the Hydraulic Control Unit. See revised Section 3.9.1.1.3.*

*Revised draft FSAR page attached.

3.9.1.1.3 Hydraulic Control Unit Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40 year life and the Hydraulic Control Unit are as follows:

	Transient	Category	Cycles
a.	Normal startup & shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure tests	normal/upset	10
d.	Scram tests (cold)	normal/upset	300
e.	Operational scrams (hot)	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Drive cycles	normal/upset	1000
h.	Scram with stuck scram discharge valve	normal/upset	1
i.	OBE[Δ]	~~emergency~~ normal/upset	~~1~~ 10
k.	SSE	faulted	1

[Δ] ~~The frequency of this cycle would indicate emergency category. However, for conservatism, this OBE condition was analyzed as upset but without fatigue considerations.~~
See attached insert.

Insert:

The frequency of occurrence of this event would indicate emergency category. However, for conservatism, this event was analyzed as normal and upset condition with 10 cycles considered for fatigue evaluation.

[illegible]

Q. 110.021
(3.9.1)

Provide descriptions, including the testing procedures, of the experimental stress analysis referred to in Section 3.9.1.3.1 of the FSAR and which was conducted to verify the design adequacy of the piping seismic shock suppressors.

Response:

See revised FSAR section 3.9.1.3.1 and 3.9.3.4.*

*Draft revised FSAR pages attached.

3.9.1.3.1 Experimental Stress Analysis of Piping Components

The following is ~~a list of~~ the only NSSS component~~s~~ upon which experimental stress analysis was used. ~~These~~ This component~~s~~ ~~have~~ has been tested to verify ~~their~~ its design adequacy:

a. ~~Piping seismic shock suppressors~~

~~b.~~ Pipe whip restraints

Descriptions of the ~~support and~~ whip restraint tests are contained in section~~s~~ ~~3.9.3.4 and~~ 3.6. ~~respectively.~~

3.9.1.3.2 Orificed Fuel Support, Vertical and Horizontal Load Tests

The orificed fuel support experimental stress analysis is discussed in 3.9.1.4.2.5.

3.9.1.4 Considerations for the Evaluation of Faulted Conditions

All Seismic Category I equipment is evaluated for the faulted loading conditions. However, emergency stress limits rather than faulted stress limits were used in many cases. In essentially all cases, actual stresses are within elastic limits. The following paragraphs in 3.9.1.4 show examples of the treatment of faulted conditions for the major components on a component by component basis. Additional discussion of faulted analysis can be found in 3.9.3, 3.9.5, and Table 3.9-2.

3.9.2.2 and 3.7 discuss the treatment of dynamic loads resulting from the postulated SSE. 3.9.2.5 discusses the dynamic analysis of loads affected on NSSS equipment resulting from blowdown. Deformations under faulted conditions have been evaluated in critical areas. In all cases the identified design limits, such as clearance limits, are not violated.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Control Rod Drives

The ASME-III Code components of the CRD have been analyzed for abnormal conditions h and i shown in 3.9.1.1.1. The loads and stresses are within the elastic limits of the material.

~~b.~~ a. Spring Support

The design load on spring supports is the load caused by dead weight. The supports are calibrated to ensure that they support the design load at both their hot and cold settings. Spring supports provide a specified down travel and up travel in excess of the specified thermal movement.

~~c.~~ b. Snubbers

The design load on snubbers includes those loads caused by seismic forces (operating basis earthquake and safe shutdown earthquake) system anchor movements and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

The snubbers were designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions. They are designed to be able to carry the load under normal, upset, emergency, and faulted loading conditions.

~~The snubbers were also tested dynamically to ensure that they can perform as required in the following manner:~~

~~(1) The snubbers were subjected to either force or displacement that varies approximately as the sine wave.~~

~~(2) The frequency (Hz) of the input motion or force was verified at small increments within the specified range.~~

~~(3) The resulting relative displacements and corresponding loads across the working components including end attachments, were recorded.~~

~~(4) The test was conducted with the snubber at various temperatures.~~

See attached page for insert

Insert to Page 3.9-70:

Two snubbers of each size and each model were tested under upset and faulted loads in the manner described below:

A) Snubbers were tested dynamically to insure that they could perform as required under upset loading condition in the following manner:

1. The snubbers were subjected to a force that varied approximately as the sine wave.

2. The frequency (Hz) of the input force was in increments of 5 Hz within the range of 3 to 33 Hz.

3. The test was conducted with the snubber at room temperature and at 200°F.

4. The peak load in both tension and compression was equal to or higher than the rated load of the snubbers.

5. The duration of the test at each frequency was 10 seconds or more.

See attached page for insert.

~~(5) The peak load in both tension and compression was equal to or higher than the rated load.~~

~~(6) The duration of the test at each frequency was specified.~~

~~The snubber will be tested dynamically at a frequency within a specified frequency range and at a minimum specified temperature for the faulted load. Test duration will be specified. Snubbers will be tested for various abnormal environment conditions.~~

Upon completion of the above abnormal environmental transient test, the snubber shall be tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.

~~d.~~ c. Rigid Supports

The design load on rigid supports includes those loads caused by dead weight, thermal expansion, primary secondary forces, i.e., operating basis earthquate (OBE) and safe shutdown earthquake (SSE), system anchor displacements, and reaction forces caused by relief valve discharge, turbine stop valve closure, etc.

Rigid supports are designed in accordance with NF-3000 to be capable of carrying the design load for all operating conditions.

3.9.3.4.2 ECCS Pumps

The HPCS, LPCS, and RHR pumps have been tested in the shop and will be tested as defined in 3.9.3.2. These tests prove the adequacy of the support structure for the pump assembly under operating conditions. Furthermore, the stress calculation summary provided in 3.9.3.1 defined the stress levels in the critical support areas, namely, the pressure boundary parts and non-pressure boundary parts. The stress level margins prove the adequacy of the equipment.

Insert to Page 3.9-71:

B) Snubbers were tested dynamically to insure that they could perform as required under emergency and faulted loading conditions in the following manner:

1. The snubbers were subjected to a force that varied approximately as the sine wave.

2. The test was conducted with the snubbers at room temperature.

3. The peak load in both tension and compression was equal to 1.5 times the rated load of the snubbers.

4. The duration of the test was 10 seconds.

[illegible]

Q. 110.022
(3.9.2)

Supplement the preoperational piping vibration test program described in Section 3.9.2.1 of the FSAR with detailed information in the manner discussed in Section 3.9.2 of the Standard Review Plan (SRP), NUREG-75/087. In your response, emphasize the measures you will take to conduct visual inspections and measurements of vibration. In addition to the piping discussed in Section 3.9.2.1 (i.e., the recirculation piping and the RHR suction piping), include the following piping systems in your response: (1) all safety-related ASME Class 1, 2 and 3 piping systems; (2) other high energy piping systems inside seismic Category I structures; (3) those portions of high energy systems whose failures could adversely affect the functioning of any safety-related structure, system or component; and (4) the seismic Category I portions of moderate energy piping systems located outside containment.

Response:

See revised 3.9.2.1.1, 3.9.2.1.2, 3.9.2.1.3, and 3.9.2.1.7.*. These modified sections describe and clarify the Preoperational and Startup Piping Vibration Program. The preopoerational program includes all the piping systems described in items (1) through (4) in the question. Note that during the preoperational program, all systems contained in the preoperational program described in Chapter 14 are operated at rated flow condition and the piping systems are visually inspected. The exceptions to these are the feedwater, main steam, recirculation, RWCU and RCIC piping systems which cannot be operated at rated conditions until the startup program. Therefore, these systems are specifically included in the Piping Vibration Startup Test which will use remote monitoring equipment located in the drywell to measure piping vibration and expansion in these systems. The portion of the piping systems located outside the drywell will be visually inspected during initial operation and conditions listed in 3.9.2.1.3.

*Draft revised FSAR page changes attached.

whose failure would degrade an essential component is defined in 9.1 and is classified as Seismic Class I. These components were subjected to an elastic dynamic finite element analysis to generate loadings. This analysis utilizes appropriate seismic floor response spectra and combines loads at frequencies up to 33 HZ in three directions. Imposed stresses were generated and combined for normal, upset, and faulted conditions. Stresses were compared, depending on the specific safety class of the equipment, to industrial codes, ASME, ANSI or industrial standards, AISC allowables.

3.9.1.4.13 Balance of Plant Equipment

With the exception of pipe whip restraint design, the faulted condition was evaluated in accordance with ASME Section III by elastic systems and components analysis. Inelastic stress analysis methods were not utilized for design of any of these components. Pipe whip restraint design is described in 3.6.2.

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

The test program is divided into three phases: preoperational vibration, startup vibration, and operation transients.

3.9.2.1.1 Preoperational Vibration Testing

~~The purpose of the preoperational vibration test phase is to verify that operating vibrations in the recirculation and RHR suction piping are within acceptable limits. This phase of the test uses visual observation to supplement remote measurements. If, during steady state operation, visual observation indicates that vibration is significant, measurements are made with a hand held vibrograph. Visual observations, manual and remote measurements are made during the following steady state conditions:~~

a. ~~Recirculation pumps minimum flow;~~

b. ~~Recirculation pumps at 50% of rated flow;~~

Insert to Page 3.9-22:

During the preoperational test phase it is verified that operating vibrations in all piping systems included in the preoperational test program are within acceptable limits. This phase of the test uses visual observation. If, during the initial system operation, visual observation indicates that piping vibration is significant, measurements are made with a hand-held vibrograph. The results of those measurements will be reviewed by the appropriate engineering group to determine the acceptability of the measured vibration values. If the measured vibration values are not acceptable, appropriate design modifications will be made and the system retested. Visual observations are made during initial operation of all piping systems. During the preoperational test program described in FSAR Chapter 14.2, all systems with the exception of the recirculation, main steam, RCIC, feedwater and RWCU piping are operated at rated system flow condition. These remaining piping systems are monitored and/or visually inspected during the startup program. Refer to 3.9.2.1.3 and 14.2.12.3.33.

~~c. Recirculation pumps at 75% of rated flow;~~

~~d. Recirculation pumps at 100% of rated flow;~~

~~e. RHR suction piping at 100% of rated flow in the shutdown cooling mode.~~

3.9.2.1.2 Small Attached Piping

During visual observation ~~of each of the above test conditions, (a) through (e)~~, special attention is given to small attached piping and instrument connections to ensure that they are not in resonance with the recirculation pump motors or flow induced vibrations. If the operating vibration acceptance criteria are not met, ~~the owner will be requested to take~~ appropriate corrective action, will be taken and retesting performed. ~~No analysis of small piping will be done~~.

3.9.2.1.3 Startup Vibration

The purpose of this phase of the program is to verify that the main steam, recirculation, RWCU, and RCIC steam piping vibration are within acceptable limits. Because of limited access due to high radiation levels, ~~no visual observation~~ remote monitoring is required during this phase of the test. Remote measurements are made during the following steady state conditions: Insert sentence from attached page.

a. Main steam flow at 25% of rated;

b. Main steam flow at 50% of rated;

c. Main steam flow at 75% of rated;

d. Main steam flow at 100% of rated;

e. Recirculation flow at minimum, 50%, 75%, and 100% of rated along the 100% load li

~~e~~ f. RCIC turbine steam line at 100% of rated;

~~f~~ g. RHR suction piping at 100% of rated flow in the shutdown cooling mode.

The piping vibration startup test is described in 14.2.12.3.33.

3.9.2.1.4 Operating Transient Loads

The purpose of the operating transient test phase is to verify that pipe stresses are within Code Limits. The

Insert to Page 3.9-23:

However, during the initial nuclear heat-up to rated temperature and pressure visual inspection of major drywell piping systems will be performed in conjunction with the thermal expansion program to confirm acceptable vibration levels.

If the test measurements indicate failure to meet Level 2 criteria, the following corrective actions are taken after completion of the test:

a. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds limits, the source of the vibration must be identified. Action, such as suspension adjustment is taken to correct any discrepancies.

b. Instrumentation Inspection. The instrumentation installation and calibration are checked and any discrepancies corrected.

c. Repeat Test. If (a) and (b) above identify a malfunction or discrepancy that could account for failure to comply with Level 2 criteria and appropriate corrective action has been taken, the test is repeated.

d. Documentation of Discrepancies. If the test is not repeated, the discrepancies found under actions (a) and (b) above are documented in the test evaluation report and correlated with the test condition. The test is not considered complete until the test results are reconciled with the acceptance criteria.

3.9.2.1.7 Measurement Locations

Remote ~~shock~~ expansion and vibration measurements are made in the three orthogonal directions ~~near the first downstream S/R valve on each steam line; and in the three orthogonal directions on the piping between the recirculation pump discharge and the first downstream valve.~~. During preoperational testing prior to fuel load, visual inspection of all piping is made, and any visible vibration measured with a handheld instrument.

Insert attached page.

Insert to Page 3.9-26:

at appropriate locations on the main steam, recirculation, feedwater, RCIC, RHR, SRV discharge, and RWCU piping. The exact locations are not finalized but will be documented in the Startup Vibration Test Procedure described in 14.2.12.3.33.

For each of the selected vibration remote measurement locations, Level 1 and 2 deflection and ~~acceleration~~ limits are prescribed in the startup test ~~specification~~ procedure. Level 2 limits are based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits are based on maximum allowable Code stress limits.

3.9.2.1.8 RCIC Pump Assembly

The RCIC pump construction is a barrel type on a large cross-section pedestal. Qualification by analysis was performed. The seismic design analysis is based on 3g horizontal and 1g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal. The pump mass, support system and accessory piping have been shown by analysis to have a natural frequency greater than 33 Hz.

The RCIC pump assembly has been analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90% of the allowable.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

This subsection describes the criteria for seismic qualification of mechanical safety-related equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component by component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment was qualified as a unit, for example, motor powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in 3.10 and 3.11. Seismic qualification testing is also discussed in 3.9.3.2 and 3.9.3.5. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in 3.10.

Q. 110.023
(3.9.2)

In Section 3.9.2.4 of the FSAR, you indicate that you will perform a nonprototypical preoperational flow test in the WNP-2 facility to determine whether there are any vibrational effects (i.e., wear or loose parts). The basis you provide for this approach is what the internal design configurations of the WNP-2 facility are substantially similar to the prototype BWR/4 plants. You further indicate that the effects of the only design change made in the jet pumps would be verified by the vibration measurement and inspection programs in the Tokai-2 facility. Accordingly, provide the Tokai-2 jet pump test data. If such information cannot be provided in a timely manner or in the event that the Tokai-2 information is not acceptable, we will require that you classify the WNP-2 facility as prototypical in accordance with the guidance contained in Regulatory Guide 1.20, Revision 2, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing", May 1976, and perform the following tests and analysis:

a. Make instrumented measurements at the jet pumps and at the shroud head during cold-flow, precritical and startup tests.

b. Provide an analysis to verify that the BWR/4 data and the measurements made in Item (a) above provide assurance that the internals of a BWR/5 facility will not be adversely affected by flow-induced vibrations.

Response:

Details of the Tokai-2 test results are available in Licensing Topical Report NEDE-24057-2-P, "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (Amendment No. 2)", issued June 1979.

Tokai-2 test results show that vibration amplitudes of the jet pumps are within acceptable limits and showed vibration characteristics similar to those observed in BWR/4 plants.

In addition to the NEDE Report, please refer to revised 3.9.2.4 in the WNP-2 FSAR for a more recent description of the WNP-2 plans.*

*Draft revised FSAR page change attached.

e dynamic modal analyses also form the basis for interpretation of the prototype plant preoperational and initial startup test results (3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±10,000 psi.

3.9.2.4 Confirmatory Flow-Induced Vibration Testing of Reactor Internals

Reactor internals for WNP-2 are substantially the same as the internals design configurations which have been tested in prototype BWR/4 plants. The only exception is the jet pumps, which are of the BWR/5 design. A vibration measurement and inspection program has been conducted in the Tokai-2 plant, to verify the design of the jet pumps with respect to vibration. Results can be made available for NRC review after completion of the data analysis and interpretation.

A comprehensive vibration assessment of BWR/4 and BWR/5 internals is presented in a Licensing Topical Report (reference 3.9-6). This report also contains additional information on the jet pump vibration measurement and inspection programs rformed in the Tokai-2 plant.

WNP-2 reactor internals will be tested in accordance with provisions of Regulatory Guide 1.20, Revision 2 for non-prototype, category IV plants using Tokai-2 as the limited valid prototype. The test procedure will require vibration measurements to determine ~~the vibration measurements to determine~~ the vibration characteristics of reactor vessel internals during the initial approach to full power operation. Vibratory responses are recorded at various power levels and recirculatory flow rates using accelerometers on the shroud head assembly and strain gages on two selected jet pump riser pipe braces.

[illegible]

Q. 110.024
(3.9.2)

In the first amendment of the General Electric topical report, NEDE-24057, "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants", (Reference 3.9-6 of the FSAR), it is implied in the third sentence of the response to request no. 3.a. that feedwater spargers with a triple thermal sleeve and top-mounted discharge nozzles will be used in the WNP-2 facility. The report also indicates that spargers of this type have performed satisfactorily with respect to vibrations during testing. On this basis, the cited report indicates that no additional vibration measurements are planned. Accordingly, provide detailed information in Section 3.9.2.3 of the FSAR regarding these sparger tests to demonstrate that these tests are applicable to the WNP-2 facility. This information should include descriptions of the test setup, the testing procedures, and the test results.

Response:

Triple thermal sleeve feedwater spargers are not used in the WNP-2 facility. Instead, welded-in spargers such as in Hatch, Brunswick, and Tokai-2 are used in WNP-2. Vibration tests at these earlier plants have shown that vibration amplitudes of the welded-in spargers are well below acceptable limits. The Licensing Topical Report NEDE-24057-2-P, "Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (Amendment No. 2)", issued in June 1979, provides a list identifying the type of spargers used in each plant.

Q. 110.025
(3.9.3)

Section III of the ASME Boiler and Pressure Vessel Code defines the primary stress limit under design condition for the combination of membrane plus bending stress rather than the stress limits for bending alone. Accordingly, revise Section 3.9.3.1 of the FSAR to include the membrane plus bending stress limits in the design of the pumps in the reactor core isolation cooling system (RCIC), the ECCS and the standby liquid control systems (SLCS).

Response:

The membrane plus bending stress limits of the ASME code were satisfied in the design of the RCIC pump, SLC pump, and ECCS pumps. The primary stress design limit for the combination of local membrane stress plus bending stress is 150% of that allowed for the general membrane stress. Accordingly, Section 3.9.3.1 of the FSAR has been revised.*

*See attached draft FSAR page changes.

d. Faulted, or emergency conditions include:

Design pressure
Design temperature
Safe shutdown earthquake
Inlet and exhaust piping nozzle loads

Stress limits for pressure boundary are 120% of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending (plus local membrane).

e. Nozzle loading definition includes:

Upset - Inlet F = (3500 - M)/3
Exhaust F = (7000 - M)/3

Faulted (or Emergency) - Inlet F = (4200 - M)/3
Exhaust F = (8400 - M)/3

Where F (lbs) and M (ft-lb) are the resultant force and moment on the respective nozzle.

Table 3.9-2(q) contains a summary of the RCIC turbine components calculated and allowable loads.

3.9.3.1.10 RCIC Pump

of the 1971 Edition, Winter 1971 Addenda of the
The RCIC pump has been designed and fabricated to the requirements ~~for an~~ ASME Code, Section III as a Class 2 component. As such, the following information is offered:

The RCIC pump is tested in conjunction with the RCIC turbine under operating conditions. A monthly operation test is performed where the RCIC pump takes condensate from the aboveground storage tank and at design flow discharges condensate back to the aboveground storage tank via a closed test loop.

Design conditions for the RCIC pump include:

a. Available NPSH - 21 feet

b. Total head - High speed 2850 feet
Low speed 610 feet

c. Constant flow rate 625 gpm

d. Normal ambient operating temperature - $60^{o}F$ to $100^{o}F$

e. Normal plus Upset conditions which control the pump design include:

Design pressure	- 1500 psig
Design temperature	- $40^{o}F$ - $140^{o}F$
Operating basis earthquake	- 1/2 of SSE
Suction nozzle loads	- Fo = 1940 lbs, Mo = 2460 ft-lbs
Discharge nozzle loads	- Fo = 3715 lbs, Mo = 4330 ft-lbs

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane and 1.5S for bending plus ~~(~~local membrane~~)~~.

f. Faulted, or Emergency conditions include:

Design pressure	- 1500 psig
Design temperature	- $40^{o}F$ - $140^{o}F$
Safe shutdown earthquake:	- Horizontal - 3.0g Vertical - 0.5g
Suction nozzle loads	- Fo = 3715 lbs, - Mo = 4330 ft-lbs
Discharge nozzle loads	- Fo = 4450 lbs, Mo = 5200 ft-lbs

Stress limits for pressure boundary are 120% of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending plus ~~(~~local membrane~~)~~.

a. Normal plus Upset condition:

Design pressures are tabulated above. The operating basis earthquake, seismic accelerations are 1.5g horizontal and 0.5g vertical. Stress limits for pressure boundary are ASME Section III Code allowable stress (1.0S) for general membrane and 1.5S for bending ~~(~~plus local membrane~~)~~.

b. Faulted or Emergency condition:

Design pressures are tabulated above. The safe shutdown earthquake, seismic accelerations are 3g horizontal and 1g vertical. Stress limits for the pressure boundary are 120% of ASME Section III Code allowable stress (1.2S) for general membrane and 1.8S for bending ~~(~~plus local membrane~~)~~.

Table 3.9-2(n) summarizes the design calculations for the ECCS pumps.

3.9.3.1.12 Standby Liquid Control Pump

The standby liquid control pump has been designed and fabricated following the requirements for an ASME Code, Section III for Class 2 component. As such, the following information is offered:

The SLC pump and motor are ~~functional~~ functionally tested by pumping demineralized water through a closed test loop under operating conditions. The SLC pump is capable of injecting the net contents of the storage tank into the reactor in not less than 50 minutes and not more than 125 minutes. The pump is capable of injecting flow into the reactor against pressure up to the set point of the reactor relief valves.

Design conditions for the SLC pump include:

a. Flow rate — 43 gpm

b. Available NPSH, ~~maximum~~ Minimum — 12.9 psi

c. Maximum operating discharge pressure — 1220 psig

d. Ambient conditions:

Temperature	$70^{\circ}F - 100^{\circ}F$
Relative Humidity	20% - 95%

e. Normal plus upset conditions which control the pump design include:

Design pressure	1400 psig
Design temperature	$150^{\circ}F$
Operating basis earthquake	1/2 of SSE
Suction nozzle loads	Fo = 770 lbs Mo = 490 ft-lbs
Discharge nozzle loads	Fo = 370 lbs Mo = 110 ft-lbs

Stress limits for pressure boundary are ASME Code allowable stress (1.0S) for general membrane, and 1.5S for bending plus local membrane.

f. Faulted or emergency conditions include:

Design pressure	1400 psig
Design temperature	150°F
Safe shutdown earthquake	Horizontal 3.0g Vertical 0.5g
Suction nozzle loads	Fo = 920 lbs Mo = 590 ft-lbs
Discharge nozzle	Fo = 440 lbs Mo = 130 ft-lbs

Stress limits for pressure boundary are 120 percent of ASME Code allowable stress (1.2S) for general membrane and 1.8S for bending plus ~~(~~local membrane~~)~~.

g. Nozzle loading:

Pump nozzles are subject to loading from the connecting pipe. The nozzle pipe reactions to the allowable forces and moments on the equipment is expressed as:

$$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$$

where Fi = The largest absolute value of the three actual external orthogonal forces (Fx, Fy, Fz) that may be imposed by the pipe; and

Mi = The largest absolute value of the three actual external orthogonal moments (Mx, My, Mz) permitted from the pipe when they are combined simultaneously for a specific condition.

TABLE 3.9-2 (n)

ECCS PUMPS

The following is a summary of the design calculations on pump components:

	Calculated Stress (psi)			
	RHR ~~RHP~~	LPCS	HPCS	Allowable psi)
Pressure Boundary Parts				
Suction shell	18756	11025	11345	21000
Discharge nozzle	8040	8040	12060	17500
Suction nozzle	27383	14246	14248	27000
Torispherical head of shell	10365	4711	5139	17500
Stuffing box	2028	2230	7847	15000
Nozzle head lower plate	9635	2516	11582	15000
Mech. seal press. bolting	7600	7600	13660	25000
Mounting flange	11293	9838	5846	17500
Nozzle bolting	20978	15676	16545	25000
Non-Pressure Boundary Components				
Motor mounting bolting	21075	18259	12693	25000
Motor mounting flange	860	153	8946	17500

Q. 110.026
(SRSS)

We have accepted for reactor coolant pressure boundary components, the use of the square-root-of-the-sum-of-the-squares (SRSS) methodology for combining the dynamic structural responses due to LOCA and SSE loads. Our acceptance of this approach is documented in NUREG-0484, "Methodology for Combining Dynamic Responses," September 1978. At this time, we have not accepted the use of the SRSS methodology for combining responses from other combination of dynamic loads and for other components and supports. However, our review of additional applications of the SRSS methodology is continuing and we are concentrating on the proposed Kennedy-Newmark criteria. Refer to Appendix I of the GE topical report, NEDO 24010-1, "SRSS Application Criteria As Applied to Mark II Load Combination Cases," Supplement 1, October 1978. It is anticipated that we can support our position and criteria for a more general application of the SRSS methodology. Accordingly, provide a list of all components for which a combination of dynamic responses by the SRSS methodology is proposed, including a list of the dynamic loads which are combined. This listing should specifically include such loads as the OBE inertia loads, the OBE anchor point movement loads, the safety/relief valve (SRV) loads, the turbine stop valve closure loads, the Mark II pool dynamic loads, the SSE loads, and the LOCA loads, including the annulus pressurization loads.

Response:

The list of affected components including the list of dynamic loads will be provided in a future amendment to the WNP-2 "Plant Design Assessment for SRV and LOCA Loads".

Q. 110.027

Verify that the design of the WNP-2 facility complies with the resolution of the generic issues proposed by the Mark II Owner's Group. In particular, provide verification that you comply with our positions on the load cases, the structural acceptance criteria for piping and the demonstration of functional capability of the safety-related piping systems. Appendix A to Section 110 contains the load cases and the structural acceptance criteria while Appendix B to Section 110 presents the criteria for demonstrating functional capability. Note the following two clarifications to Appendix 110-A:

a. For load cases 1 and 2, all Service Level B requirements of the ASME Boiler and Pressure Vessel Code are to be met, including the requirements regarding the fatigue usage factors for Class 1 systems. The loads resulting from the initial actuation of the SRV's and the subsequent continuous suppression pool vibrations should be considered in your analysis for the number of cycles consistent with the 40-year design life of the WNP-2 facility.

b. For load case 10, SRV_x should be assumed to be those loads resulting from the actuation of a single safety-relief valve.

Response:

The degree of compliance with resolution of generic issues proposed by the Mark II Owner's Group for lead plants and NRC position on load cases, etcetera will be covered in future revisions of the WNP-2 "Plant Design Assessment for SRV and LOCA Loads".

Q. 110.028
(3.9.3)

Provide information regarding the effects of seismic sloshing loads on the safety-related piping and components in accordance with the agreement between the NRC staff and the Mark II Owner's Group that this information will be provided in the applications of each individual Mark II facility.

Response:

The analytical results and method of analysis utilized to determine seismic sloshing loads in the suppression chamber of the primary containment vessel are discussed in 3.8.2.4.3 of the FSAR, in response to FSAR Question 022.020.

~~An~~ Effects of the are seismic slosh loads for the Mark II containment design ~~is~~ presented in the Appendix of the Plant Design Assessment Report (DAR), Revision 2, in response to DAR Question M020.44 (submitted to NRC as FSAR Amendment No. 6, August 1979).

The response below uses the following references:

a. Response to FSAR Question 022.020 in FSAR Amendment No. 3.

b. Text of DAR, Rev. 2, submitted to NRC as FSAR Amendment No. 6, August 1979.

c. Table C-4, Appendix C of DAR, Rev. 2, submitted to NRC as FSAR Amendment No. 6, August 1979.

The effects of seismic sloshing loads on safety-related piping and components are minimal and do not jeopardize their function. This is demonstrated by the following analytical results:

a. The maximum displacement and velocity of the water surface due to seismic sloshing are 9.5 inches and 17.2 inches per second, respectively (from Ref. 1); these occur at the water surface, at the containment boundary and in the vertical direction. At all other points in the pool, the velocity of the water in any direction is less than 17.2 inches per second.

b. The period of water oscillation (the time required for the water to oscillate one complete cycle) is approximately 3.5 seconds (from Ref. 1), while the fundamental period of all structures in the pool is less than 0.2 seconds (from Ref. 2). As a result their is no dynamic amplification effect from the seismic sloshing load.

c. The sloshing velocity drag load on a cylindrical structure corresponding to the velocity of 17.2 inches per second is less than 0.02 psi. All submerged structures are assessed for SRV and LOCA loads in excess of 3 psi (horizontal pressure near the water surface; taken from Ref. 2). The sloshing velocity drag load of 0.02 psi is therefore negligible, in comparison.

Q. 110.029
(3.9.3)

Provide in Section 3.9.3.4 of the FSAR, the bases for the allowable buckling loads, including the allowable buckling stress, under faulted conditions for all the ASME Class 1 component supports in the nuclear steam supply system (NSSS) and the balance of plant (BOP). Provide a comparison of the calculated loads in the reactor vessel support skirt with the critical buckling loads of the skirt under the most limiting faulted loading condition. Describe your analytical techniques to determine both the calculated loads under faulted conditions and the critical buckling load of the WNP-2 support skirt. Indicate the most limiting load combination considered in the buckling analyses of the reactor vessel support skirt.

Response:

See revised 3.9.3.4 (page 3.9-69).*

*Draft revised FSAR page attached. See also the response to Question 110.021.

tance, and flow discontinuities (shock waves) are considered. This model also considers the influence of valve opening time and the effect of loop seal water contained in the up-stream valve seat.

The unbalanced transient hydraulic forcing function acting on the piping system computed from the flow model is then used to determine the transient dynamic responses of the piping structural model. Adapting the lumped-parameter method incorporated with the modal analysis of piping system, the time history modal responses are computed. Computations of maximum stress intensities for ASME Code Class 1 piping or maximum stress levels for ASME Code Class 2 and 3 piping are based on the dynamic analysis of the system.

3.9.3.4 Component Supports

~~Component supports are discussed in 5.4.14. Further elaboration is presented below.~~ Insert on next page.

3.9.3.4.1 Piping

Piping supports are designed in accordance with Subsection NF of ASME Section III. Supports are either designed by load rating ~~per paragraph NF-3260~~ or to the stress limits for Component linear supports ~~per paragraph NF-3231~~. In general, the load combinations for the various operating conditions correspond to those used to design the supported pipe. Design transient cyclic data are not applicable to piping supports as no fatigue evaluation is necessary to meet the Code requirements.

~~The design criteria and dynamic testing requirements for~~ component supports are as follows:

a. Component Supports

All components supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed. All component supports are designed in accordance with the rules of Subsection NF of the Code.

Insert to Page 3.9-69:

Component supports are discussed in 5.4.14 and further discussion of design loading combinations, design procedures and acceptability criteria is presented below:

All component supports of the linear-type for ASME Class 1 components in both the Nuclear Steam Supply System and the Balance of Plant are designed in accordance with Section III, Subsection NF, of the ASME Boiler and Pressure Vessel Code. The bases for allowable buckling loads and allowable buckling stresses are the allowable load equations and stress equations given in the 1971 Edition, Winter 1973 Addenda of the ASME Code, as referenced below:

Subsection NA, "General Requirements", Appendix XIII; "Design of Linear Type Supports by Analysis", paragraph XIII-1130; "Compression", and paragraph XIII-1190, "Combined Stresses";

Subsection NA, Appendix XIII, paragraph XIII-1300 "Stability and Slenderness and Width Thickness Ratios";

Subsection NA, Appendix F "Rules for Evaluation of Faulted Conditions"

All components supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.

Critical buckling loads, for plant faulted conditions, are determined by methods described in Appendix F, and allowable loads are limited to two-thirds of critical buckling loads.

In design of the reactor vessel support skirt as a plate and shell-type component support, the allowable compressive load was limited to 90 percent of the load which produces a stress equivalent to yield stress in the material, divided by the safety factor for the plant condition being evaluated. The safety factor for the faulted condition was 1.125. The effects of fabrication and operational eccentricity were included in stress calculations.

An analysis of the reactor pressure vessel support skirt under design basis faulted conditions shows that the support skirt meets the limit of calculated stress being less than 0.67 times the critical buckling stress, assuming that the critical buckling stress corresponds to the material yeild stress at temperature. The design basis faulted condition for this analysis included compressive loads due to the design basis maximum earthquake, overturning moments and shears due to the jet reaction load from a postulated severed pipe, and compressive effects on the support skirt from thermal and pressure expansion of the reactor vessel.

SSE loads predicted for the WNP-2 plant are less than fifty percent of the design basis maximum earthquake loads in the above analysis. Hydrodynamic event loads are in the process of development, and the most limiting load combination for the WNP-2 plant faulted condition will be defined in a future revision of the WNP-2 "Plant Design Assessment Report for SRV and LOCA Loads". It is expected that sufficient margin exists between the design basis faulted loads and the most limiting load combination which will be defined for WNP-2 plant faulted conditions to demonstrate the reactor pressure vessel support skirt is adequately designed to prevent buckling.

Q. 110.030
(3.9.3)

Provide the following additional information in Section 3.9.3.4 of the FSAR, regarding the operability assurance of snubbers:

a. Identify and tabulate all mechanical and hydraulic snubbers installed on safety-related systems.

b. Describe your methods and procedures for verifying operability of those snubbers identified in your response to Item (a) above, during the startup test program.

c. If additional snubbers are installed after plant start-up, commit to provide documentation for verifying: (1) the operability of these additional snubbers; and (2) that these additional snubbers will not interfere with the normal operation of the WNP-2 facility.

d. Provide an inservice inspection and testing program for the snubbers, including a discussion of the accessibility for: (1) their maintenance; and (2) the repair and replacement of snubbers, if required.

Response:

a. There are no hydraulic snubbers installed on safety-related systems at WNP-2. Mechanical snubbers are used exclusively. Because of normal construction problems (interferences, etc.), the list of snubbers is not yet complete. When the list is complete, a list of all safety-related snubbers will be included in 3.9.3.4 of the FSAR.

b. A startup, installation and inservice test program for snubbers has not yet been completely developed. WPPSS is monitoring the development of various O&M standards and is currently working on these procedures. The snubber inspection and testing program will be discussed in 3.9.3.4 of the FSAR. Se also 14.2.12.3.17 for a brief discussion of program requirements.

c. Same as (b) above.

d. Same as (b) above.

Both the list of snubbers and the inspection and testing program will be submitted later when the list is complete and the program developed.

Q. 110.031
(3.9.3)

Verify that the criteria for the design of pressure relief device stations conform with our positions in Regulatory Guide 1.67, "Installation of Overpressure Protection Devices", October 1973. If you adopt alternate criteria, demonstrate that your criteria provide a level of conservatism equivalent to that in the regulatory guide cited above.

Response:

The design of pressure relief device stations and associated piping is discussed in revised 3.9.3.1.14 and 3.9.3.3.3.*

*See draft revised FSAR page changes.

2/4

Fo = The allowable value of Fi when all moments are zero; and

Mo = The allowable value of Mi when all forces are zero.

A summary of the design calculations for the standby liquid control pump components is contained in Table 3.9-2(1).

3.9.3.1.13 Safety/Relief Valves and Main Steam Isolation Valves

Load combination, analytical methods, calculated stresses, and allowable limits are shown for the safety/relief and main steam isolation valves in Table 3.9-2(g) and 3.9-2(h) respectively. The RHR thermal relief valve on penetration X-20 (see Fig. 6.2-31K) is designed per ASME Section III, NB-3513

3.9.3.1.14 Safety Relief Valve Discharge Piping

3.9.3.1.14.1 Main Steam Safety Relief Valve Piping

This piping is designed in accordance with the ASME Code, Section III. Subsection ND for Class 3 piping within the drywell and Subsection NC for Class 2 piping within the suppression chamber. The load combinations and allowables are shown in Table 3.9-17.. The main steam safety relief valves relieve to closed discharge systems; Regulatory Guide 1.67 is therefore not applicable.

3.9.3.1.14.2 RHR Suction Shutdown Thermal Relief Valve Piping

~~This~~ The discharge piping for the thermal relief valve referenced in 3.9.3.1.13 relieves into the containment equipment drain system and is designed as ASTM B31.1 and Seismic Category I supported. ~~The RHR shutdown line thermal relief system discharges into the containment equipment drain system.~~ However, due to the very small discharges quantities of fluid to relieve pressure, the intent of Regulatory Guide 1.67 is not considered applicable. See Appendix C.2.

3.9.3.1.15 Reactor Water Cleanup (RWCU) System Pump

The RWCU pump is not part of a safety system and is not desegned to Seismic Category I requirements. However, an analysis was done which shows that the fundamental frequency (first mode) of the pump or the motor or the combination of

pump and motor is greater than 33 cps; hence, static analysis was used.

The static analysis considers static equilibrium forces on the equipment including the effect of seismic forces of 0.1g horizontal and 0.1g vertical. This analysis considers piping loads as well as torsional moment produced by the rotating assembly. No dynamic analysis is performed.

No experimental or inelastic stress analysis was used in the pump design.

Qualification testing of sensitive electrical/pneumatic equipment to meet performance requirements defined in Tables 3.11-1, 3.11-2 and 3.11-3 is completed.

Seismic tests have been conducted on the safety relief valves and the natural frequencies have been determined to be ≥ 33Hz. The tests also determined that the equipment remains functional during application of the specified "G" loads.

In addition to testing described above and in 3.9.2.2.2, the sensitive electrical/pneumatic equipment of the safety/relief valve has been qualified to performance requirements during and after emergency environment conditions defined in Tables 3.11-1, 3.11-2 and 3.11-3.

The MSIV and S/RV (Safety/Relief Valve) analytical qualification results are shown in Tables 3.9-2(h) and 3.9-2(g) respectively.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design criteria for all safety and relief piping ~~valves~~ are in ~~accordance with the rules in Subarticles NB-3677 and NC-3677~~ of ASME Section III, and the rules of Code Case 1569, applicable to the classification of the piping component under investigation. For ~~open~~ relief systems the design criteria and the analyses used to calculate maximum stresses and stress intensities are in accordance with Subarticles NB-3600 and NC-3600 of ASME Section III. The maximum stresses are calculated based upon the full discharge loads, including the effects of the system dynamic response, and the system design internal pressure. Stresses are determined for all significant points in the piping system including the safety valve inlet pipe nozzle and the nozzle to shell juncture.

3.9.3.3.1 Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation, to generate a time-history of the fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-3 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in 3.9.3.1. The Code stress limits, corresponding to load combinations classification as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.3.2 Open Relief Systems

The total steady state discharge thrust load for an open discharge system is expressed as the sum of the pressure and momentum forces as follows:

$\frac{F}{A} = 144 (P) + \frac{V^2 \varnothing}{g}$ where F = Total Reaction Force lbf.

A = Exit Flow Area, ft^2

P = Exit Pressure, lbf/in^2 gage

V = Exit Fluid Velocity, ft/sec

ρ = Exit Fluid Density, lbm/ft^3

g = Gravity Acceleration, 32.2 $\frac{lbm\text{-}ft}{lbf\text{-}sec^2}$

change 'ø' to 'ρ' etc.

typo

To ensure consideration of the effects of the suddenly applied loads on the valve nozzle and pipe junction, a dynamic load factor is computed. The calculation of dynamic load factor is based on modeling the valve and nozzle as a single degree of freedom dynamic system. The lumped mass of this system corresponds to the weight of the valve and nozzle and is assumed to be at the valve center of gravity. The

Regulatory Guide 1.67, Rev. 0, October 1973

Installation of Overpressure Protection Devices

Regulatory Guide Intent:

This regulatory guide describes a method acceptable to the NRC staff for implementing General Design Criteria 1 of 10CFR50, Appendix A, with regard to the design of piping for safety valve and relief valve stations which have open discharge systems with limited discharge pipes and which have inlet piping that neither contains a water seal nor is subject to slug flow of water upon discharge of the valves.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified RHR shutdown suction line thermal relief valve located between the containment isolation valves. However, the intent of the Regulatory Guide does not apply due to the very short duration of the thermal relief function and small discharge.

General Compliance or Alternate Approach Assessment:

This Regulatory Guide is not considered to be applicable to this valve due to its small size (3/4" x 1") and its very short operation time. The only purpose of the valve is to relieve the excess pressure caused by the difference of thermal expansion between the pipe and the water contained between the containment isolation valves.

Specific Evaluation Reference:

Refer to 3.9.3.~~3~~ 1.11.

Similar Application Reference:

Non-applicable.

Q. 110.032
(3.9.3)

In Section 3.9.3.2 of the FSAR, you indicate that active valves will be qualified for operability under seismic loading on a prototypical basis. We agree that a prototypical test can qualify a limited range of similar valves. However, you do not adequately describe the characteristics you consider in determining whether a valve is similar to the tested prototype valve and, therefore, can be qualified by analysis only. Accordingly, provide a discussion of how you establish the similarity of valves to a tested prototype. This discussion should include, but not be limited to, those characteristics such as valve type, size, geometry, pressure rating, stress level, manufacturer, actuator type and actuator load rating.

Response:

The active Category I valves have been qualified by seismically testing at least one valve of each design type. Many of the valves have been qualified by testing an identical valve assembly. In other cases the candidate valves are constructed using the same design parameters as the parent valve. These valves have the same manufacturer and the same general appearance and actuator type, but differ primarily in size and pressure rating.

The differences in size, pressure rating, material, and actuator load rating are being reviewed as part of our overall plan for re-evaluating seismic equipment qualifications. The basis upon which the selection of tested valves was made will be available to your Seismic Qualification Review Team (SQRT) along with the remainder of our documentation for equipment qualification.

Q. 110.033
(3.6.2)
(3.9.3)

For ASME Class 1, 2 and 3 components that could be exposed to either jet impingement loads or to pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, describe how you determine the stress levels in the targeted components. In your response, include a discussion of the structural effects throughout the targeted system from the loads cited above (i.e., those loads associated with postulated pipe breaks) in combination with other applicable loads. Provide assurance that the calculated stress levels are kept below the Service Level D limits of Section III of the ASME Code. If applicable, more conservative limits on stress levels should be imposed for active components or where piping functional capability is required.

Response:

The complete response to NRC Question 110.033 awaits the results of the ongoing pipe break and missile study, and will be presented in a future amendment to the FSAR.

At this time it is anticipated that if the availability of ASME Section III Class 1, 2 and 3 components is needed to safely shut down the plant, and if those components are exposed to postulated jet impingement loads, pipe whip impact loads, and missile loads, protective measures will be taken to preclude such loading. If it is determined that protective measures are not required on the basis that the calculated stress levels due to the postulated pipe break for missile loads in combination with other applicable loads are kept below the Service Level D limits of ASME Section III, such structural analysis will be in accordance with industry-accepted methods. If this approach is used, the methodology and results will be reported in the FSAR.

Q. 110.034
RSP
(3.9.6)

In accordance with 10CFR50.50a(g), we require submittal of your program for inservice testing of ASME Class 1, 2 and 3 pumps and valves. Our positions on this matter are presented in Section 3.9.6 of the SRP. Appendix C to Section 110 provides a suggested format for this submittal and includes a discussion of the information we require to justify any requests for relief from our positions on this matter.

Response:

In accordance with 10CFR50.55a(g), we will prepare a program for operational readiness testing of ASME Class 1, 2 and 3 pumps and valves. Our intentions are to submit this program for your review approximately one year prior to scheduled fuel load. Our program will be in agreement with your positions as presented in Section 3.9.6 of the SRP and the program will be submitted in a format similar to the one described in Appendix C to Series 110 questions.

Q. 110.035
(3.10)
(3.9.2)

It is not clear in the FSAR how the seismic analyses of seismic Category I electrical and mechanical equipment have taken into consideration the three components of the seismic accelerations. Accordingly, describe how your analyses have considered the three spatial components of seismic excitation. Regulatory Guide 1.92, Revision 1, "Combining Modal Responses and Spatial Components in Seismic Response Analysis", February 1976, provides methods acceptable to the staff for combining the responses to the three spatial components of seismic excitation.

Response:

Please refer to 3.7.2.1.8.3, 3.9.2.2, 3.9.2.2.16 and 3.10.1.2 for the information requested.*

*Draft FSAR page changes attached.

An alternative simplified dynamic analysis method is used for cold and/or limber piping systems, such as equipment drains and instrumentation lines. This method consists of applying constant horizontal and vertical load factors conservatively derived from floor response spectra. The load factors are derived by utilizing the response spectra for a conservatively chosen fundamental frequency based on the maximum span length of an assumed simply supported beam. These load factors are then increased to account for the multi-mode response of the piping system. Locations of pipe controls are chosen to limit the pipe span length to less than that length utilized in establishing the seismic load factors. These span lengths are also selected to limit the stresses and deflections to acceptably low values. The horizontal and vertical loading factors applied to the spans are combined in the same manner as described for the detailed dynamic analysis above.

3.7.2.1.8.3 Dynamic Analysis of Equipment

Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Category I equipment are given in Table 3.9-2. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response spectrum analysis is performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance of plant using the envelope of response spectra for supports. Modal maxima are combined as described in 3.7.2.1.5. The analysis is performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal or vertical excitation are considered to act simultaneously and are added directly, as described in 3.7.2.6 and 3.7.2.7. the absolute values

The relative displacements between anchors are determined from the dynamic analysis of the structures. If significant, these relative displacements are then used in a static analysis to

For each of the selected remote measurement locations, Level 1 and 2 deflection and acceleration limits are prescribed in the startup test specification. Level 2 limits are based on the results of the stress report adjusted for operating mode and instrument accuracy; Level 1 limits are based on maximum allowable Code stress limits.

3.9.2.1.8 RCIC Pump Assembly

The RCIC pump construction is a barrel type on a large cross-section pedestal. Qualification by analysis was performed. The seismic design analysis is based on 3g horizontal and 1g vertical accelerations. Results are obtained by using acceleration forces acting simultaneously in two directions, one vertical and one horizontal. The pump mass, support system and accessory piping have been shown by analysis to have a natural frequency greater than 33 Hz.

The RCIC pump assembly has been analytically qualified by static analysis for seismic loading as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are substantially less than 90% of the allowable.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment

This subsection describes the criteria for seismic qualification of mechanical safety-related equipment and also describes the qualification testing and/or analysis applicable to this plant for all the major components on a component by component basis. In some cases, a module or assembly consisting of mechanical and electrical equipment was qualified as a unit, for example, motor powered pumps. These modules are generally discussed in this paragraph rather than providing discussion of the separate electrical parts in 3.10 and 3.11. Seismic qualification testing is also discussed in 3.9.3.2 and 3.9.3.2.5. Electrical supporting equipment such as control consoles, cabinets, and panels which are part of the NSSS are discussed in 3.10.

Consideration of spatial components of seismic accelerations are taken into account in the analyses of ~~balance of plant~~ seismic Category I mechanical equipment in accordance with 3.7.2.1.8.3.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its Seismic Category I function during and after an earthquake was demonstrated by tests and/or analysis. Selection of testing, analysis or a combination of the two was determined by the type, size, shape, and complexity of the equipment being considered. When practical, the Seismic Category I operations were performed simultaneously with vibratory testing. Where this was not practical, the operation and/or loads were simulated by mathematical analysis and applied in addition to physical tests.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or static bend testing is also used to show there are no natural frequencies below 33 Hz. If a natural frequency lower than 33 Hz is discovered, dynamic tests may be conducted and in conjunction with mathematical analysis used to verify operability and structural integrity at the required seismic input conditions.

When the equipment is qualified by dynamic test the response spectrum, or time history of the attachment point is used in determining input motion. [See attached page for insert.]

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady state input of low magnitude. Seismic conditions are simulated by testing using random vibration input or single frequency input (within equipment capability) at frequencies through 35 Hz. Which ever method is used, the input motion during testing envelops the actual input motion expected during earthquake conditions.

The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.

Equipment having an extended structure, such as a valve operator, is analyzed by applying static equivalent seismic SSE loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a static bend test is used to

Insert to Page 3.9-28:

Equipment is tested in its operational mode and verified during and after the test. The tested equipment is either an exact duplicate of the supplied equipment or is representative of a family of equipment of the same design and structure.

determine spring constant and operational capability at maximum equivalent seismic load conditions.

3.9.2.2.1.1 Random Vibration Input

When random vibration input is used, the actual input motion envelops the appropriate floor input motion at the individual modes. However, single frequency input, such as sine beats, can be used provided one of the following conditions are met:

a. The characteristics of the required input motion is dominated by one frequency.

b. The anticipated response of the equipment is adequately represented by one mode.

c. The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelop the corresponding response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Motion

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa, then the input motion is applied to one direction at a time. ~~In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.~~ See attached page for insert.

3.9.2.2.1.3 Fixture Design

The fixture design will simulate the actual service mounting and cause no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Equipment testing is conducted on prototypes of the equipment installed in this plant.

Insert to Page 3.9-29:

In the case of single frequency input, the time phasing of the input is applied as follows:

a. Phase-incoherent - The time phasing of the inputs are applied in the vertical and horizontal direction such that a purely rectilinear input is avoided.

b. Phase-coherent - The vertical and horizontal inputs are applied initially in phase. The input is then applied 180° out of phase. The specimen is then rotated 90° horizontally and the in phase and 180° out of phase motion reapplied.

3.9.2.2.2.14 Main Steam Safety/Relief Valves

Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the safety/relief valve (including electrical, pneumatic devices) was dynamically tested at seismic accelerations equal to or greater than the SSE levels determined for this plant. Satisfactory operation of the valves was demonstrated during and after the test.

3.9.2.2.2.15 Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

The fuel pool cooling and cleanup system pump and motor have not been analyzed as Seismic Category I equipment since this was not a requirement of the construction permit.

3.9.2.2.2.16 Balance of Plant Safety-Related Mechanical Equipment

Balance of plant Seismic Category I equipment, components and accessories are designed based on results determined analytically (see 3.9.2.2) or through dynamic testing. ~~The appropriate dynamic tests were performed by the various contractors in accordance with the testing procedures described in IEEE Standard 344, "IEEE Guide for Seismic Qualification of Class I Elecrtic Equipment for Nuclear Power Generating Stations.~~" The dynamic tests met the seismic loading requirements as defined by the applicable floor response spectrum curves for the apporpriate damping coefficients.

replace by INSERT on next page

3.9.2.3 Dynamic Response of Reactor Internals under Operational Flow Transients and Steady State Conditions

The major reactor internal components within the vessel were subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady state conditions are not predetermined by detailed analysis. Special analyses of the response signals measured from reactor internals of similar

Insert to Page 3.9-36:

The dynamic program is performed to confirm the ability of the equipment to function as needed during and after an earthquake of magnitude up to and including the SSE. These test programs implement the criteria stated in Section 3.9.2.2, 3.9.2.2.1, 3.9.2.2.1.1, 3.9.2.2.1.2, 3.9.2.2.1.3, and 3.9.2.2.1.4.

There is no Non-Safety Class 1E equipment which, if it were spuriously actuated, could adversely affect Safety Class 1E equipment, except as noted in Table 3.10-5.

Consideration of spatial components of seismic accelerations are taken into account in the seismic analyses of ~~balance-of-plant~~ instrumentation and electrical equipment in accordance with 3.7.2.1.8.3.

3.10.1.2.1 BOP Equipment

The qualification criteria for Seismic Category I instrumentation and electrical equipment is as follows:

a. Seismic Category I instrumentation and electrical equipment is designed to withstand, without loss of nuclear safety function, Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) acceleration forces transferred to the equipment mounting location.

b. From basic input ground motion data, horizontal and vertical floor response spectra (seismic response curves) for the various buildings and elevations were developed and included in all purchase specifications for Category I equipment.

c. All equipment was qualified by analysis and/or testing in accordance with the SSE and OBE floor response spectra for the particular building location applicable to their equipment. Horizontal and vertical loads for all equipment are assumed to occur simultaneously in the most unfavorable combinations. When analysis is used, stresses from the earthquake effects are combined with stresses due to normal design loads (idle or operating conditions) so as to produce the most severe stress combination. Stresses due to normal design loads when combined with seismic stresses resulting from the OBE are maintained within allowable material working stress limits set forth in appropriate design standards and codes. Stresses due to normal design loads when combined with seismic stresses resulting from the SSE are limited to prevent loss of function of the safety related equipment. For the purposes of calculation, the "no-loss-of-function" stresses are limited to 9/10 of the minimum material yield point or as otherwise specified by appropriate standards or codes.

[illegible]

Q. 110.036
RSP
(3.9.2)
(3.10)

Hydrodynamic vibratory loadings in the suppression pool can be induced by the flow of a steam-water-air mixture into the suppression pool. This flow may result either from actuation of the safety/relief valves or from a postulated pipe break. In either case, the resultant vibrations in the suppression pool may affect structures, systems and components in other portions of the reactor building. These induced vibratory loadings can be of various magnitudes and of various frequency content for the following load cases: SRV_1, SRV_x, SRV_{ADS}, SRV_{ALL}, IBA, and DBA.

Accordingly, we require you to demonstrate that the electrical and mechnical equipment which is necessary to achieve and maintain a cold shutdown, are capable of performing their safety-related function under the most severe of the following combinations of seismic and vibratory loadings induced by the vibratory hydrodynamic loads in the suppression pool:

a. SRV_x or SRV_{ALL} (whichever is controlling) + OBE
b. SRV_x or SRV_{ALL} (whichever is controlling) + SSE
c. SRV_{ADS} + OBE + IBA
d. SRV_{ADS} + SSE + IBA
e. SSE + DBA
f. SRV_1 + SSE + DBA.

Provide a commitment that all NSSS and BOP seismic Category I mechanical and electrical equipment will be qualified for the most severe combination of seismic and hydrodynamic vibratory loadings. (Note that the applicants for operating licenses for similar facilities have stated that, in general, the load cases which include SRV_{ALL} impose the most severe hydrodynamic vibrary loadings on safety-related equipment. However, this does not preclude the possibility that other hydrodynamic loads might be limiting for particular components at the WNP-2 facility.)

Response:

Suppression pool hydrodynamic loads due to postulated IBA, DBA, SRV, SRV_x, SRV_{ADS} and SRV_{ALL} events are being developed

for WNP-2 plant, and will be provided in a future amendment to the "Plant Design Assessment for SRV and LOCA Loads." The SRV building responses will be appropriately combined with OBE, SSE, IBA and DBA building responses to provide the basis for evaluating acceptability of electrical and mechnical seismic Category I equipment orignally qualified to seismic only dynamic loading. Detailed re-evaluation of each component of seismic Category I equipment will not be performed where direct comparison of original qualification Required Response Spectra with new seismic plus hydrodynamic Required Response Spectra demonstrates satisfactory qualification of the equipment. When such comparisons cannot be made, other means of evaluating the original qualification against the new dynamic load combinations will be used. In regard to the load combination SRV_1 + SSE + DBA, WNP-2 plant design adequacy assesements for seismic Category I electrical and mechanical equipment will be performed on the generic basis established by the BWR Mark II Owner's Group for this load combination.

Q. 110.037
(3.10)
(3.9.2)

A review of the design adequacy of your safety-related electrical and mechanical equipment under seismic and hydrodynamic loadings will be performed by our Seismic Qualification Review Team (SQRT) during a site visit when this team will inspect and evaluate selected equipment. (Note that Item 110.026 of this enclosure is concerned with the structural capability of safety-related components whereas this item is concerned with the functional operability of components.) Accordingly, provide the following additional information which we will review prior to our SQRT site visit:

a. A demonstration of the adequacy of the original single-axis, single-frequency tests or, alternatively, the analyses of equipment qualified in accordance with the criteria of IEEE Standard 344-1971.

b. The qualification of equipment for the combined seismic and hydrodynamic vibratory loadings. The frequency of response due to this vibratory input may exceed 33 hertz and negate the original assumption of a component's rigidity in some cases.

Appendix D to Section 110 describes the SQRT and its procedures. Section V.2.A of this appendix indicates information which we require for our review.

Several OL applicants with similar facilities have stated in their Design Assessment Closure Reports that equipment will be qualified by testing for the required response spectra (RRS) representing the hydrodynamic and seismic loads combined by the SRSS methodology. Similarly, when qualified by analysis, these applicants have indicated that the peak dynamic responses of equipment to the hydrodynamic and seismic loads will also be combined by the SRSS methodology. For your information, we do not accept at this time, the combination of the hydrodynamic and seismic loadings using the SRSS methodology to obtain the RRS or the peak dynamic responses. Accordingly, provide a compilation of the RRS listed below for each floor of the seismic Category I buildings:

c. The RRS for the OBE or the SSE, whichever is controlling. If the OBE is controlling, explain why.

d. The controlling RRS due to hydrodynamic loads in the suppression pool.

e. Items (c) and (d) above combined using the SRSS methodology.

f. Items (c) and (d) above combined by absolute sum.

Response:

The information will be provided in a future amendment to the WNP-2 "Plant Design Assessment Report for SRV and LOCA Loads".

ATTACHMENT 1

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NUMBER 2
DOCKET NO. 50-397

ITEMIZED RESPONSE TO IE BULLETIN 79-08

ITEM 1:

Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 5a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

RESPONSE:

This item is generally related to the operations of the plant including operational procedures, maintenance procedures, training instructions, etc. Since WNP-2 is not in operation yet, presently many of the plant procedures which would cover this Item have not been prepared or are in draft form. However, a review of the information in IE Bulletins 79-05 and 79-05A has been conducted and will be considered in the preparation of procedures for WNP-2, and the training program.

The basis for implementing these procedural and training improvements will be the program established in NEDO-24708 and related documents.

ITEM 2:

Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

RESPONSE:

We have performed a review of the containment isolation design and determined that the WNP-2 design conforms to these requirements. Details of this review will be provided in our response to NUREG-0578 item 2.1.4.

ITEM 3:

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense.

RESPONSE:

WPPSS does not have an operating BWR. The responses provided below are based on WNP-2 plant design.

The auxiliary heat removal systems provided to remove decay heat from the reactor core and containment following loss of the feedwater systems are:

High Pressure Core Spray System (HPCS)
Reactor Core Isolation Cooling (RCIC) System
Low Pressure Core Spray System (LPCS)
RHR System - LPCI Mode
RHR System - Suppression Pool Spray Mode
RHR System - Suppression Pool Cooling Mode
Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI) System

The description that follows details the operation of the systems needed to achieve initial core cooling followed by containment cooling and then followed by extended core cooling for long term plant shut down, assuming the reactor is scrammed and isolated from the main condenser.

INITIAL CORE COOLING

Following a loss of feedwater and reactor scram, a low reactor water level signal (level 2) will automatically initiate main steam line isolation valve closure. At the same time this signal will put the HPCS and RCIC Systems into the reactor coolant make-up injection mode. These systems will continue to inject water into the vessel until a high water level signal (level 8) automatically trips RCIC and closes the HPCS injection valve. The HPCS pump remains running on minimum flow bypass.

Following a high reactor water level 8 trip, the HPCS injection valve will automatically reopen when reactor water level decreases to low water level 2. The RCIC System must be manually reset by the operator in the control room before it will automatically re-initiate after a high water level 8 trip.

The HPCS and RCIC System have redundant supplies of water. Normally they take suction from the condensate storage tank (CST). The HPCS and RCIC System suctions will automatically transfer from the CST to the suppression pool if the CST water is depleted or the suppression pool water level increases to a high level.

ITEM 3 (Continued)

The operator can manually initiate the HPCS and RCIC Systems from the control room before the level 2 automatic initiation level is reached. The operator has the option of manual control after automatic initiation. The operator can verify that these systems are delivering water to the reactor vessel by:

a) Verifying reactor water level increases when systems initiate (see water level discussion in response to Question 4).

b) Verify systems flow using flow indicators in the control room.

c) Verify system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to other than the reactor.

Therefore, the HPCS and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Low Pressure Core Spray (LPCS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

CONTAINMENT COOLING

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety relief valve (SRV) discharge to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in the containment/suppression pool cooling mode, or the suppression pool spray mode, i.e., RHR suction from and discharge to the suppression pool.

The operator could verify proper operation of the RHR system containment cooling function from the control room by:

a) Verifying RHR and Service Water (SW) system flow using system control room flow indicators.

b) Verify correct RHR and SW system flow paths using control room position indication of motor-operated valves.

c) On branch lines that could divert flow from the required flow paths, close the motor-operated valves and note the effect on RHR and SW flow rate.

[illegible]

ITEM 3 (Continued)

EXTENDED CORE COOLING

When the reactor has been depressurized, the RHR system can be placed in the long term shutdown cooling mode. The operator manually terminates the containment cooling mode of one of the RHR loops and places the loop in the shutdown cooling mode as follows:

1. Trip the RHR pump to be used for shutdown cooling,
2. Close associated motor-operated valve in the suppression pool suction and LPCI discharge line to the vessel,
3. Open shutdown cooling suction valves from and discharge valves to the reactor vessel, and
4. Restart the RHR pump.

In this operating mode, the RHR system can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

ITEM 4:

Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

RESPONSE:

Reactor vessel water level in the WNP-2 BWR is continuously monitored by eleven (11) indicators or recorders for normal, transient and accident conditions. In general, those monitors used to provide manual safety equipment initiation are arranged in a redundant array with two intruments, one in each of two independent electrical divisions. Thus, adequate information is provided to the operator for manual initiation of safety actions and for assurance of the vessel water level at all times.

Those sensors used to provide automatic safety equipment initiation are arranged in a four quadrant vessel tap configuration with the four sensors divided electrically between two divisions.

In addition, the operating procedures will reflect the requirements for the operators to also rely upon the information provided by the instrumentation discussed in 5(b).

The range of reactor vessel water level from below the top of the active fuel area up to the top of the vessel is covered by a combination of narrow and wide-range instruments. Level is indicated and/or recorded in the control room.

A separate set (to that described above) of narrow-range and wide-range level instrumentation on separate condensing chambers provides reactor level control via the reactor feedwater system. This set also indicates or records in the control room (three narrow-range level indicators and one wide-range level recorder).

The safety-related systems or functions served by safety-related reactor water level instrumentation are:

Reactor Core Isolation Coolant System (RCIC)
High Pressure Core Spray System (HPCS)
Low Pressure Core Spray System (LPCS)
Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI)
Automatic Depressurization System (ADS)
Nuclear Steam Supply Shutoff System (NS^4)
Reactor Protection System (RPS)
Standby Gas Treatment System (SGTS)
Emergency Power System
Secondary Containment Isolation
Main Control Room and Critical Switch Gear HVAC
Standby Service Water System
Containment Instrument Air System
Trip of Non-essential Loads

ITEM 4 (Continued)

Low reactor vessel water level is used in the initiation logic of all systems listed above. In addition, the RCIC and HPCS systems shutdown on high reactor vessel water level. HPCS will automatically restart if low reactor level is again reached. (See response to question 3 for further discussion on this.) In the case of RCIC, manual resetting is required if high reactor vessel water level is reached.

ITEM 5:

Review the action directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g. vessel integrity).

b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

RESPONSE:

This Item is generally related to the operations of the plant including operational procedures, maintenance procedures, training instructions, etc. Since WNP-2 is not in operation yet, presently many of the plant procedures which would cover this Item have not been prepared or are in draft form. However these items will be considered in the preparation of procedures for WNP-2, and the training program.

The basis for implementing these procedural and training improvements, will be the program established in NEDO-24708 and related documents.

ITEM 6:

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

RESPONSE:

A review of the emergency core cooling systems (ECCS) indicated that the system valves' positions are suitably controlled by the following means:

1. Automatic actuation of power operated valves within the system is provided to isolate the boundary/bypass paths and to align the system for proper operation. Main control room valve position indication is provided for these valves. The handswitches in the control room for these valves are spring return to the auto position to allow the valve to operate automatically if required.

2. Manual actuation of power operated valves within the system is administratively and procedurally controlled to assure proper system lineup exists at all times except during short periods of testing. The handswitches in the control room are keylocked type with the key removable in the "safe" position. The keys are located in an administratively controlled locker.

3. Manual valves within the main flow path are provided with locking provisions to ensure correct valve positions. Manual valves which are not accessible during power operation (i.e., located in drywell) are also provided with main control room position indicating lights.

4. Manual valves on branch piping to the main flow path piping are provided with locking provisions if incorrect valve position could affect system safety function. Exceptions are the piping high point vents, low point drains, and test connections valves which are verified procedurally to be aligned properly for operation.

   Procedures to assure that valves remain positioned properly after maintenance, testing, etc. will be written prior to startup testing and operation. Examples of such controls are listed below:

   A. If valve positions are to be changed for surveillance purposes, the surveillance procedure will have steps requiring return to normal valve line-up prior to completion. Start and completion of surveillance procedures will be logged in the control room logbook.

ITEM 6 (continued)

B. If maintenance is performed on a safety related system that requires only valve position to be changed from that specified in the valve line-up, the following sequence will occur:

(1) The request for the valve position change will be approved by the Shift Supervisor before it is implemented.

a. The Shift Supervisor will assure that Technical Specifications are met before approving the change.

(2) A list of valves and/or boundaries of valves so changed will be kept in a file or logbook accessible to all Shift Supervisors.

a. The change in status of any safety related system from operable to inoperable or vice versa will be logged in a logbook that will be reviewed by each on-coming operator at the controls and Shift Supervisor.

(3) When work has been completed, the order to return all valves to their previous position will be approved by the Shift Supervisor.

(4) The Shift Supervisor will not consider the system operable until all valves identified within the boundaries of the maintenance activities have been returned to the position specified in the valve line-up and written evidence to this effect has been presented to him.

C. When possible, Operations will perform a functional test or Surveillance Operability Test following maintenance on any safety related system. When such tests are not possible, a complete valve and electrical lineup will be performed within the tagged boundary and a partial functional test will be performed where possible to provide assurance that systems are in fact functional after maintenance.

D. System line-up changes, other than those covered by step-by-step procedures will be logged and abnormal line-ups will be covered during shift turnover.

E. During periodic tours, Operators and Supervisory personnel will conduct spot checks of fluid system and electrical line-ups.

ITEM 7:

Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

a. Whether interlocks exist to prevent transfer when high radiation indication exists.

b. Whether such systems are isolated by the containment isolation signal.

c. The basis on which continued operability of the above features is assured.

RESPONSE:

It is WPPSS intent to integrate the system review identified by Item 7a, b and c with Item 2.1.6a of NUREG-0578. WPPSS is currently obtaining consultant services to perform the system review on Item 2.1.6a specifically on the following:

1. Identify systems which could transport radioactive fluids to outside containment.

2. Identify the fluids which these systems could contain after a transient or accident.

3. Determine if the fluid contained could have a high radioactive burden.

4. For those systems whose contained fluid could have a high radioactive burden, examine the system for potential paths of leakage to outside the containment boundary.

When all the above items are completed, WPPSS will establish a surveillance program for leak detection and review existing preventive maintenance programs for systems identified in Item 4.

ITEM 8:

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

RESPONSE:

As previously mentioned, we do not have complete operating procedures as yet, however.these recommendations will most certainly be included.in the maintenance and test procedures.

ITEM 9:

Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

RESPONSE:

This requirement will be included in WNP-2's procedures and an open continuous communications channel will be established with the NRC.

ITEM 10:

Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

RESPONSE:

WNP-2 is currently involved in a review of this matter and it will be handled through our response to NUREG-0578 item 2.1.5b, and the H. R. Denton concern regarding High Point Vents on the Reactor Coolant System.

ITEM 11:

Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

RESPONSE:

WNP-2 has not yet submitted technical specifications, however a draft is being prepared for submittal in 1980 keeping in mind the lessons learned from the Three Mile Island accident. Modifications to the draft will be made as necessary to implement the recommendations presented as a result of TMI.

V

ATTACHMENT 2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NUMBER 2
DOCKET NO. 50-397

Response to Bulletins and Orders Task Force Requests
for
Plant Unique Data

A. BYPASS CAPACITY

Plant Steam Bypass Capacity, % Rated 25

B. SYSTEMS AND COMPONENTS SHARED BETWEEN UNITS

Not applicable to WNP-2 since it is a single unit facility.

C. PLANT-SPECIFIC SYSTEM INFORMATION

	System	General		Water Sources		Instrumentation and Control		Frequency of System and Component Tests
		Safety Classification	Seismic Category	Safety Classification	Seismic Category	Safety Classif.	Seismic Category	
1.	RCIC	1/2	I	G, 2	II, I	2	I,II	
2.	Isolation Condenser	N/A						
3.	HPCS	1/2	I	G, 2	II, I	2	I,II	
4.	HPCI	N/A						
5.	LPCS	1/2	I	2	I	2	I,II	
6.	LPSI	1/2	I	2	I	2	I,II	See Sheet 2
7.	ADS	1/2	I	---	---	2	I,II	
8.	SRV	1/2	I	---	---	2	I,II	
9.	RHR (including shutdown cooling, steam condensing, suppression pool cooling, containment spray modes)	1/2	I	2	I	2	I,II	
10.	SSW	3	I		I	2	I	
11.	RBCCW	G	II	G	II	2	II	
12.	CRDS	1/2	II,I	G	II	2	II	
13.	CST	G	II	---	___	G	II	
14.	Main Feedwater	G, 1	II, I	G	II	2	II	

TESTING FREQUENCY

	Pumps (System)	Valves	I & C	
RCIC	In accordance with Section XI	In accordance with Section XI	Per Tech. Specs. calibration generally on quarterly basis	Channel functional check, generally monthly
HPCS	"	"	"	"
LPCS	"	"	"	"
LPCI	"	"	"	"
ADS	"	"	"	"
SRV	--	"	--	--
RHR	"	"	"	"
[illegible]W	"	"	"	"
RBCCW			→Non-Tech. Spec. System	
CRDS	Operated routinely	(no special testing)	" " " "	Non-Safety
CST	--	--	--	--
Main Feed	Operated routinely		" " " "	

D. PRIMARY CONTAINMENT ISOLATION SYSTEM DATA (See FSAR Table 6.2-16, attached)

ABBREVIATIONS/LEGEND

Valve Type

AO	air operated
MO	motor operated
PC	positive closing
EHO	electro-hydraulic operated
SO	solenoid operated
EF	excess flow

Location

I	inside containment
O	outside containment

Power to Open/Close

AC	AC electrical power
DC	DC electrical power
Process, pro	process flow
PP	process fluid overpressurization
spr	spring

Normal Position

O	open
C	close

Process Fluid

W	water
A	air
S	steam
H	hydraulic fluid

Termination Zone

TB	turbine building
RB	reactor building
Rad W	radwaste building
SB	service building

D. PRIMARY CONTAINMENT ISOLATION SYSTEM DATA (Cont'd.)

ISOLATION SIGNAL CODES

Code or Group	Parameter(s) Sensed for Isolation	Set Point (Unit)
L	Reactor vessel low water level (Trip 3) - (A scram occurs at this level also. This is the higher of the three low water level signals.)	Not defined yet for WNP-2
A	Reactor vessel low water level (Trip 2)	
C	High radiation - Main steam	
D	Line break - Main steamline (steamline high space temperature or high steam flow).	
F	High drywell pressure (core standby cooling systems are started).	
K	Line break in RCIC system line to turbine (high RCIC pipe space temperature, high steam flow, or low steam line pressure).	
M	Line break in RHR shutdown piping (hi suction flow)	
P	Low main steamline pressure at inlet turbine (RUN mode only).	
S	Low drywell pressure	
U	High reactor vessel pressure	
W	High temperature at outlet of cleanup system non-regenerative heat exchanger	
Y	Standby liquid control system actuated	
Z	Reactor building ventilation exhaust plenum high radiation	
RM	Remote manual switch located in main control room	
G	Low condenser vacuum	
H	Turbine Building high temperature	
J	Line break in cleanup system - high space temperature	

ISOLATION SIGNAL CODES (Cont'd.)

Code or Group	Parameter(s) Sensed for Isolation	Set Point (Unit)
T	High leakage flow	Not defined yet for WNP-2
X	"K" plus RHR/RCIC equipment area high temperature	
N	High drywell pressure and low reactor pressure	
R	RHR equipment area high temperature	
V	Reactor vessel low water level (Trip 1)	
E	Reactor water cleanup system high differential flow	

TABLE 6.2-16

PRIMARY CONTAINMENT ISOLATION

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
MS Line A	18A	3.2-2 3.2-25 6.2-31j	55	A	MS-V-22A	AO Globe	I	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	-	No	S	Valves	T.B.	No	1, 15
					MS-V-28A	AO Globe	O	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	4	No	S	Valves	T.B.	No	1, 15
					MS-V-67A	MO Gate	O	AC	AC	A,C, G,D, P,H	RM	O	C	C	AS-IS	1-1/2	Std	5	No	S	Valves	T.B.	No	15
					MSLC-V-3A	MO Gate	O	AC	AC	30	RM	C	C	O	AS-IS	1-1/2	Std	10	Yes	S	Valves	R.B.	No	
MS Line B	18B	3.2-2 3.2-25 6.2-31j	55	A	MS-V-22B	AO Globe	I	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	-	No	S	Valves	T.B.	No	1, 15
					MS-V-28B	AO Globe	O	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	4	No	S	Valves	T.B.	No	1, 15
					MS-V-67B	MO Gate	O	AC	AC	A,C, G,D, P,H	RM	O	C	C	AS-IS	1-1/2	Std	5	No	S	Valves	T.B.	No	15
					MSLC-V-3B	MO Gate	O	AC	AC	30	RM	C	C	O	AS-IS	1-1/2	Std	10	Yes	S	Valves	R.B.	No	
MS Line C	18C	3.2-2 3.2-25 6.2-31j	55	A	MS-V-22C	AO Globe	I	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	-	No	S	Valves	T.B.	No	1, 15
					MS-V-28C	AO Globe	O	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	4	No	S	Valves	T.B.	No	1, 15
					MS-V-67C	MO Gate	O	AC	AC	A,C, G,D, P,H	RM	O	C	C	AS-IS	1-1/2	Std	5	No	S	Valves	T.B.	No	15
					MSLC-V-3C	MO Gate	O	AC	AC	30	RM	C	C	O	AS-IS	1-1/2	Std	10	Yes	S	Valves	R.B.	No	

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
MS Line D	18D	3.2-2 3.2-25 6.2-31j	55	A	MS-V-22D	AO Globe	I	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	-	No	S	Valves	T.B.	No	1, 15
					MS-V-28D	AO Globe	O	Air	Air/ Spr	A,C, G,D, P,H	RM	O	O/C	C	C	26	3-10	4	No	S	Valves	T.B.	No	1, 15
					MS-V-67D	MO Gate	O	AC	AC	A,C, G,D, P,H	RM	O	C	C	AS-IS	1-1/2	Std	5	No	S	Valves	T.B.	No	15
					MSLC-V-3D	MO Gate	O	AC	AC	30	RM	C	C	O	AS-IS	1-1/2	Std	10	Yes	S	Valves	R.B.	No	
MS Line Drain	22	3.2-2 6.2-31f	55	A	MS-V-16	MO Gate	I	AC	AC	A,C, G,D, P,H	RM	O	C	C	AS-IS	3	Std	-	No	S	Valves	T.B.	.19	
					MS-V-19	MO Gate	O	DC	DC	A,C, G,D, P,H	RM	O	C	C	AS-IS	3	Std	6	No	S				

AMENDMENT NO. 5
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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RFW Line A	17A	3.2-2 6.2-31b	55	A	RFW-V-10A	Check	I	Process	Process	-	-	0	O/C	O/C	-	24	-	-	No	W	Valves	T.B.	1.5	16
					RFW-V-32A	PC Check	0	Process	Pro/Spr	-	-	0	O/C	O/C	-	24	-	2	No	W				
					RFW-V-65A	MO Gate	0	AC	AC	31	Manual	0	O/C	O/C	AS-IS	24	Std	8	No	W				
					RWCU-V-40	MO Gate	0	AC	AC	47	Manual	0	0	C	AS-IS	6	Std	24	No	W				
RFW Line B	17B	3.2-2 6.2-31b	55	A	RFW-V-10B	Check	I	Process	Process	-	-	0	O/C	O/C	-	24	-	-	No	W	Valves	T.B.	1.5	16
					RFW-V-32B	PC Check	0	Precess	Pro/ Spr	-	-	0	O/C	O/C		24		2	No	W				
					RFW-V-65B	MO Gate	0	AC	AC	31	Manual	0	O/C	O/C	AS-IS	24	Std	8	No	W				
					RWCU-V-40	MO Gate	0	AC	AC	47	Manual	0	0	C	AS-IS	6	Std	24	No	W				
RRC Hydraulic Lines		3.2-3	57	B																				
Cylinder	76f				HY-V-17A	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	3/4	<5	5	No	H	Valves	R.B.	No	28
Cylinder	76b				HY-V-18A	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	3/4	<5	5						
Shuttle	76e				HY-V-19A	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	1/2	<5	5						
Drain	76c				HY-V-20A	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	1/2	<5	5						
Cylinder	77f				HY-V-17B	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	3/4	<5	5	No	H	Valves	R.B.	No	28
Cylinder	77b				HY-V-18B	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	3/4	<5	5						
Shuttle	77e				HY-V-19B	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	1/2	<5	5						
Drain	77c				HY-V-20B	SO Globe	0	AC	Spring	A,F	RM	0	0	C	C	1/2	<5	5						

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
HPCS to Reactor	6	3.2-7 6.2-31L	55	A	HPCS-V-5	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					HPCS-V-4	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	17	9						
LPCS to Reactor	8	3.2-7 6.2-31L	55	A	LPCS-V-6	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					LPCS-V-5	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	27	22						
HPCS pump suction from suppression pool	31	3.2-7 6.2-31n	56	B	HPCS-V-15	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	18	18	3	Yes	W	Valves	R.B.	No	18 24
LPCS pump suction	34	3.2-7 6.2-31n	56	B	LPCS-V-1	MO Gate	O	AC	AC	46	Manual	O	O	O/C	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18 24
HPCS test line	49	3.2-7 6.2-31f	56	B	HPCS-V-23	MO Globe	O	AC	AC	F,V	RM	C	C	C	AS-IS	12	Std	6	Yes	W	Valves	R.B.	No	18
HPCS pump min. flow					HPCS-V-12	MO Gate	O	AC	AC	38	RM	C	C	C	AS-IS	4	4	53						
HPCS suction relief					HPCS-RV-14	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	65						19
HPCS discharge relief					HPCS-RV-35	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	70						19
LPCS test line	63	3.2-7 6.2-31f	56	B	LPCS-V-12	MO Globe	O	AC	AC	F,V	RM	C	C	C	AS-IS	12	Std	4	Yes	W	Valves	R.B.	No	18
LPCS pump min. flow					LPCS-V-11	MO Globe	O	AC	AC	38	RM	C	C	O/C	AS-IS	3	Std	87						
LPCS suction relief					LPCS-RV-31	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	25						19
LPCS discharge relief					LPCS-RV-18	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	50						19
SLC to Reactor	13	3.2-5 6.2-31m	55	A	SLC-V-7	Check	I	Process	Process	-	-	C	C	C	-	1-1/2	-	-	No	W	Valves	R.B.	No	
					SLC-V-6	Check	O	Process	Process	-	-	C	C	C	-	1-1/2	-	6						
					SLC-V-4A	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21
					SLC-V-4B	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
DW Service Line	92	9.2-4 6.2-31L	56	B	DW-V-157	Gate	I	Manual	Manual	-	-	LC	LC	LC	-	2	-		No	W	Valves	S.B.	.13	
					DW-V-156	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	5						
RHR Condensing Mode Steam Supply	21	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	Yes	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-64	MO Gate	O	DC	DC	X	RM	C	C	C	AS-IS	10	16	2						
RCIC Turbine Steam Supply	45	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	No	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-8	MO Gate	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4	Std	2						
RCIC Pump Minimum Flow	65	3.2-8 6.2-31h	56	B	RCIC-V-19	MO Globe	O	DC	DC	33	RM	C	C	C	AS-IS	2	5	7	No	W	Valves	R.B.	No	22
RCIC Turbine Exhaust	4	3.2-8 6.2-31n	56	B	RCIC-V-68	MO Gate	O	DC	DC	35	Manual	O	O	O/C	AS-IS	10	Std	10	No	S	Valves	R.B.	No	22
RCIC Turbine Exhaust Vacuum Breaker	116	3.2-8 6.2-31i	56	B	RCIC-V-110	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	9	No	A	Valves	R.B.	No	17
					RCIC-V-113	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	5						
RCIC Vacuum Pump Discharge	64	3.2-8 6.2-31q	56	B	RCIC-V-69	MO Gate	O	DC	DC	36	Manual	O	O	O/C	AS-IS	1-1/2	Std	4	No	W	Valves	R.B.	No	22
RCIC Pump Suction from Suppression Pool	33	3.2-8 6.2-31n	56	B	RCIC-V-31	MO Gate	O	DC	DC	32	Manual	C	C	O/C	AS-IS	8	Std	2	No	W	Valves	R.B.	No	23
RPV Head Spray	2	3.2-8 6.2-31e	55	A	RCIC-V-66	Check	I	Process	Process	-	-	C	O	O/C	-	6	-	-	No	W	Valves	R.B.	No	3
					RCIC-V-13	MO Gate	O	DC	DC	34	RM	C	O/C	O/C	AS-IS	6	15	2	No	W	Valves	R.B.	No	
					RHR-V-23	MO Globe	O	DC	DC	L,U, M,R	RM	C	O/C	C	AS-IS	6	Std	7	Yes	W	Valves	R.B.	No	

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time (7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Drywell Spray Loop A	11A	3.2-6 6.2-31g	56	B	RHR-V-16A	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	16	10	26	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17A	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	16	10	24						
Drywell Spray Loop B	11B	3.2-6 6.2-31g	56	B	RHR-V-16B	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	16	10	12	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17B	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	16	10	2						
LPCI Loop A	12A	3.2-6 6.2-31L	55	A	RHR-V-41A	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42A	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	14	12	21						
LPCI Loop B	12B	3.2-6 6.2-31L	55	A	RHR-V-41B	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42B	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	14	12	20						
LPCI Loop C	12C	3.2-6 6.2-31L	55	A	RHR-V-41C	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42C	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	14	12	20						
Shutdown Cooling Return Loop A	19A	3.2-6 6.2-31m	55	A	RHR-V-50A	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123A	MO Gate	I	AC	AC	F,L, U,M, R	RM	C	O/C	C	AS-IS	1		-						
					RHR-V-53A	MO Globe	O	AC	AC	M,L, U,R	RM	C	O	C	AS-IS	12	40	5						
Shutdown Cooling Return Loop B	19B	3.2-6 6.2-31m	55	A	RHR-V-50B	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123B	MO Gate	I	AC	AC	F,L, U,M, R	RM	C	O/C	C	AS-IS	1		-						
					RHR-V-53B	MO Globe	O	AC	AC	M,L, U,R	RM	C	O	C	AS-IS	12	40	2						
Shutdown Cooling Suction	20	3.2-6 6.2-31k	55	A	RHR-V-9	MO Gate	I	AC	AC	L,U, M,R	RM	C	O	C	AS-IS	20	40	-	Yes	W	Valves	R.B.	No	
					RHR-V-8	MO Gate	O	AC	AC	L,U, M,R	RM	C	O	C	AS-IS	20	40	14						

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RHR Loop A: pump test line	47	3.2-6 6.2-31p	56	B	RHR-V-24A	MO Globe	O	AC	AC	F,V	RM	C	C	O/C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25A	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	33	Yes	W	Valves	R.B.	No	18 19
heat exch. steam relief					RHR-RV-55A	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	22	Yes	S	Valves	R.B.	No	18 19
heat exch. condensate					RHR-V-11A	MO Gate	O	AC	AC	F,V	RM	C	O/C	C	AS-IS	4	-	18	Yes	W	Valves	R.B.	No	18
heat exch. condensate relief					RHR-RV-36	Relief	O	PP	Spring	-	-	C	C	C	-	8	-	20	Yes	W	Valves	R.B.	No	18 20
pump minimum flow					RHR-FCV-64A	MO Globe	O	AC	AC	38	RM	O	C	O/C	AS-IS	3	8	22	Yes	W	Valves	R.B.	No	18
heat exch. thermal relief					RHR-RV-1A	Relief	O	PP	Spring	-	-	C	C	C	-	1-1/2	-	188	Yes	W	Valves	R.B.	No	18 19
heat exch. vent					RHR-V-73A	MO Globe	O	AC	AC	39	Manual	C	O/C	C	AS-IS	2	Std	175	Yes	A	Valves	R.B.	No	18
FDR system inter-tie					RHR-V-121	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	3	-	6	No	W	Valves	R.B.	No	
CAC system Loop A drain					RHR-V-134A	MO Gate	O	AC	AC	37	Manual	C	C	O/C	AS-IS	2	Std	44	Yes	W	Valves	R.B.	No	18
pump A suction relief					RHR-RV-88A	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18
RHR Loop B pump test line	48	3.2-6 6.2-31p	56	B	RHR-V-24B	MO Globe	O	AC	AC	F,V	RM	C	C	O/C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25B	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18 19
heat exch. steam relief					RHR-RV-55B	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	20	Yes	S	Valves	R.B.	No	18 19
pump A&B suction relief					RHR-RV-5	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	20	Yes	W	Valves	R.B.	No	18 19

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
heat exch. condensate	RHR-V-11B	MO Gate	O	AC	AC	F,V	RM	C	O/C	C	AS-IS	4	Std	15	Yes	W	Valves	R.B.	No	18
pump minimum flow	RHR-FCV-64B	MO Globe	O	AC	AC	38	RM	O	C	O/C	AS-IS	3	8	22	Yes	W	Valves	R.B.	No	18
flush line relief	RHR-RV-30	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	34	Yes	W	Valves	R.B.	No	18 19
heat exch. thermal relief	RHR-RV-1B	Relief	O	PP	Spring	-	-	C	C	C	-	1-1/2	-	189	Yes	W	Valves	R.B.	No	18 19
heat exch. vent	RHR-V-73B	MO Globe	O	AC	AC	39	Manual	C	O/C	C	AS-IS	2	Std	190	Yes	A	Valves	R.B.	No	18
CAC system Loop B drain	RHR-V-134B	MO Gate	O	AC	AC	37	Manual	C	C	O/C	AS-IS	2	Std	44	Yes	W	Valves	R.B.	No	18
pump B suction relief	RHR-RV-88B	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RHR Loop A Suppression Pool Suction	35	3.2-6 6.2-31n	56	B	RHR-V-4A	MO Gate	O	AC	AC	46	Manual	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop B Suppression Pool Suction	32	3.2-6 6.2-31n	56	B	RHR-V-4B	MO Gate	O	AC	AC	46	Manual	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop C Suppression Pool Suction	36	3.2-6 6.2-31n	56	B	RHR-V-4C	MO Gate	O	AC	AC	46	Manual	O	O	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop A:	117	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RV-95A	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	24	Yes	S	Valves	R.B.	No	18 19
condensate pot drain					RHR-V-124A	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	11	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-124B	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	12	Yes	W	Valves	R.B.	No	18
RHR Loop B:	118	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RV-95B	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	21	Yes	S	Valves	R.B.	No	18 19
condensate pot drain					RHR-V-125A	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	17	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-125B	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	14	Yes	W	Valves	R.B.	No	18
RHR Loop C:	26	3.2-6 6.2-31f	56	B																				
pump test line					RHR-V-21	MO Globe	O	AC	AC	F,V	RM	C	C	C	AS-IS	18	Std	34	Yes	W	Valves	R.B.	No	18
discharge header relief					RHR-RV-25C	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18 19
pump C suction relief					RHR-RV-88C	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	37	Yes	W	Valves	R.B.	No	18 19
pump minimum flow					RHR-V-64C	MO Globe	O	AC	AC	38	RM	O	C	O/C	AS-IS	3	8	30	Yes	W	Valves	R.B.	No	18
Suppression Pool Spray Loop A	25A	3.2-6 6.2-31h	56	B	RHR-V-27A	MO Gate	O	AC	AC	F,V	RM	C	C	O/C	AS-IS	6	10	5	Yes	W	Valves	R.B.	No	2, 18, 24
Suppression Pool Spray Loop B	25B	3.2-6 6.2-31h	56	B	RHR-V-27B	MO Gate	O	AC	AC	F,V	RM	C	C	O/C	AS-IS	6	10	6	Yes	W	Valves	R.B.	No	2, 18, 24

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
CAC Division 1 discharge to drywell	96	3.2-17 6.2-31g	56	B	CAC-V-2	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	4	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2A	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	6						
CAC Division 2 suction from drywell	97	3.2-17 6.2-31g	56	B	CAC-V-15	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1B	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	4						
CAC Division 2 discharge to drywell	98	3.2-17 6.2-31g	56	B	CAC-V-11	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	8	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2B	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	10						
CAC Division 1 suction from drywell	99	3.2-17 6.2-31g	56	B	CAC-V-6	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	4	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1A	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 discharge to wetwell	102	3.2-17 6.2-31g	56	B	CAC-V-4	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	3	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4A	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	5						
CAC Division 2 discharge to wetwell	103	3.2-17 6.2-31g	56	B	CAC-V-13	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	7	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4B	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	9						
CAC Division 2 suction from wetwell	104	3.2-17 6.2-31g	56	B	CAC-V-17	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	5	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3B	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 suction from wetwell	105	3.2-17 6.2-31g	56	B	CAC-V-8	MO Gate	O	DC	DC	37	Manual	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3A	EHO Globe	O	AC	AC	37	Manual	C	C	O/C	C	2-1/2	Std	6						

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RB to Wetwell Vacuum Breakers	119	3.2-15 6.2-31q	56	B	CSP-V-9	AO Butfy	O	Spring	Air	40	RM	C	C	C	O	24	4	1	Yes	A	Valves	R.B.	No	17
					CSP-V-10	PC Check	O	Process	Process	-	RM	C	C	C	-	24	-	4	Yes	A	Valves	R.B.	No	26
RB to Wetwell Vacuum Breakers & Wetwell Ventilation Supply	66	3.2-15 6.2-31b 6.2-31q	56	B	CSP-V-5	AO Butfy	O	Spring	Air	40	RM	C	C	C	O	24	4	7	Yes	A	Valves	R.B.	No	17
					CSP-V-7	PC Check	O	Process	Process	-	RM	C	C	C	-	24	-	10	Yes	A	Valves	R.B.	No	26
					CSP-V-4	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	24	4	14	No	A	Valves	R.B.	No	
					CSP-V-3	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	24	4	17	No	A	Valves	R.B.	No	
RB to Wetwell Vacuum Breakers & Wetwell Ventilation Exhaust	67	3.2-15 6.2-31j 6.2-31q	56	B	CSP-V-6	AO Butfy	O	Spring	Air	40	RM	C	C	C	O	24	4	9	Yes	A	Valves	R.B.	No	17
					CSP-V-8	PC Check	O	Process	Process	-	RM	C	C	C	-	24	-	16	Yes	A	Valves	R.B.	No	26
					CEP-V-4A	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	24	4	10	No	A	Valves	R.B.	No	
					CEP-V-3A	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	24	4	12	No	A	Valves	R.B.	No	
					CEP-V-4B	AO Gate	O	Air	Spring	F,A,Z	RM	C	C	C	C	2	1	10	No	A	Valves	R.B.	No	
					CEP-V-3B	AO Gate	O	Air	Spring	F,A,Z	RM	C	C	C	C	2	1	12	No	A	Valves	R.B.	No	
Drywell Ventilation Supply	53	3.2-15 6.2-31b	56	B	CSP-V-2	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	30	4	1	No	A	Valves	R.B.	No	17
					CSP-V-1	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	30	4	4	No	A	Valves	R.B.	No	
Drywell Ventilation Exhaust	3	3.2-15 6.2-31j	56	B	CEP-V-1A	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	30	4	12	No	A	Valves	R.B.	No	17
					CEP-V-2A	AO Butfy	O	Air	Spring	F,A,Z	RM	C	C	C	C	30	4	8	No	A	Valves	R.B.	No	
					CEP-V-1B	AO Gate	O	Air	Spring	F,A,Z	RM	C	C	C	C	2	1	12	No	A	Valves	R.B.	No	
					CEP-V-2B	AO Gate	O	Air	Spring	F,A,Z	RM	C	C	C	C	2	1	8	No	A	Valves	R.B.	No	

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time (7) (11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RCC Inlet Header	5	3.2-14 6.2-31t	56	B	RCC-V-104	MO Gate	O	AC	AC	F,A	–	O	O		AS-IS	10	Std	5	No	W	Valves	R.B.	No	17
					RCC-V-5	MO Gate	O	AC	AC	F,A	–	O	O		AS-IS	10	Std	3						
RCC Outlet Header	46	3.2-14 6.2-31o	56	B	RCC-V-21	MO Gate	O	AC	AC	F,A	–	O	O		AS-IS	10	Std	3	No	W	Valves	R.B.	No	
					RCC-V-40	MO Gate	I	AC	AC	F,A	–	O	O		AS-IS	10	Std	–						
Suppression Pool Cleanup Suction	100	3.2-12 6.2-31i	56	B	FPC-V-153	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	2	No	W	Valves	R.B.	No	17
					FPC-V-154	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	7						
Suppression Pool Cleanup Return	101	3.2-12 6.2-31o	56	B	FPC-V-156	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	3	No	W	Valves	R.B.	No	17
					FPC-V-149	Globe	O	Manual	Manual	–	–	LC	LC	LC	–	6	–	41						
RWCU From Reactor	14	3.2-11 6.2-31k	55	A	RWCU-V-1	MO Gate	I	AC	AC	A,J, E,W	RM	O	O	C	AS-IS	6	Std	–	No	W	Valves	Rad. W.	.35	
					RWCU-V-4	MO Gate	O	DC	DC	A,J, E,Y,W	RM	O	O	C	AS-IS	6	Std	4						
RRC Pump A seal Water	43A	3.2-3 6.2-31c	56	B	RRC-V-13A	Check	I	Process	Process	–	–	O	O	C	–	3/4	Std	–	No	W	Valves	R.B.	No	
					RRC-V-16A	MO Gate	O	AC	AC	45	Manual	O	O	C	AS-IS	3/4	Std	2						
RRC Pump B seal water	43B	3.2-3 6.2-31c	56	B	RRC-V-13B	Check	I	Process	Process	–	–	O	O	C	–	3/4	Std	–	No	W	Valves	R.B.	No	
					RRC-V-16B	MO Gate	O	AC	AC	45	Manual	O	O	C	AS-IS	3/4	Std	2						
RRC Sample Line	77Aa	3.2-3 6.2-31d	55	A	RRC-V-19	SO Globe	I	AC	Spring	A,C, D,P	RM	O	O	C	C	3/4	<5	–	No	W	Valves	T.B.	.05	
					RRC-V-20	AO Globe	O	Air	Spring	A,C, D,P	RM	O	O	C	C	3/4	Std							

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Drywell Equipment Drain	23	3.2-9 6.2-31k	56	B	EDR-V-19	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					EDR-V-20	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	4						
Drywell Floor Drain	24	3.2-10 6.2-31k	56	B	FDR-V-3	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					FDR-V-4	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	3						
Decontamination Soltn. Supply Header	94	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
Decontamination Soltn. Return Header	95	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
CIA for Safety Relief Valve Accumulators	56	3.2-21 6.2-31c	56	B	CIA-V-21 CIA-V-20	Check MO Globe	O O	Process AC	Process AC	- 41	- Manual	C O	C O	C O	- AS-IS	3/4 3/4	- Std	3 10	No	A	Valves	R.B.	No	17
CIA Line A for ADS Accumulators	89B	3.2-21 6.2-31c	56	B	CIA-V-31A CIA-V-30A	Check MO Globe	O O	Process AC	Process AC	- 42	- Manual	C O	C O	C O	- AS-IS	1/2 1/2	- Std	5 15	No	A	Valves	R.B.	No	17
CIA Line B for ADS Accumulators	91	3.2-21 6.2-31c	56	B	CIA-V-31B CIA-V-30B	Check MO Globe	O O	Process AC	Process AC	- 42	- Manual	C O	C O	C O	- AS-IS	1/2 1/2	- Std	2 15	No	A	Valves	R.B.	No	17
CRD Insert Lines (185 separate lines)	9	3.2-4	56	B						See Note 4														
CRD Withdrawal lines (185 separate lines)	10	3.2-4	56	B						See Note 4														

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TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Air line for testing RHR-V-50A	42d	6.2-31r 3.2-6	56	B	PI-VX-42d	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-216	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-50B	69c	6.2-31r 3.2-6	56	B	PI-VX-69c	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-221	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41A	61f	6.2-31r 3.2-6	56	B	PI-VX-61f	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-219	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41B	54Bf	6.2-31r 3.2-6	56	B	PI-VX-54Bf	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-218	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41C	62f	6.2-31r 3.2-6	56	B	PI-VX-62f	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-220	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing LPCS-V-6	78d	6.2-31r 3.2-7	56	B	PI-VX-78d	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-222	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing HPCS-V-5	78e	6.2-31r 3.2-7	56	B	PI-VX-78e	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	valves	R.B.	No	25
					PI-VX-223	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RCIC-V-66	54Aa	6.2-31r 3.2-8	56	B	PI-VX-54Aa	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-217	Globe	0	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing WW-DW vacuum relief valves	82e	6.2-31r 9.3-1	56	B	CAS-V-453	SO Globe	0	AC	Spring	44	-	C	C	C	C	1	<5	5	No	A	Valves	R.B.	No	25
					CAS-CVX-82e	Check	0	Process	Process	-	-	C	C	C	-	1	-	7						
Air line for maintenance	93	9.3-1 6.2-31t	56	B	-	Pipe Cap	I	-	-	-	-	C	C	C	-	2	-	-	No	A	Cap & Valve	S.B.	No	
					SA-V-109	Gate	0	Manual	Manual	-	-	LC	LC	LC	-	2	-	1						
Tip lines	27a-c		54	-	C51J004	SO Ball	0	AC	AC	L,F	RM.	C	C	C	C	3/8	<5	2	No	A	Valves	R.B.	No	29
					C51J004	Shear	0	-	Explosive	43	-	0	0	0	0	3/8	-	2						

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Radiation Monitor (S-SR-20) Supply line	85b	6.2-31s 3.2-15	56	B	PI-VX-250	SO Gate	O	AC	Sring	F,A	RM	O	O	C	C	1	<5	-	No	A	Valves	R.B.	No	25
					PI-VX-251	SO Gate	O	AC	Spring	F,A	RM	O	O	C	C	1	<5	-						
Radiation Monitor (S-SR-20) Return line	72f	6.2-31s 3.2-15	56	B	PI-VX-253	SO Gate	O	AC	Spring	F,A	RM	O	O	C	C	1	<5	-	No	A	Valves	R.B.	No	25
					PI-CVX 72f	Check	O	Process	Process	-	-	O	O	C	-	1	-	-						
Radiation Monitor (S-SR-21) Supply line	73e	6.2-31s 3.2-15	56	B	PI-VX-256	SO Gate	O	AC	Spring	F,A	RM	O	O	C	C	1	<5	-	No	A	Valves	R.B.	No	25
					PI-VX-257	SO Gate	O	AC	Spring	F,A	RM	O	O	C	C	1	<5	-						
Radiation Monitor (S-SR-21) Return line	29c	6.2-31s 3.2-15	56	B	PI-VX-259	SO Gate	O	AC	Spring	F,A	RM	O	O	C	C	1	<5	-	No	A	Valves	R.B.	No	25
					PI-CVX-29c	Check	O	Process	Process	-	-	O	O	C	-	1	-	-						
All Instrument lines from reactor	-		55	A	-	EF Check	O	Spring	EF	-	-	O	O	O	-	3/4 & 1	-	-		-	Valves	R.B.	No	27
						Globe	O	Manual	Manual	-	-	O	O	O	-	3/4 & 1	-	-						
All Instrument lines from primary containment	-		56	B	-	EF Check	O	Spring	EF	-	-	O	O	O	-	1 & 1-1/2	-	-		-	Valves	R.B.	No	27
						Globe	O	Manual	Manual	-	-	O	O	O	-	1 & 1-1/2	-	-						
Instrument lines (Hydrogen monitors) return to containment	-	3.2-15	56	B	-	Check	O	Process	Process	-	-	C	C	O	-	1	-	-	Yes	A, S	Valves	R.B.	No	27
						Globe	O	Manual	Manual	-	-	O	O	O	-	1	-	-						

TABLE 6.2-16 (Continued)

PRIMARY CONTAINMENT AND REACTOR VESSEL
ISOLATION SIGNAL CODES FOR TABLE 6.2-16**

Signal	Description
L*	Reactor vessel low water level (Trip 3) - (A scram occurs at this level also. This is the higher of the three low water level signals).
A*	Reactor vessel low water level (Trip 2)
C*	High radiation - Main steam
D*	Line break - Main steamline (steamline high space temperature or high steam flow).
F*	High drywell pressure (core standby cooling systems are started).
J*	Line break in cleanup system - high space temperature.
K*	Line break in RCIC system line to turbine (high RCIC pipe space temperature, high steam flow, or low steam line pressure).
M*	Line break in RHR shutdown piping (hi suction flow)
P*	Low main steamline pressure at inlet turbine (RUN mode only).

* These are the isolation functions of the primary containment and reactor vessel isolation system; other functions are given for information only.

** See notes 30 through 46 for isolation signals generated by the individual system process control signals or for remote-manual closure based on information available to the operators. These notes are referenced in the "isolation signal" column.

TABLE 6.2-16 (Continued)

Signal	Description
S	Low drywell pressure
U	High reactor vessel pressure
W	High temperature at outlet of cleanup system non-regenerative heat exchanger
Y	Standby liquid control system actuated
Z*	Reactor building ventilation exhaust plenum high radiation
RM	Remote manual switch located in main control room
G*	Low condenser vacuum
H*	Turbine Building high temperature
T*	High leakage flow
X*	"K" plus RHR/RCIC equipment area high temperature
N*	High drywell pressure and low reactor pressure
R*	RHR equipment area high temperature
V*	Reactor vessel low water level (Trip 1)
E*	Reactor water cleanup system high differential flow

* These are the isolation functions of the primary containment and reactor vessel isolation system; other functions are given for information only.

TABLE 6.2-16 (Continued)

ABBREVIATIONS/LEGEND

Valve Type

AO	air operated
MO	motor operated
PC	positive closing
EHO	electro-hydraulic operated
SO	solenoid operated

Location

I	inside containment
O	outside containment

Power to Open/Close

AC	AC electrical power
DC	DC electrical power
Process, pro	process flow
PP	process fluid overpressurization
spr	spring

Normal Position

O	open
C	close

Process fluid

W	water
A	air
S	steam
H	hydraulic fluid

Termination Zone

TB	turbine building
RB	reactor building
Rad W	radwaste building
SB	service building

TABLE 6.2-16 (Continued)

NOTES FOR TABLE

Type C testing is discussed in Figure 6.2-31 which shows the isolation valve arrangement. Unless otherwise noted (see notes 4, 27, 28,29) all valves listed in Table 6.2-16 are type C tested.

1. Main steam isolation valves require that both solenoid pilots be de-energized to close valves. Accumulator air pressure plus spring set act together to close valves when both pilots are de-energized. Voltage failure at only one pilot does not cause valve closure. The valves are designed to fully close in less than 10 seconds.

2. Suppression cooling valves have interlocks that allow them to be manually reopened after automatic closure. This setup permits suppression pool spray, for high drywell pressure conditions, and/or suppression water cooling. When automatic signals are not present, these valves may be opened for test or operating convenience.

3. Testable check valves are designed for remote opening with zero differential pressure across the valve seat. The valves will close on reverse flow even though the test switches may be positioned for open. The valves open when pump pressure exceeds reactor pressure even though the test switch may be positioned for close.

4. The isolation provisions for the CRD lines are commensurate with the essential requirement that the control rods are inserted on a scram. Isolation of the hydraulic lines is provided by check valves 115 and 138 and solenoid valves 120, 122, and 123 on the hydraulic control units (HCU) and by air operated valves F010, F011 on the scram discharge header (see Figures 4.6-5a and b). The HCU's and scram discharge headers as well as the hydraulic lines themselves are Seismic I, and are qualified to the appropriate accident environment. The failure and scram position of all power operated valves are compatible with system isolation and, at the same time, rod insertion on a scram. Addition of power operated isolation valves on the hydraulic lines themselves could prevent control rod insertion. Manual isolation valves 101 and 102 allow for further isolation if it becomes necessary. The hydraulic lines are small and terminate in the reactor building which is served by the standby gas treatment system. The hydraulic lines and their manual isolation valves in the scram discharge header and its air operated valves are code group B.

TABLE 6.2-16 (Continued)

The hydraulic control unit (HCU) is a General Electric factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive by the application of precisely timed sequences of pressures and flows to accomplish slow insertion or withdrawal of the control rods for power control, and rapid insertion for reactor scram.

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide.

Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, 1) all welds are penetrant tested (PT), 2) all socket welds are inspected for gaps between pipe and socket bottom, 3) all welding is performed by qualified welders, and 4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Group A, B, or C. This is supplemented by the QC techniques.

The CRD lines will be included in the type A test leakage since the reactor pressure vessel and the nonseismic portions of the CRD system are vented during the performance of the type A test. The CRD insert and withdraw lines are compatible with the criteria intended by 10CFR50, Appendix J, for type C testing, since the acceptance criterion for type C testing allows demonstration of fluid leakage rates by associated bases.

TABLE 6.2-16 (Continued)

5. Alternating current motor-operated valves required for isolation functions are powered from the AC standby power buses. Direct current operated isolation valves are powered from station batteries.

6. All motor-operated isolation valves remain in the last position upon failure of valve power. All air-operated valves close on motive air failure or in the safest position.

7. The standard minimum closing rate is 12 inches of nominal valve diameter per minute for gate valves and 4 inches of valve stem travel per minute for globe valves. For example, a 12 inch gate valve will close in one minute.

8. Reactor building ventilation exhaust plenum high radiation signal (Z) is generated by two trip units; this requires one unit at high trip or both units at down scale (instrument failure) trip, in order to initiate isolation.

9. Primary Containment and Reactor Vessel Isolation Signals (PCRVIS) are indicated by letters. Isolation signals generated by the individual system process control signals or for remote manual closure based on information available to the operator are discussed in the referenced notes in the "isolation signal" column.

10. Normal status position of valve (open or closed) is the position during normal power operation of the reactor (see Normal Position column).

11. The specified closure rates are as required for containment isolation or system operation, whichever is less. Reported times are in seconds.

12. All isolation valves are Seismic I.

13. Used to evaluate primary containment leakage which may bypass the secondary containment emergency filtration system. See 6.2.3.2.

14. Reported sizes are the valve nominal diameters in inches. Size indicated is containment side of relief valve when relief valve size is not equal on both sides.

TABLE 6.2-16 (Continued)

15. Leakage control system provided, see 6.7.

16. Bypass leakage of secondary containment is not considered during design basis LOCA, see 6.2.3.2.

17. Valve operability will be improved because the environmental conditions are better outside the primary containment from the standpoint of humidity, radiation, pressure and temperature transients, and post-LOCA pipe whip and jet impingement.

18. These lines connect to systems outside of the containment which meet the requirements for a closed system as set by NRC SRP 6.2.4, Section II, paragraph 3e. These systems are considered an extension of the primary containment. Any leakage out of these systems will be processed by the standby gas treatment system.

19. Relief valve setpoint greater than 77.5 psig (1.5 times containment design pressure).

20. Relief valve setpoint is 75 psig.

21. Cannot be reshut after opening without disassembly.

22. See 6.2.4.3.2.2.1.2

23. See 6.2.4.3.2.2.2

24. Due to redundancy within the emergency core cooling systems, some subsystems may be secured during the long term cooling period. In addition RHR loops A and B have several discharge paths (LPCI, Drywell Spray, Suppression Chamber Spray, Suppression Pool Cooling) which the operator may select during the 30 day post-LOCA period.

25. Applicable portion of the flow diagrams 3.2-6, -7, -8, -15 and 9.3-1 to be updated to reflect the configurations shown on Figures 6.2-31r and -31s.

26. The disc on the check valve is maintained in the close position during normal operation by means of a spring actuated lever arm and magnets embedded in the periphery of the disc. The magnetic and spring forces maintain the disc shut until the differential force to open the valve exceeds 0.2 psid. The check valves have position

TABLE 6.2-16 (Continued)

indication lights which can alert the operators to the fact that the check valve is not fully closed. The operator can then remotely shut the valve by means of a pneumatic operator. The operating switch is spring-return to neutral so the vacuum breaker function will not be impaired. The air supply to these valves is Quality Class I.

27. Instrument lines that penetrate primary containment conform to Regulatory Guide 1.11. The lines that connect to the reactor pressure boundary include a restricting orifice inside containment, are Seismic Category I and terminate in instruments that are Seismic Category I. The instrument lines also include manual isolation valves and excess flow check valves or equivalent (see hydrogen monitor return lines). These penetrations will not be type C tested since the integrity of the lines are continuously demonstrated during plant operations where subject to reactor operating pressure. In addition, all lines are subject to the type A test pressure on a regular interval. Leaktight integrity is also verified with completion of functional and calibration surveillance activities as well as by visual inspection during daily operator patrols as applicable.

28. Penetrations X-76 and X-77 contain lines for the hydraulic control of the reactor recirculation flow control valve. These lines contain corrosive hydraulic fluid used to position the reactor recirculation flow control valve.

These lines inside of the containment are Seismic Category I and Quality Group B. They are provided with failed closed automatic isolation valves outside the containment which receive an automatic isolation signal on high drywell pressure.

These lines meet the requirement of General Design Criterion 57 and therefore require only single automatic isolation valves outside of the containment. These lines also meet the requirement of Standard Review Plan 6.2.4. They are designed to Seismic Category I, Code Group B and the following criteria:

TABLE 6.2-16 (Continued)

a. do not communicate with either the reactor coolant system or the containment atmosphere,

b. are protected against missiles and pipe whip,

c. are designed to withstand temperatures at least equal to the containment design temperature,

d. are designed to withstand the external pressure from the containment structural acceptance test, and

e. are designed to withstand the loss-of-coolant accident transient and environment.

Even if the failed closed valve were to not shut there will be no leakage of containment atmosphere through the hydraulic control lines since the piping inside the primary containment remains intact. There are no active component failures which would compromise the integrity of the closed system inside the primary containment. Integrity of the closed system inside the primary containment is, essentially, constantly monitored since the system is under a constant operating pressure of 1800 psig. Any leakage through this system would be noticed because operation would be erratic and because of indications provided on the hydraulic control unit. In addition, in order to perform type C tests on these lines, the system would have to be disabled and drained of the corrosive hydraulic fluid. This is considered to be detrimental to the proper operation of the system in that possible damage could occur in establishing the test condition or restoring the system to normal.

These lines and associated isolation valves should therefore be considered to be exempt from type C testing.

29. Since the traversing incore probe (TIP) system lines do not communicate freely with the containment atmosphere or the reactor coolant, General Design Criteria 55 and 56 are not directly applicable to this specific class of lines. The basis to which these lines are designed is more closely described by General Design Criterion 54, which states in effect that isolation capability of a system should be commensurate with the safety importance of that isolation. Furthermore, even though the failure of the TIP system lines presents no safety consideration, the TIP system has redundant isolation capabilities.

TABLE 6.2-16 (Continued)

The safety features have been reviewed by the NRC for BWR/4 (Duane Arnold), BWR/5 (Nine Mile Point) and BWR/6 (GESSAR), and it was concluded that the design of the containment isolation system meets the objectives and intent of the General Design Criteria.

Isolation is accomplished by a seismically qualified solenoid-operated ball valve, which is normally closed. To ensure isolation capability, an explosive shear valve is installed in each line. Upon receipt of a signal (manually initiated by the operator), this explosive valve will shear the TIP cable and seal the guide tube.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of five valves may be opened at any one time to conduct calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes a cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. If a TIP cable fails to withdraw or a ball valve fails to close, the explosive shear valve is actuated. The ball valve position is indicated in the control room.

The WNP-2 TIP system design specifications require that the maximum leakage rate of the ball and shear valves shall be in accordance with the Manufacturer's Standardization Society (Hydrostatic Testing of Valves). The ball valves are 100% leak tested to the following criteria by the manufacturer:

Pressure	0 - 62 psig
Temperature	$340^{o}F$
Leak Rate	$10^{-3} cm^3/sec$

A statistically chosen sample of the shear valves is tested by the manufacturer to the following criteria:

Pressure	0 - 125 psig
Temperature	$340^{o}F$
Leak Rate	$10^{-3} cm^3/sec$ STP

TABLE 6.2-16 (Continued)

The shear valves have explosive squibs and require testing to destruction. They cannot therefore be 100% tested.

As stated above, the penetration is automatically closed following use. During normal operation the penetration will be open approximately eight hours per month to obtain TIP information. If a failure occurred such as not being able to withdraw the TIP cable, the shear valve could be closed to isolate the penetrations. Installation requirements are that the guide tube/penetration flange/ball and shear valve composite assemble not leak at a rate greater than 10^{-4} std cc/sec at 80 psig. Further leak testing of the shear valves is not recommended since destructive testing would be required.

Leak testing of the ball valves also is not recommended since the guide tube terminates in a sealed indexer housing which is kept under a positive pressure by a nitrogen or air purge. The purge make-up will be indicative of the system leakage. Note that the TIP ball valve is normally closed and thus is a part of the leakage barrier being monitored. Consequently, the personnel exposure required to conduct type C tests from inside the containment is not warranted.

30. System is initiated after a LOCA. Isolation valves will auto close on the following high leakage conditions:

    a. 5 PSI between main steam isolation valves, 60 seconds after system initiation

    b. High flow from main steam line to low pressure manifold, 150 seconds after system initiation

    c. Inboard main steam isolation valve opened, after system initiation

31. PCRVIS is not desirable since the feedwater system, although not an ESF system, could be a significant source of make-up after a LOCA which is not concurrent with a seismic event.

    Feedwater check valves on either side of the containment provide immediate leak isolation, if required. The feedwater block valves can, however, be remote-manually closed if there is no indication of feedwater flow (see 6.2.4.3.2.1.1.1).

TABLE 6.2-16 (continued)

32. The RCIC system will be initiated by low water level (B signal) and subsequently will be automatically tripped by one of the turbine shutdown signals listed below. The RCIC system will be operating at most only during the first several hours after a LOCA. The operator upon receiving indication that the RCIC system is no longer operable will complete isolation of the system by remote-manually shutting the isolation valves which have not been automatically shut. Also the operator will isolate the RCIC system on a high level alarm in the appropriate reactor building sump.

Automatic shutdown of the RCIC turbine occurs upon receipt of any one of the following signals:

a. turbine overspeed

b. high water level in the RPV

c. low RCIC pump suction pressure

d. high turbine exhaust pressure

e. closure of steam supply valves

Automatic closure of steam supply valves to the turbine occurs upon receipt of any one of the following signals:

a. high flow in steam supply line

b. high area temperature

c. low reactor pressure of 50 psig

d. high pressure between turbine exhaust diaphragms

The low reactor pressure signal of 50 psig is expected to occur almost immediately after a design basis LOCA and within several hours for a small LOCA. Leakage from this system, e.g., packing gland, pump seals, is expected to be negligible because of the small leakage rates expected and because of the short operating time (see reply to Question 212.003 for an estimate of maximum leakage rates expected and the radiological consequences).

TABLE 6.2-16 (Continued)

33. The RCIC minimum flow valve is open only between the time of system initiation and the time at which the system flow to the RPV exceeds 40 GPM. The valve is shut at all other times. RCIC-V-19 auto closes when the turbine throttle valve is closed following a turbine trip (see note 32). Should a leak occur when the valve is open, it will be detected by a high level alarm in the appropriate reactor building sump.

34. The RCIC injection valve is open only during RCIC turbine operation. Injection line check valves on either side of the containment provide immediate leak isolation, if required. RCIC-V-13 auto closes when the turbine throttle valve is closed following a turbine trip (see note 32).

35. The RCIC steam exhaust valve, RCIC-V-68, is normally open at all times. Should a leak occur, it would be detected and alarmed by the RCIC room high temperature leak detection system. (see note 32)

36. The RCIC vacuum pump discharge valve, RCIC-V-69, is normally open at all times. The valve could be remote-manually closed by the operator upon indication that vacuum (annuniated in Main Control Room) can no longer be maintained in the barometric condenser.

37. System isolation valves are normally closed. System is placed in operation only if the hydrogen monitors detect hydrogen buildup after a LOCA.

    The operator has flow indication, in the main control room, of gas leaving and entering the containment. Should these flows vary significantly from one another, it would be detected in the main control room and the process loop in service could be shutdown.

38. The minimum flow valve for an ECCS pump is open only between time of ECCS initiation and the time at which the system flow to the RPV exceeds 640 gpm. The valve is shut at all other times. Should a leak occur when the valve is open, it will be detected by a high level alarm in the appropriate reactor building sump.

39. Valve is open only during shutdown. Valve position is provided in main control room to provide the operator confirmation of valve status.

TABLE 6.2-16 (Continued)

40. Normally closed. Signalled to open if reactor building pressure exceeds wetwell pressure by 0.5 psid. Valves automatically reshut when the above condition no longer exists. Operator to use valve position indicator as confirmation of valve status.

41. Indication of containment air compressor discharge header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-20 should the containment air compressors become unavailable. The isolation check valve, CIA-V-21, provides immediate isolation.

42. Indication of nitrogen bottle header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-30(A,B) should the nitrogen bottle bank pressure decrease below the alarm setpoint. The isolation check valves, CIA-V-31(A,B) provide immediate isolation.

43. The operator's indication that remote-manual closure of the TIP shear valves is required, is failure of the TIP ball valves to close as monitored on Panel S.

44. Normally closed. Opened only when testing wetwell to drywell vacuum breakers.

45. The isolation valve can be remote-manually closed upon indication that the CRD or the RRC pumps have been tripped. The isolation check valves, RRC-V-13 (A, B), provide immediate isolation.

46. These valves are the ECCS suction and discharge isolation valves. ECCS operation is essential during the LOCA period; therefore, there are no automatic isolation signals. The valve closure requirement will be indicated by a high level alarm in the appropriate reactor building sump, which will be indicative of excessive ECCS leakage into secondary containment.

47. The isolation valve can be remote-manually closed upon indication that the RWCU pumps have been tripped. The reactor feedwater isolation check valves provide immediate isolation.

NOTES ON TYPE C TESTING (ISOLATION VALVE LEAKAGE TESTING):

1. TYPE C TESTING IS PERFORMED BY APPLYING A DIFFERENTIAL PRESSURE IN THE SAME DIRECTION AS SEEN BY THE VALVES DURING CONTAINMENT ISOLATION.

2. TYPE C TESTING IS PERFORMED BY PRESSURIZING BETWEEN THE TWO-PIECE DISK GATE VALVE.

3. TYPE C TESTING IS PERFORMED BY PRESSURIZING BETWEEN THE ISOLATION VALVES. THE TEST YIELDS CONSERVATIVE RESULTS SINCE THE INBOARD GLOBE VALVE IS PRESSURIZED UNDER THE SEAT DURING THE TEST; WHEREAS, DURING CONTAINMENT ISOLATION, IT IS PRESSURIZED ABOVE THE SEAT.

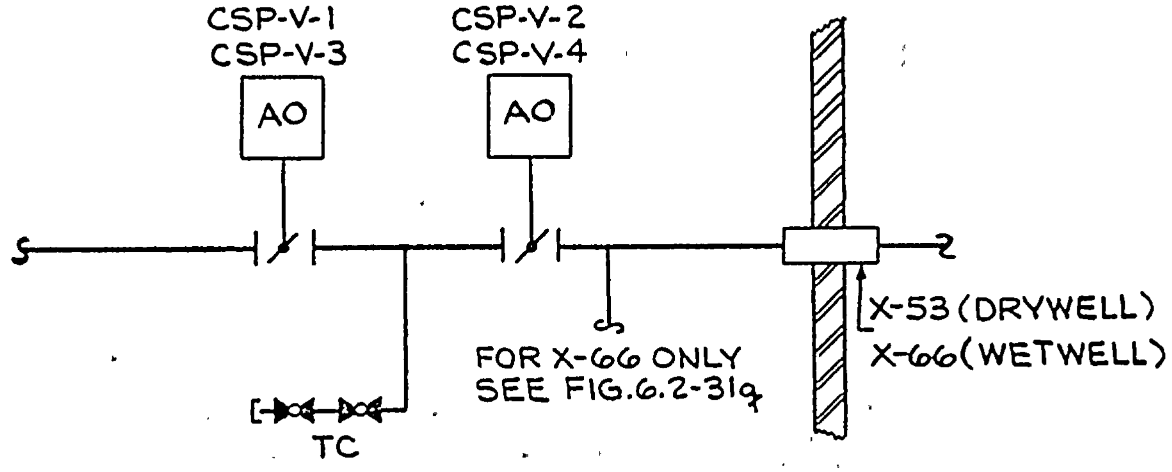
4. TYPE C TESTING IS PERFORMED BY PRESSURIZING BETWEEN THE ISOLATION VALVES. THE TEST YIELDS EQUIVALENT RESULTS FOR THE INBOARD GATE OR BUTTERFLY VALVE.*

5. TYPE C TESTING IS PERFORMED BY PRESSURIZING THE ISOLATION VALVE IN THE OPPOSITE DIRECTION AS WHEN THE VALVE PERFORMS CONTAINMENT ISOLATION. SINCE THE ISOLATION VALVE IS A GATE VALVE, THE TEST YIELDS EQUIVALENT RESULTS.*

6. TYPE C TESTING IS PERFORMED BY PRESSURIZING BETWEEN THE ISOLATION VALVES. THE TEST YIELDS EQUIVALENT RESULTS FOR THE INBOARD GATE VALVE.* THE ONE INCH GLOBE VALVE WILL HAVE TEST PRESSURE APPLIED UNDER THE SEAT; HOWEVER, THE DIFFERENCE BETWEEN TESTING A ONE INCH GLOBE VALVE OVER OR UNDER THE SEAT IS CONSIDERED NEGLIGIBLE.

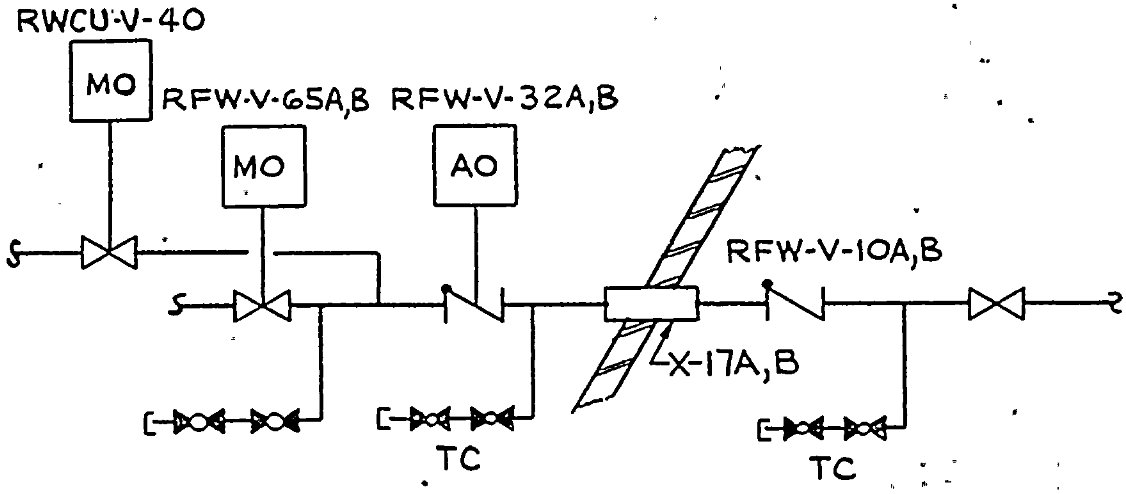
* THE GATE AND BUTTERFLY VALVES ARE BECAUSE OF SYMMETRY OF DESIGN AND BECAUSE OF CONSTRUCTION EQUALLY LEAK TIGHT IN EITHER DIRECTION. THIS FACT HAS BEEN CONFIRMED BY REVIEW OF LEAKAGE TEST DATA AND OTHER INFORMATION SUPPLIED BY THE VALVE MANUFACTURERS.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NOTES ON TYPE C TESTING	FIGURE 6.2-31a



NOTE: SEE NOTE 4 ON FIG 6.2-31a

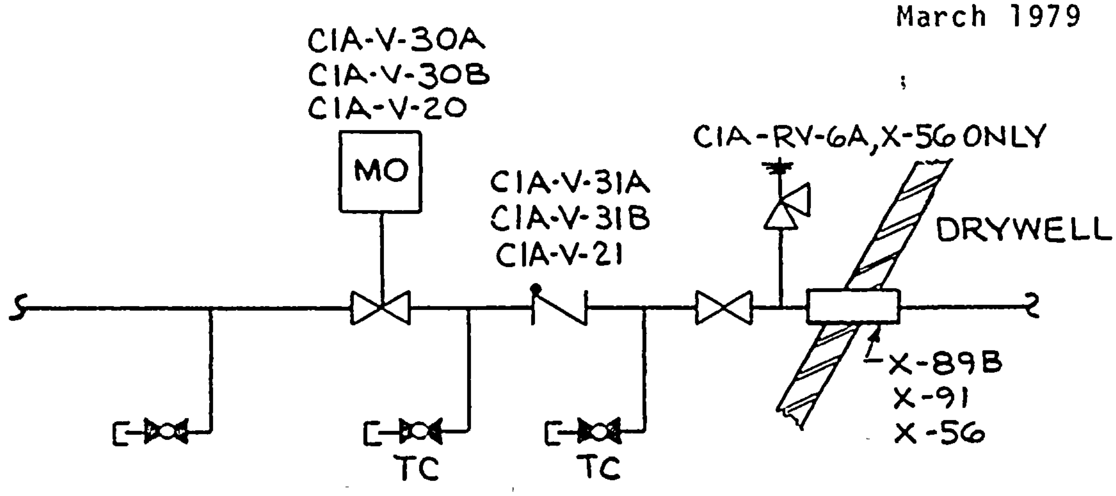
X-53 DRYWELL PURGE SUPPLY
X-66 WETWELL PURGE SUPPLY



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

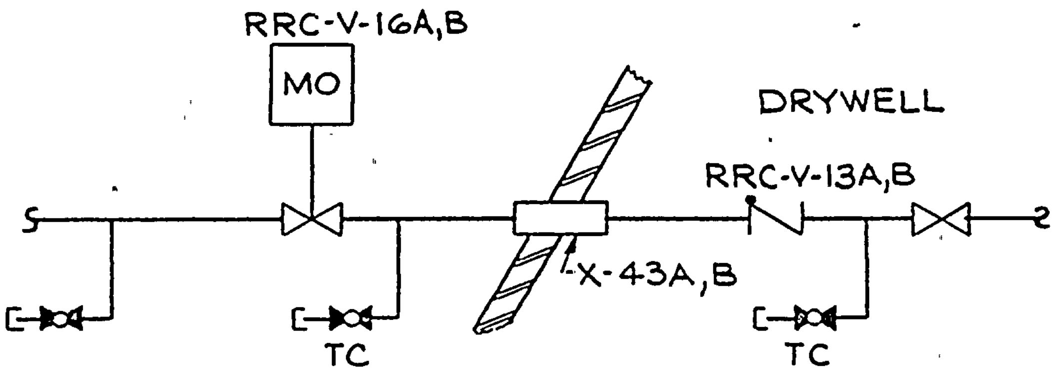
REACTOR FEEDWATER LINES

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-53, X-66, X-17A & X-17B	FIGURE 6.2-31b



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

CONTAINMENT INSTRUMENT AIR



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

RRC PUMP SEAL PURGE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS - X-89B, X-91, X-56 & X-43A, X-43B	FIGURE 6.2-31c

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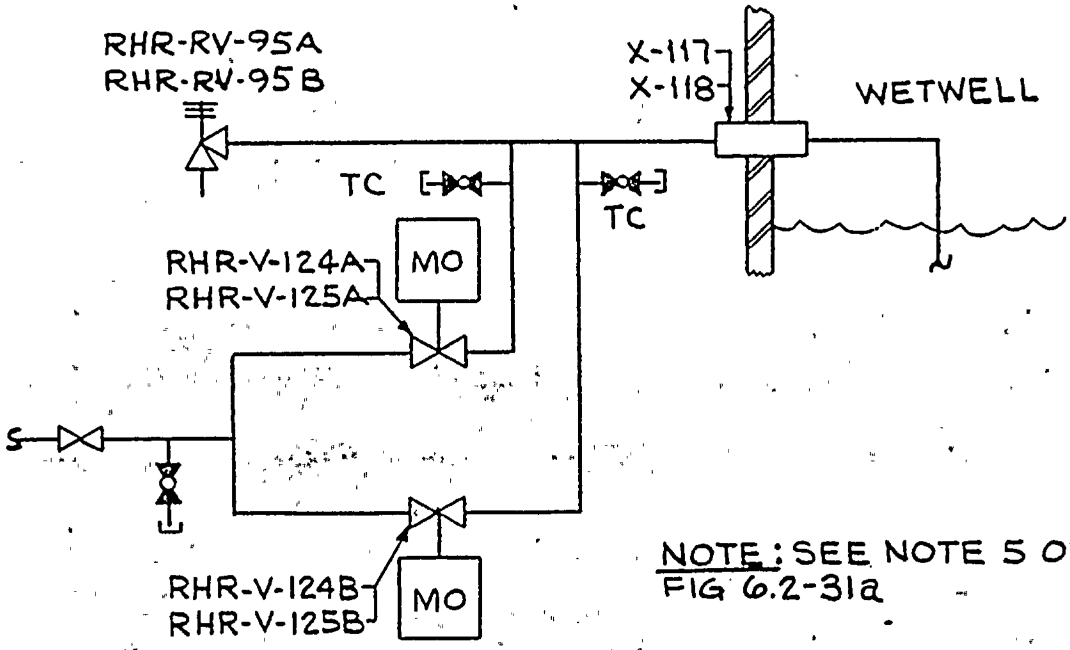
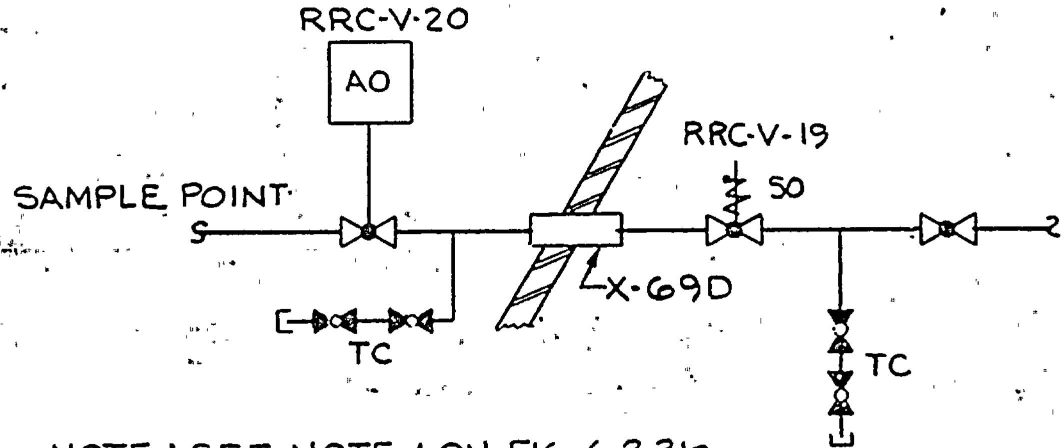


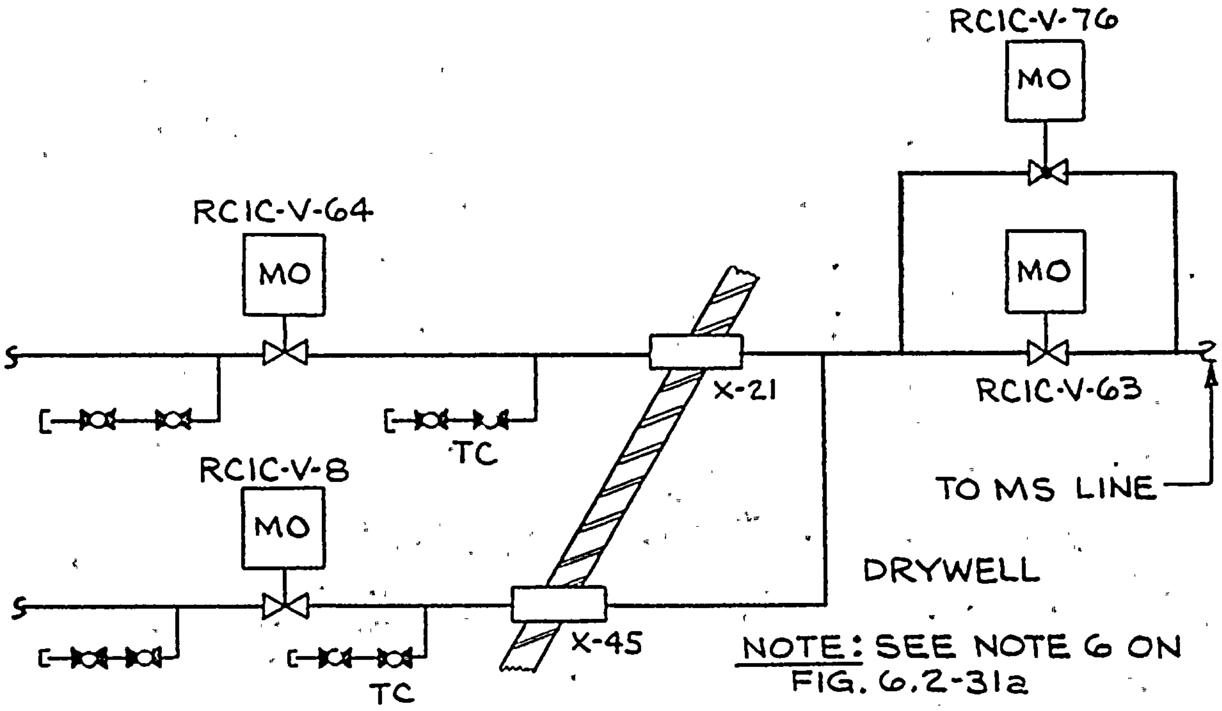
RHR STEAM RELIEF LINES



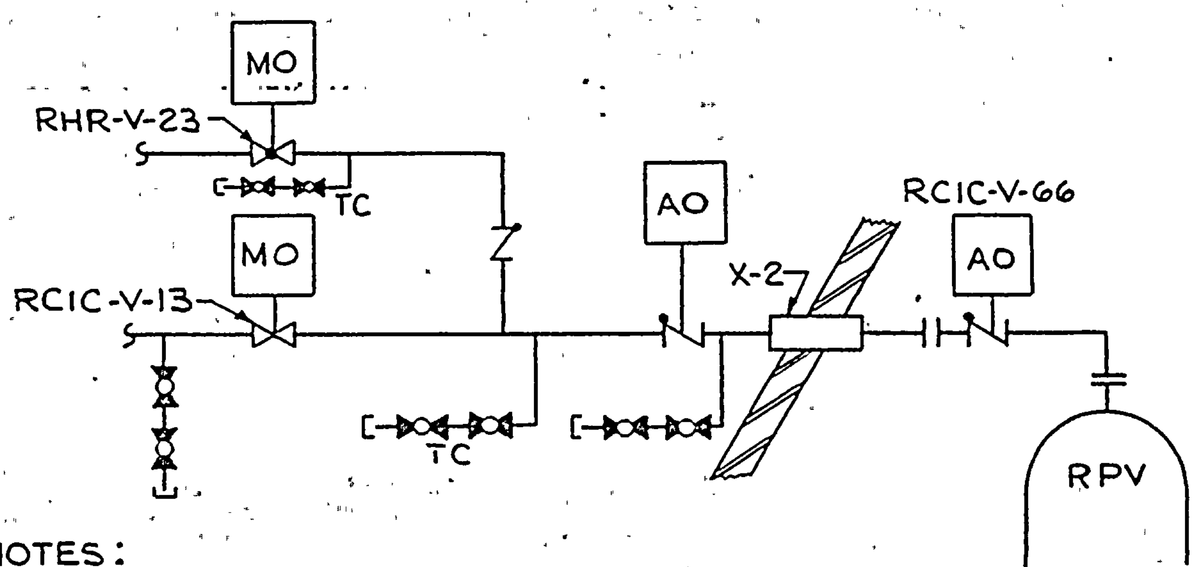
NOTE: SEE NOTE 1 ON FIG. 6.2-31a

RRC SAMPLE LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-117, X-118 & X-69D	FIGURE 6.2-31d



STEAM TO RCIC TURBINE & RHR HEAT EXCHANGER



NOTES:
RCIC-V-66 WILL BE "BENCH TESTED" ONCE THE LINE IS REMOVED FOR REFUELING. RHR-V-23 AND RCIC-V-13 CAN BE TESTED ONCE THE FLANGED CONNECTION IS BLANKED OFF AS PER NOTE 1 ON FIG. 6.2-31a

RCIC/RHR HEAD SPRAY

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-21, X-45 & X-2	FIGURE 6.2-31e

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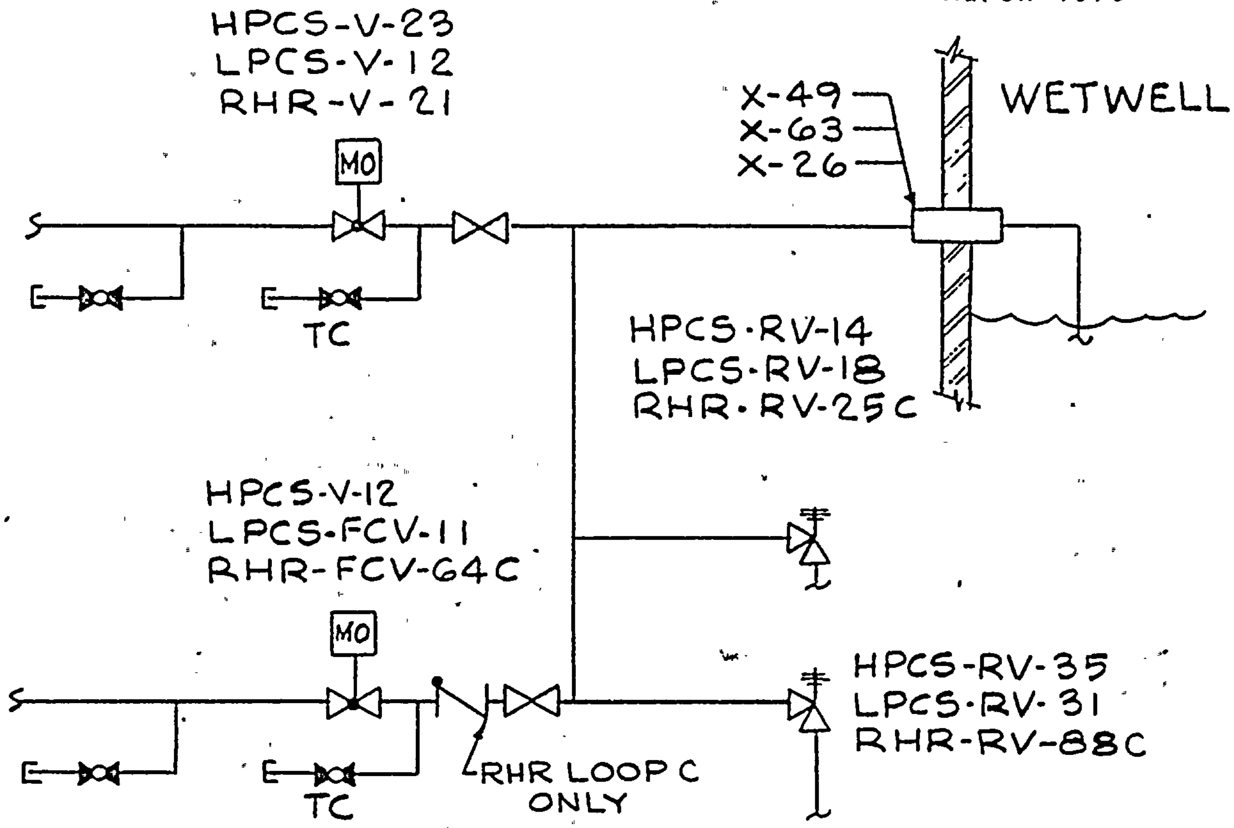
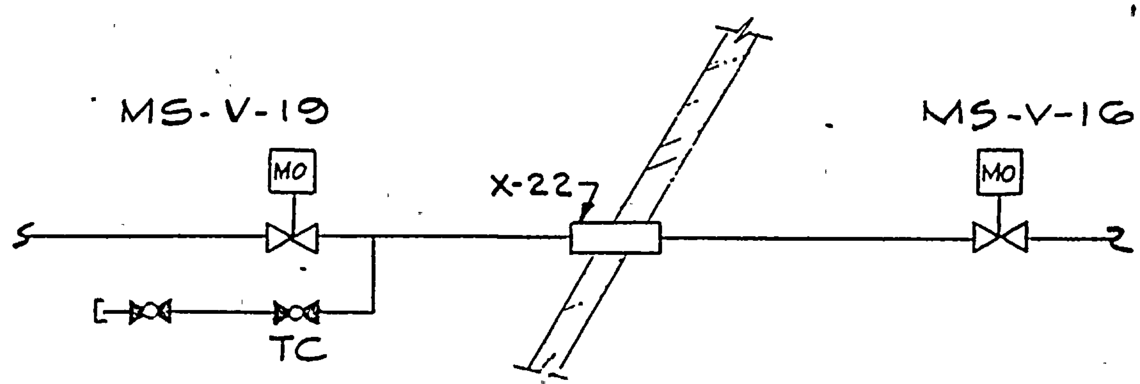


NOTE: SEE NOTE 1 ON FIG. 6.2-31a

X-49 HPCS TEST LINE

X-63 LPCS TEST LINE

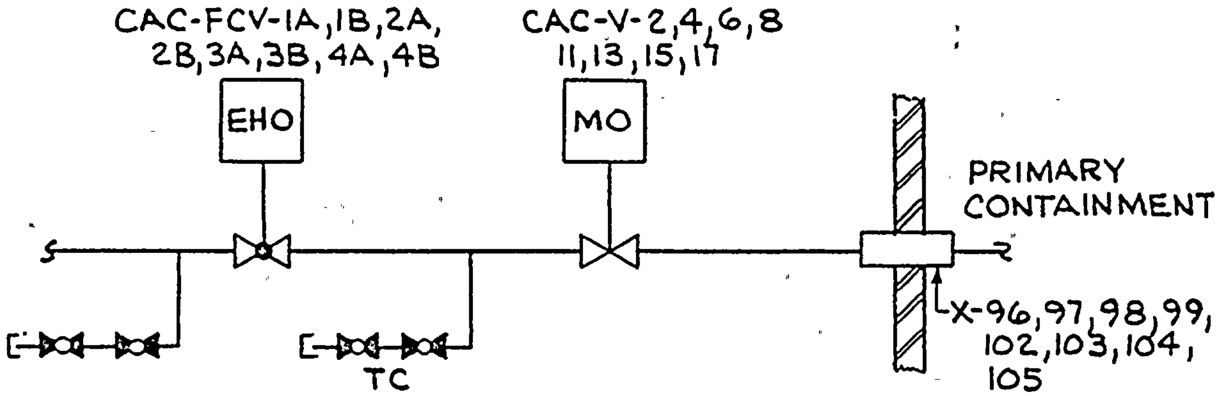
X-26 RHR LOOP C TEST LINE



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

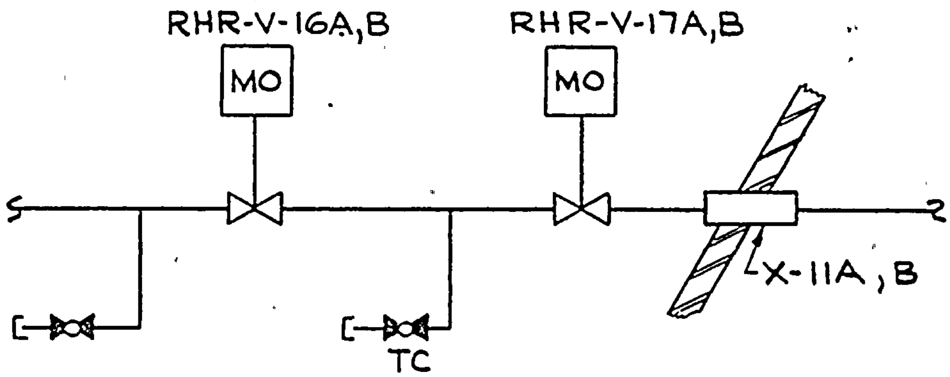
MS DRAIN LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-49, X-63 X-26 & X-22	FIGURE 6.2-31f



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

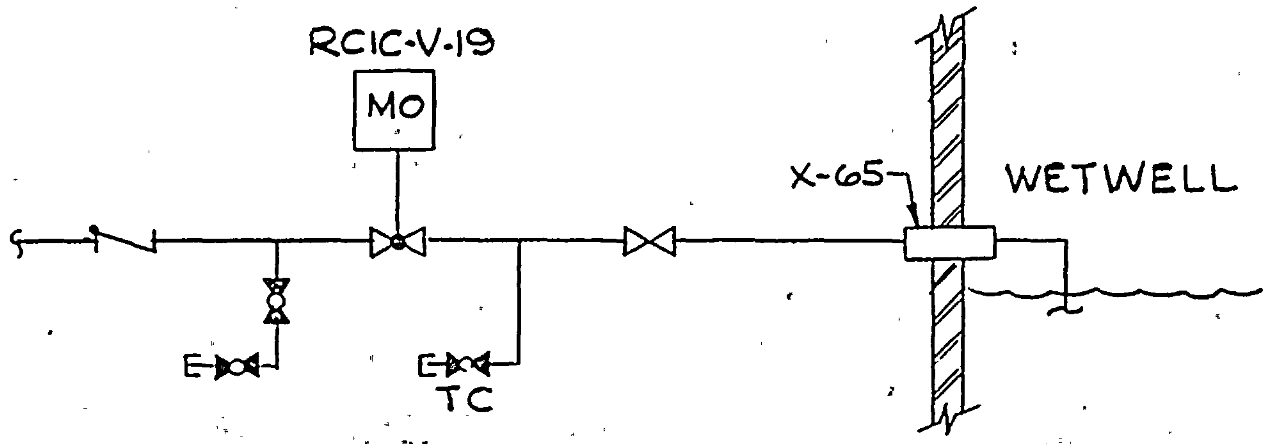
CAC SYSTEM



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

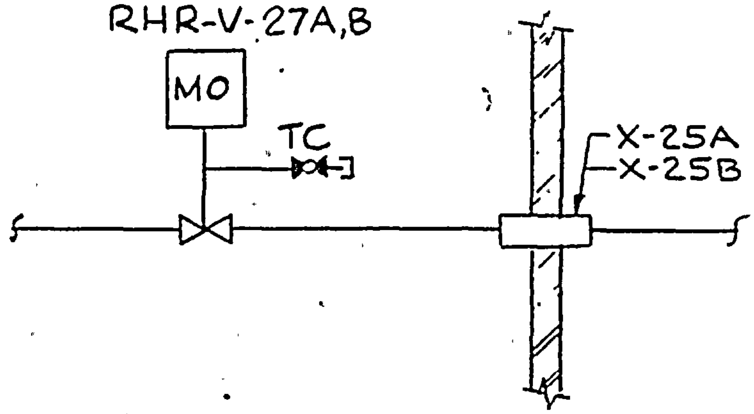
RHR DRYWELL SPRAY

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-96, X-97, X-98 X-99, X-102, X-103, X-104, X-105, X-11A & X-11B	FIGURE 6.2-31g



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

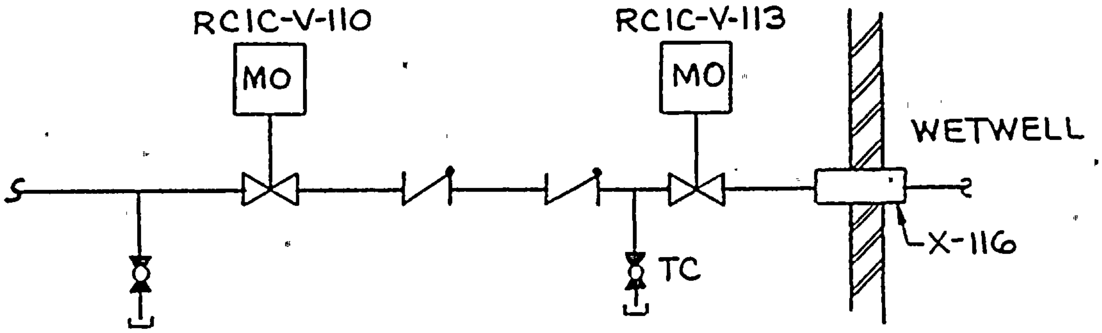
RCIC PUMP MIN. FLOW



NOTE: SEE NOTE 2 ON FIG. 6.2-31a

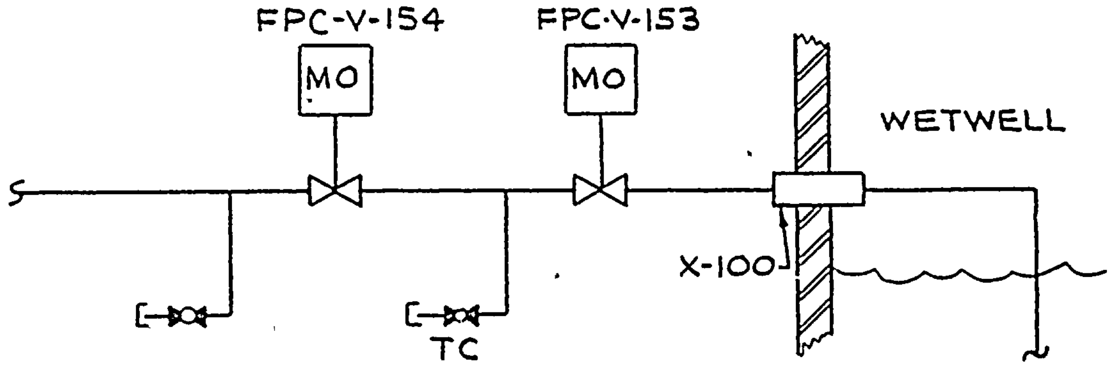
RHR WETWELL SPRAY

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-65, X-25A & X-25B	FIGURE 6.2-31h



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

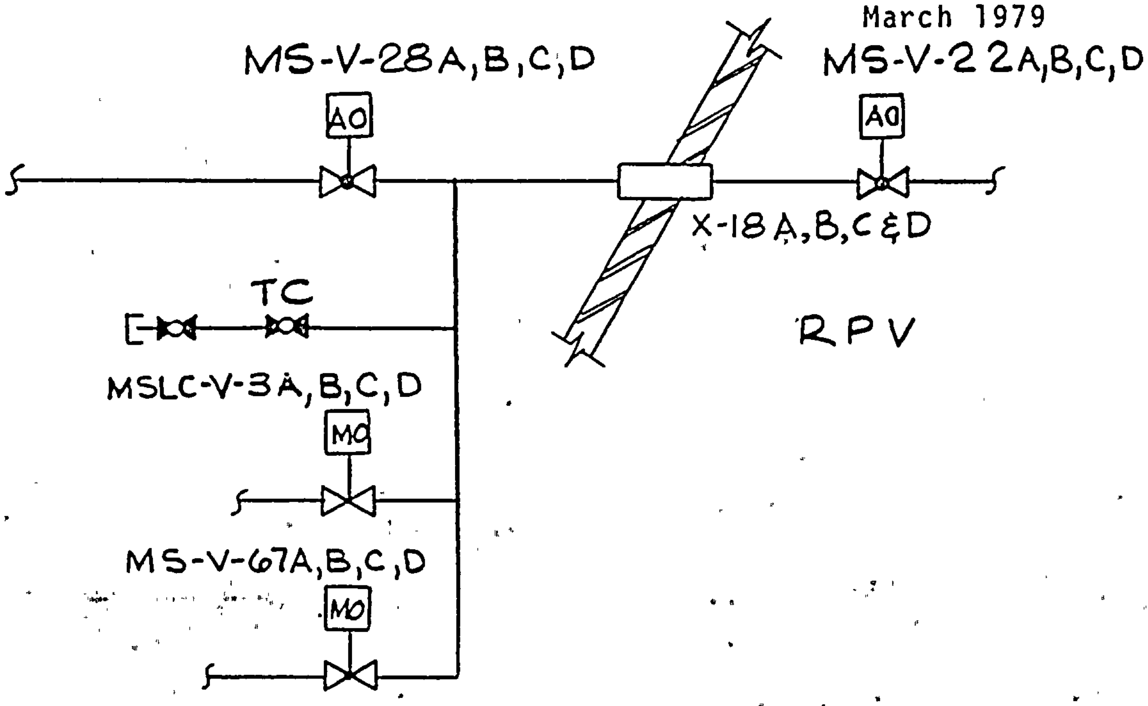
RCIC TURBINE EXHAUST VACUUM BREAKER



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

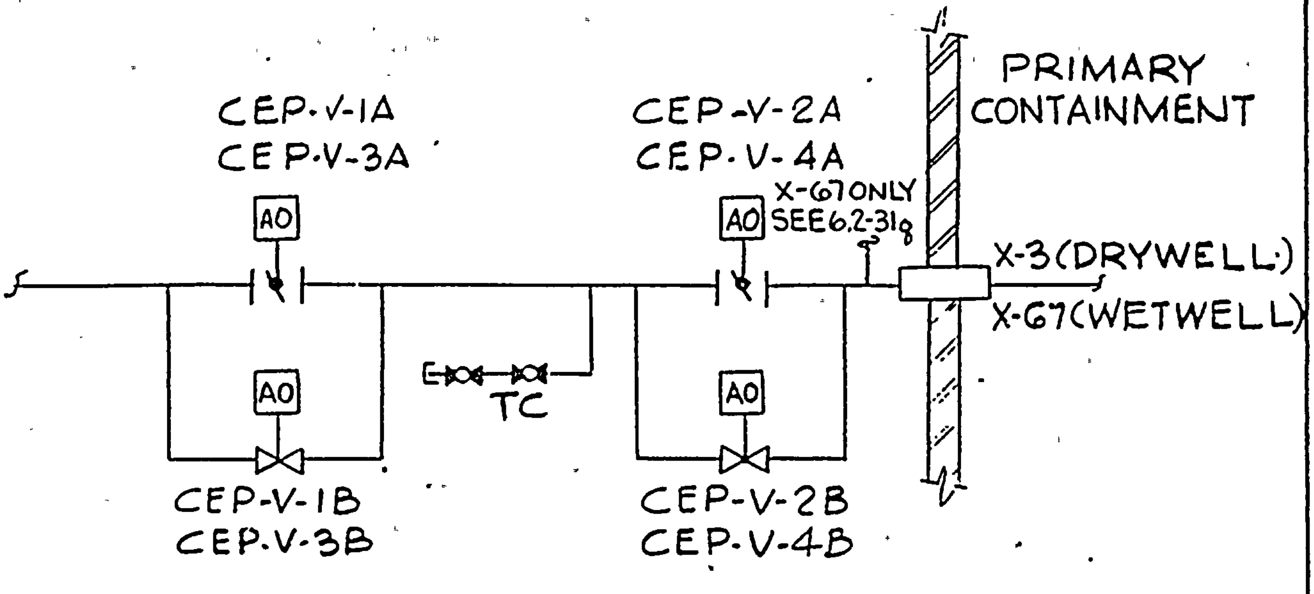
SUPPRESSION POOL CLEAN-UP SUCTION LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-116 & X-100	FIGURE 6.2-31 i



NOTE: SEE NOTE 3 ON FIG 6.2-31a

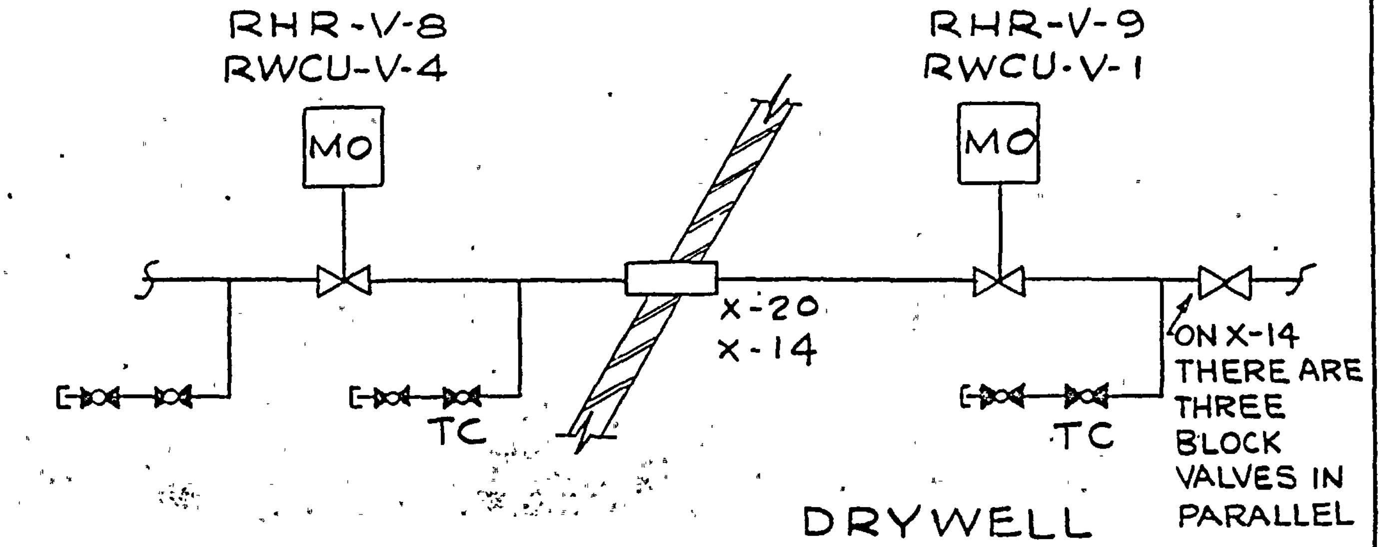
MAIN STEAM LINES



NOTE: SEE NOTE 4 ON FIG 6.2-31a

X-3 DRYWELL PURGE EXHAUST
X-67 WETWELL PURGE EXHAUST

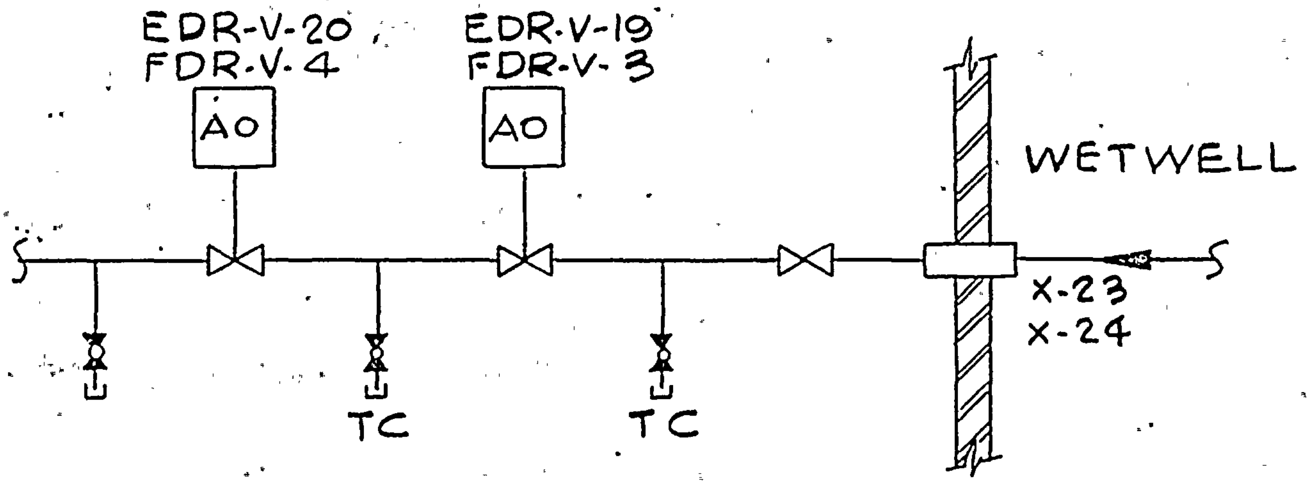
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-18A, X-18B, X-18C, X-18D, X-3 & X-67	FIGURE 6.2-31j



NOTE: SEE NOTE 1 ON FIG 6.2-31a

X-20 RHR SHUTDOWN COOLING SUPPLY

X-14 RWCU SUCTION



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

X-23 EDR FROM PRIMARY CONTAINMENT

X-24 FDR FROM PRIMARY CONTAINMENT

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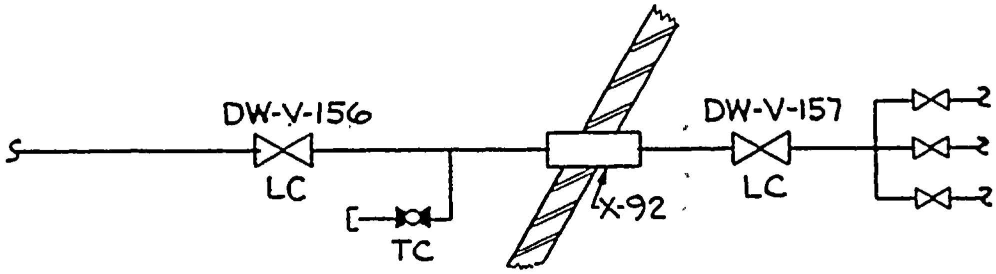
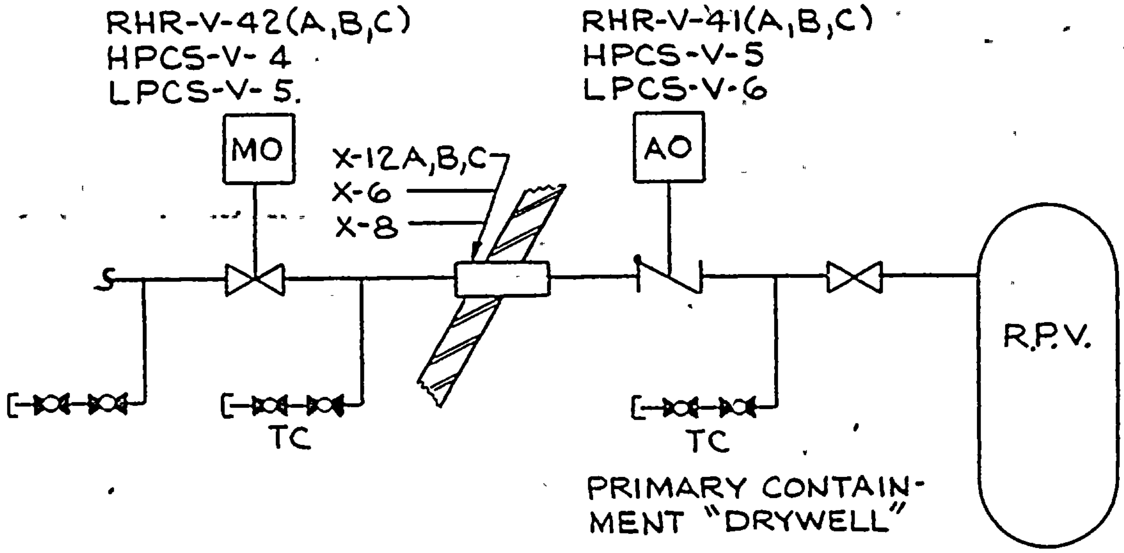
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-20, X-14, X-23 & X-24	FIGURE 6.2-31k



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

DW SYSTEM



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

X-12A RHR LOOP A LPCI TO RPV

X-12B RHR LOOP B LPCI TO RPV

X-12C RHR LOOP C LPCI TO RPV

X-6 HPCS TO RPV

X-8 LPCS TO RPV

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-92, X-12A, X-12B, X-12C, X-6 & X-8	FIGURE 6.2-31l

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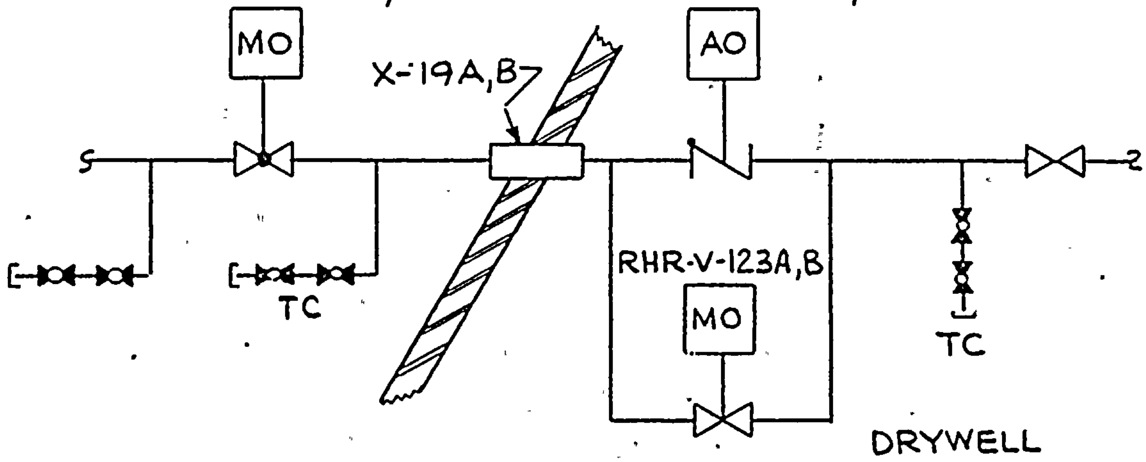
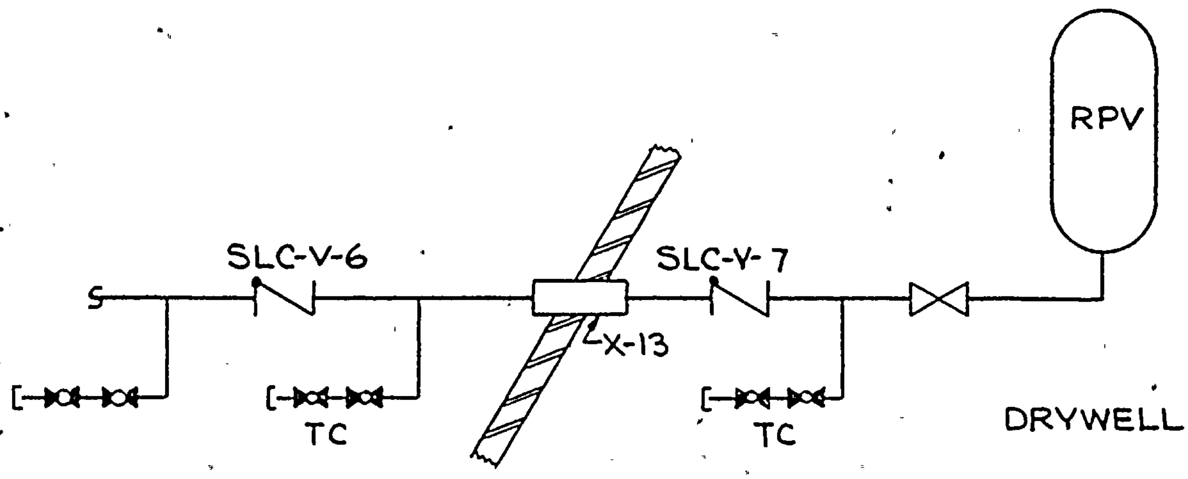


NOTE: SEE NOTE 1 ON FIG. 6.2-31a

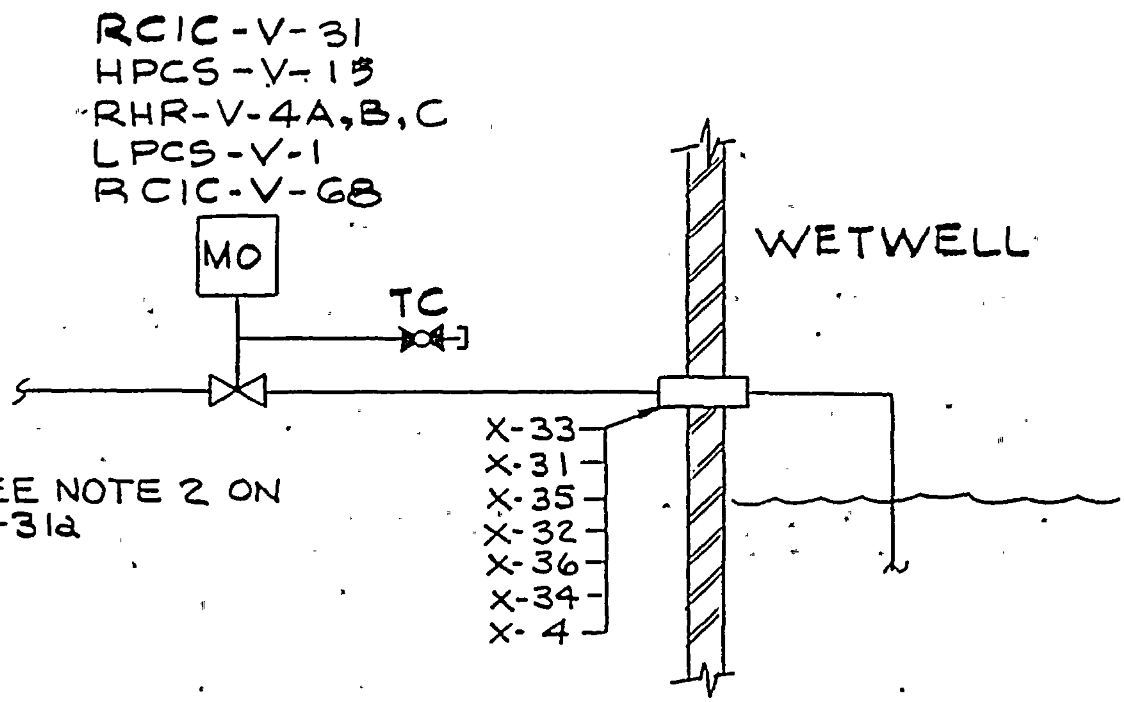
RHR SHUTDOWN COOLING RETURN



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

SLC SYSTEM INJECTION LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-19A, X-19B & X-13	FIGURE 6.2-31m



NOTE: SEE NOTE 2 ON FIG. 6.2-31a

X-33 RCIC PUMP SUCTION FROM SUPPRESSION POOL

X-31 HPCS PUMP SUCTION FROM SUPPRESSION POOL

X-35 RHR "A" PUMP SUCTION FROM SUPPRESSION POOL

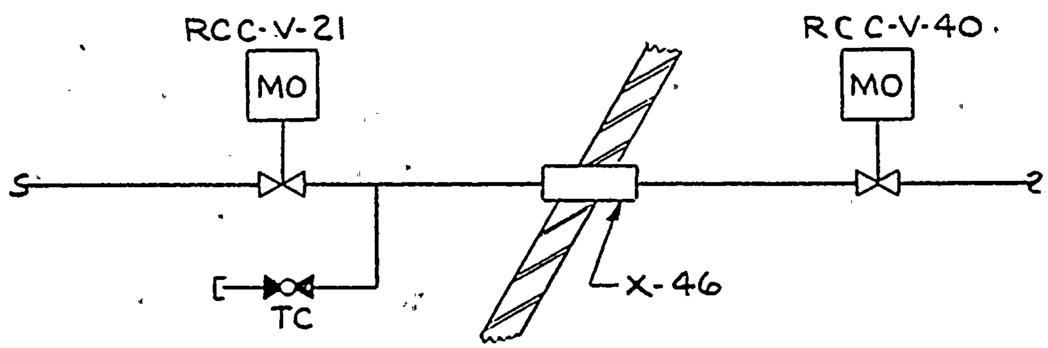
X-32 RHR "B" PUMP SUCTION FROM SUPPRESSION POOL

X-36 RHR "C" PUMP SUCTION FROM SUPPRESSION POOL

X-34 LPCS PUMP SUCTION FROM SUPPRESSION POOL

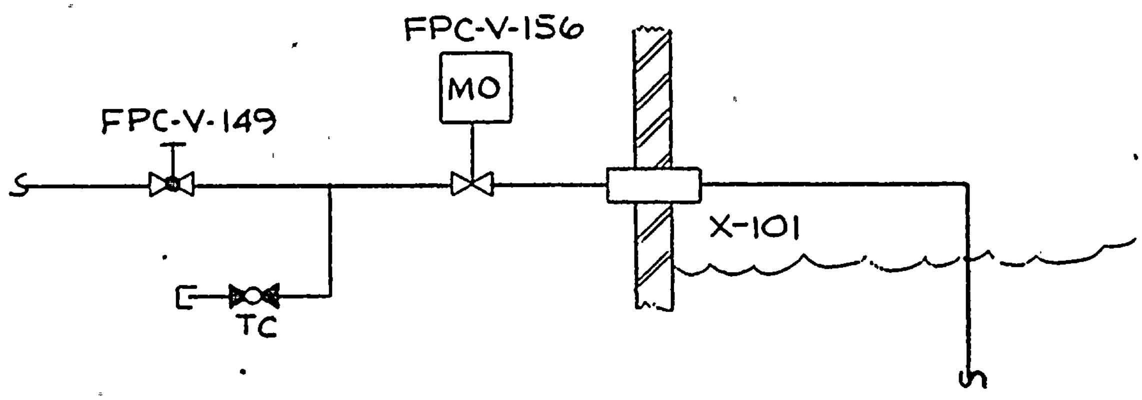
X-4 RCIC TURBINE EXHAUST

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-33, X-31, X-35, X-32, X-36, X-34, X-4	FIGURE 6.2-31n



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

RCC RETURN LINE

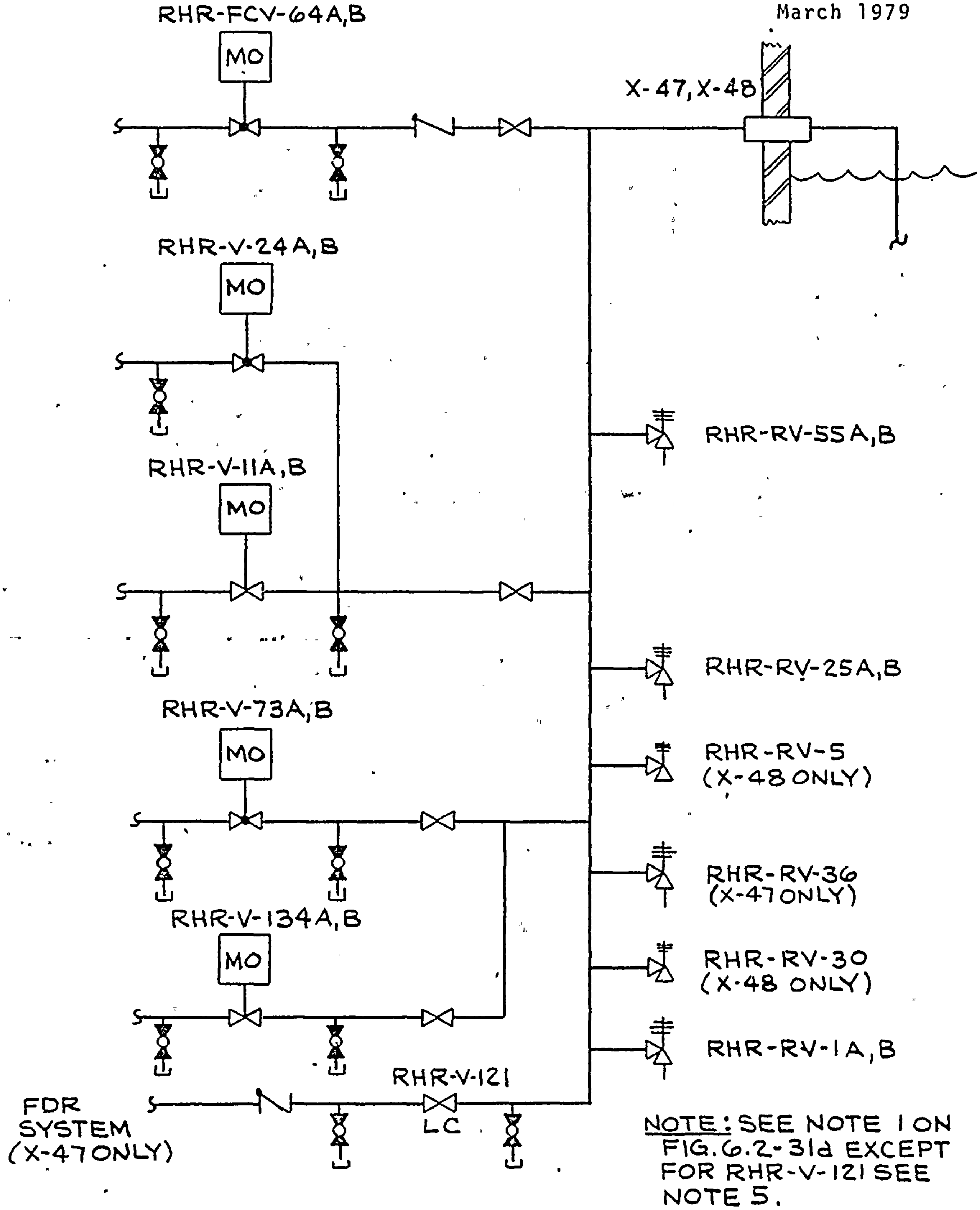


NOTE: SEE NOTE 4 ON FIG. 6.2-31a

SUPPRESSION POOL CLEAN-UP RETURN LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-46 & X-101	FIGURE 6.2-31o

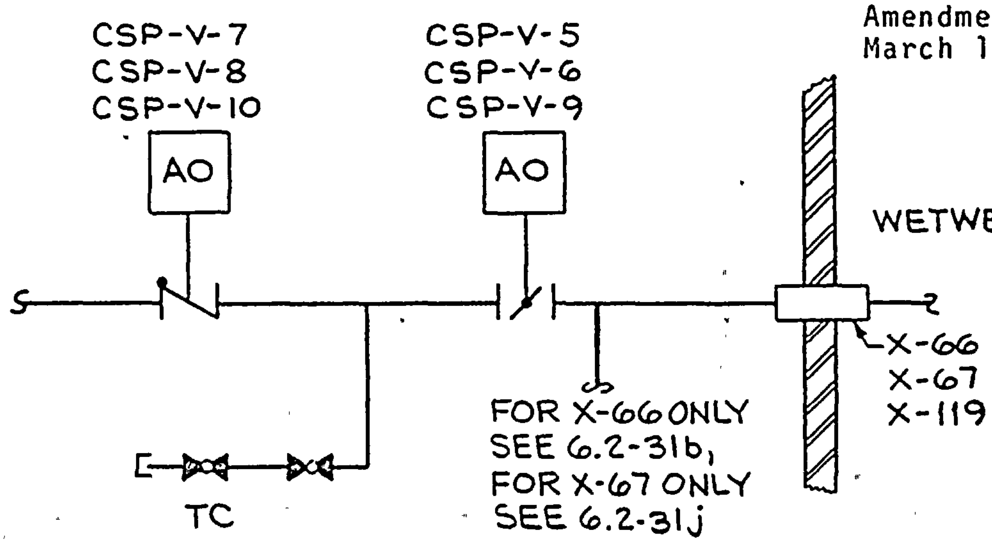
Amendment No. 3
March 1979



RHR COMBINED RETURN LINE
TO SUPPRESSION POOL

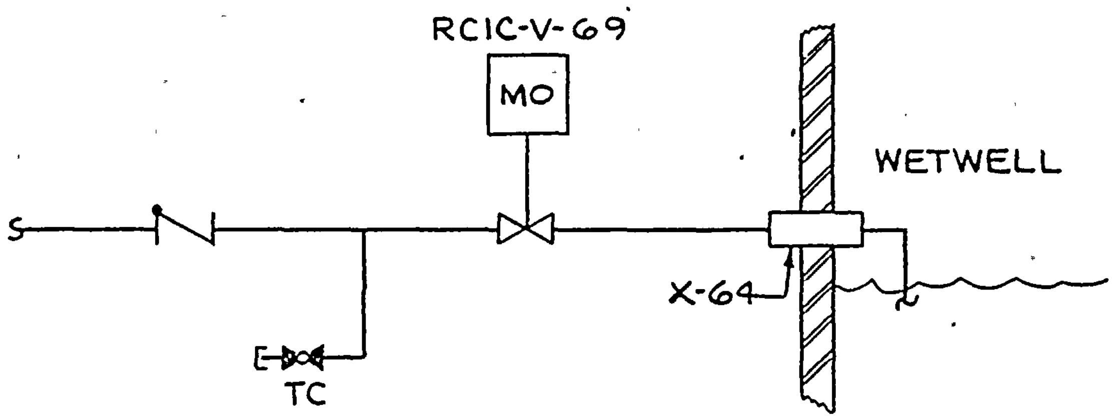
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-47 AND X-48	FIGURE 6.2-31p

Amendment No. 3
March 1979



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

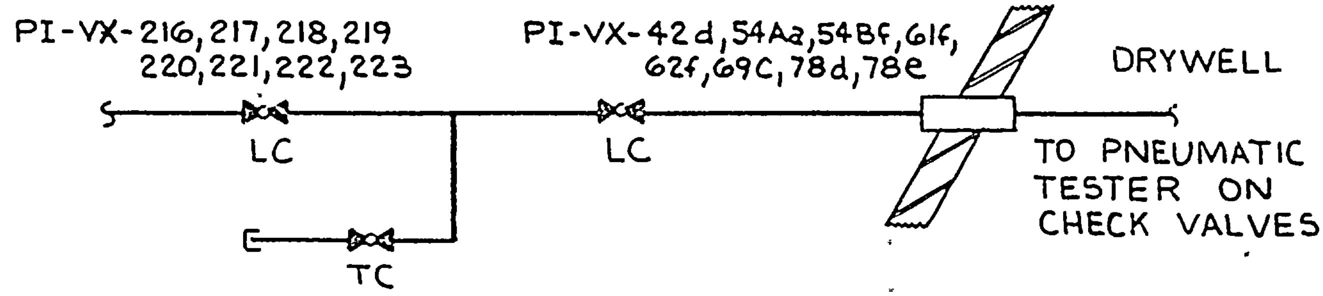
REACTOR BUILDING TO WETWELL VACUUM RELIEF



NOTE: SEE NOTE 5 ON FIG. 6.2-31a

RCIC VACUUM PUMP DISCHARGE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-66, X-67, X-119 & X-64	FIGURE 6.2-31q



NOTE: SEE NOTE 4 ON FIGURE 6.2-31a

X-42d AIR LINE FOR TESTING RHR-V-50A

X 54Aa AIR LINE FOR TESTING RCIC-V-66

X-54Bf AIR LINE FOR TESTING RHR-V-41B

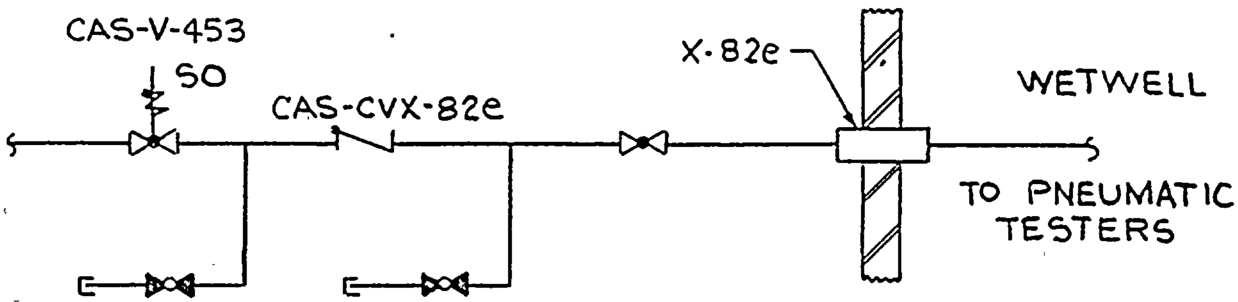
X-61f AIR LINE FOR TESTING RHR-V-41A

X-62f AIR LINE FOR TESTING RHR-V-41C

X-69c AIR LINE FOR TESTING RHR-V-50B

X-78d AIR LINE FOR TESTING LPCS-V-6

X-78e AIR LINE FOR TESTING HPCS-V-5



NOTE: SEE NOTE 1 ON FIGURE 6.2-31a

AIR LINE FOR TESTING WETWELL TO DRYWELL VACUUM BREAKERS

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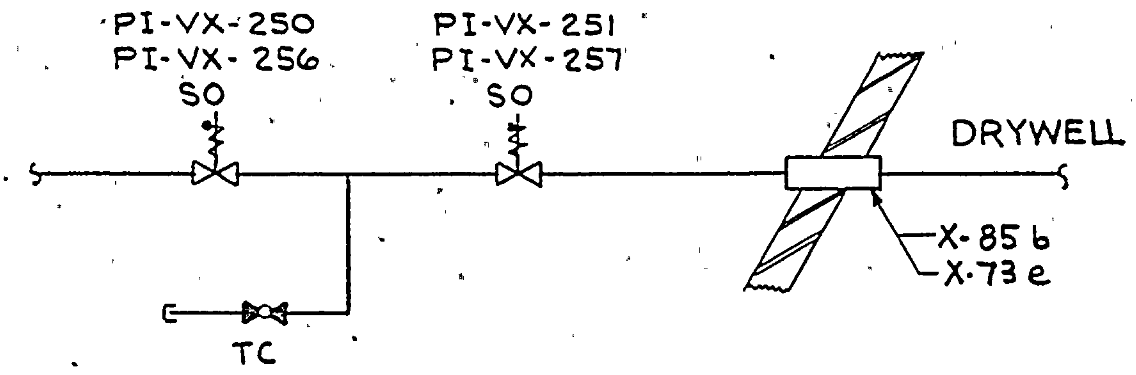
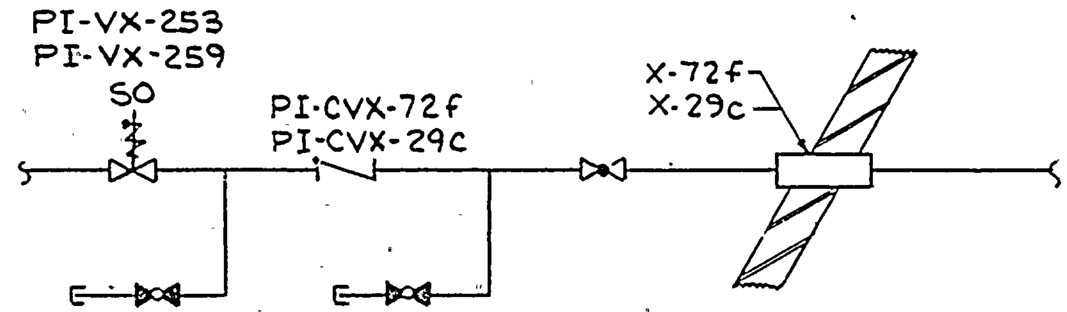
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-42d, 54Aa, 54Bf, 61f, 62f, 69c, 78d, 78e, & 82e	FIGURE 6.2-31r



NOTE: SEE NOTE 4 ON FIGURE 6.2-31a

RADIATION MONITOR SUPPLY LINE DIVISION A

RADIATION MONITOR SUPPLY LINE DIVISION B

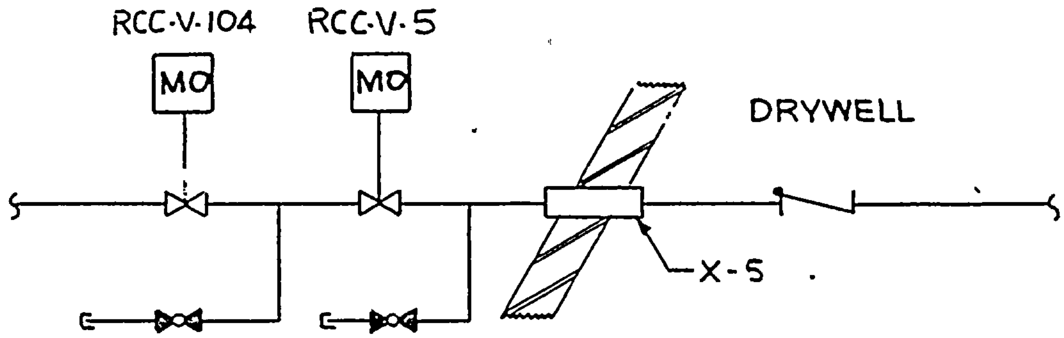


NOTE: SEE NOTE 1 ON FIGURE 6.2-31a

RADIATION MONITOR RETURN LINE DIVISION A

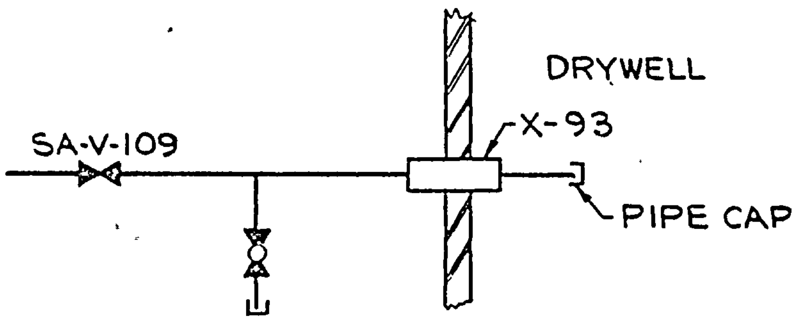
RADIATION MONITOR RETURN LINE DIVISION B

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-85b, X-73e, X-72f, & X-29c	FIGURE 6.2-31S



NOTE: SEE NOTE 4 ON FIGURE 6.2-31a

RCC SUPPLY LINE



SERVICE AIR FOR MAINTENANCE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATION X-5 & X-93	FIGURE 6.2-31t

E. DESIGN REQUIREMENTS FOR CONTAINMENT ISOLATION BARRIERS

Question: Discuss the extent to which the quality standards and seismic design classification of the containment isolation provisions follow the recommendations of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Water-Containing Components of Nuclear Power Plants", and 1.29, "Seismic Design Classification".

Response: Containment isolation provisions follow the FSAR quality group classification which meets the requirements of 10 CFR 50 and Regulatory Guide 1.26, Revision 3. Containment isolation provisions follow the requirements of Regulatory Guide 1.29 for Seismic Design Classification. See Section 3.2.2 including Table 3.2-1 for detailed application on WPPSS Nuclear Project No. 2.

F. PROVISIONS FOR TESTING

Question: Discuss the design provisions for testing the operability of the isolation valves.

Response: Isolation valve and piping arrangements have been designed to meet 10 CFR 50, Appendix J, requirements and IWV program requirements of ASME Section XI. Refer to FSAR Figures 6.2-31a to 6.2-31t for arrangement of containment isolation valves and provisions for testing.

G. CODES, STANDARDS, AND GUIDES

Question: Identify the codes, standards, and guides applied in the design of the containment isolation system and system components.

Response:

- o Containment isolation valves and associated piping and penetration meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Classes 1 or 2, as applicable.
- o Isolation valving for instrument lines which penetrate the containment conform to the requirements of Regulatory Guide 1.11, 1971, Rev. 0.
- o Containment isolation valve closure speeds limit radiological effects from exceeding guideline values established by 10 CFR 100.
- o The Design of isolation valving for lines penetrating the containment follows the requirements of 10 CFR 50, Appendix A of General Design Criteria 54, 55, 56 and 57 as noted in Table 6.2-16 of the WNP-2 FSAR. (See Section 6.2.4 of the FSAR for further detail.)

[illegible]

H. NORMAL OPERATING MODES AND ISOLATION MODES

Question: Discuss the normal operating modes and containment isolation provision and procedures for lines that transfer potentially radioactive fluids out of the containment.

Response: The containment isolation systems, in general, close those fluid lines penetrating containment that support systems not required for emergency operation. Those fluid lines penetrating containment which support engineered safety feature systems have remote manual isolation valves which may be closed from the control room, if required. See WPPSS Nuclear Project No. 2 FSAR Section 6.2.4 and Table 6.2-16 for system by system description of isolation capabilities for meeting 10 CFR 50, Appendix A Criteria 54, 55, 56, 57 and Regulatory Guide 1.11.

ATTACHMENT 3

Washington Public Power Supply System

Nuclear Project Number 2

Docket Number 50-397

Response to Bulletins and Orders Task Force Requests For Additional Information Concerning NEDO - 24708 Question Set D

Question 1. With regard to Tables 2.1.4a through 4.1.4n which provide a description, in matrix form, of system initiation, permissives, manual valve lineups, etc., it is noted that additional valves installed by AE are not included. These Tables should be complete. Furthermore, are they administratively controlled?

Response The statement that additional valves installed by the AE are not included is not applicable to WNP-2. All valves were included in the analysis for WNP-2. Furthermore, all valves are administratively controlled.

Question 2. Discrepancy between Table 2.1-1 and 2.1-2b regarding existence of FWCI for Dresden 1.

Response NA to WNP-2.

Question 3. In Figures 2.1.2 and 2.1.5, why are turbine stop valves and control valves shown open for RCIC and closed for HPCI System?

Response NA as only the RCIC pump is turbine driven on WNP-2. WNP-2 has HPCS rather than HPCI and its pump is driven by an electric motor.

Question 4. Table 2.1-2a under Items 1-4, 4-4, and 14-4, it is noted that some plants require on-site AC power for small break protection. Prolonged operation of RCIC and HPCI can require AC powered space coolers. The following information is required:

(a) How long can these systems operate without space coolers?

(b) What is operating temperature limit w/o coolers?

(c) Power source for coolers.

(d) What specific components in each system require cooling and temperature limitation on component?

Response

1. NA with respect to HPCI as WNP-2 does not have this system.

2. RCIC

   a. NA. Space cooling for the RCIC system is provided by the reactor building heating and ventilating system during normal operations and by the reactor building emergency cooling and critical electrical equipment area cooling systems. All components of the latter system are designed as engineered safety features and powered from the emergency power buses.

   b. Maximum allowable ambient temperature for the RCIC pump room - 148^{o}F (Ref. - FSAR Vol 16, Sec. 9.4.9).

c. o Reactor building heating and ventilating system - 460 V AC from normal on-site or off-site source.

o Reactor building emergency cooling and critical electrical equipment cooling system -

460 V AC from normal on-site or off-site source, backup off-site site or on-site standby source.

d. Oil cooling - maximum oil temperature 160^{o}F.

Question 5. Table 2.1a Item 13-4, Why doesn't CST require power for level indication?

Response Level indication is provided in the main control room. However, transfer of HPCS and RCIC suctions from CST to suppression pool is initiated automatically by level switches. DC power is required for these instrument and control functions.

Question 6. Table 2.1-2a Items 1-8, 2-8, 3-8, 4-8, 5-8, 6-8, 9-8 identify auxiliary systems that may require cooling for long-term operation. Answer questions 4a-d with regard to auxiliary systems.

Response a. 1. RCIC - No auxiliary systems requiring cooling.

2. Isolation Condenser - NA - WNP-2 does not have this system.

3. HPCS - If off-site power is lost, power is supplied by the HPCS diesel generator. Cooling for the diesel generator, the diesel generator building coolers and the HPCS pump room cooler is supplied by the HPCS service water system. The question is NA as the latter and the room coolers are designed as engineered safety features and powered from off-site AC or from the HPCS diesel generator.

4. HPCI - NA - WNP-2 does not have this system.

5. LPCS - The LPCS system requires standby service water for pump motor cooling. The question is NA as the standby service water pump house ventilating system is designed as engineered safety features system and powered from an essential power bus.

6. LPCI - The LPCI system (Mode A of RHR system) requires standby service water for pump seal cooling. See comments under 5. above.

9. RHR system - Same as 6 above.

b. 1. RCIC - NA

2. Isolation condenser - NA

3. HPCS - Standby service water pump house - maximum allowable ambient temperature - 114°F. HPCS diesel generator building maximum allowable ambient temperature - 120°F.

4. HPCI - NA

5. LPCS - Standby service water pump house maximum allowable ambient temperature - 114°F.

6. LPCI - Same as 5 above.

9. RHR system - same as 5 above.

C. 1. RCIC - NA.

2. Isolation condenser - NA.

3. HPCS - 460 V AC from normal on-site or off-site source, back-up off-site source or on-site standby source.

4. HPCI - NA.

5. LPCS - Same as 3 above.

6. LPCI - Same as 3 above.

9. RHR - Same as 3 above.

D. 1. NA

2. NA

3. NA

4. NA

5. NA

6. NA

9. NA

Question 7. Table 2.1-2a Item 14-8. What are requirements for feed pump ventilation system? Answer questions 4a-d with regard to this system.

Response WNP-2 does not have a feed pump ventilation system.

Question 8. Table 2.1-2a column 9b power source list is incomplete. Should identify AC requirements and if on-site or off-site, i.e., power source for auxiliary systems not identified.

Response Column 9b "Power Source" designates the power requirements for the "automatic startup logic" in Column 9a. The power sources required for system operation are listed in Column 4. Those safety systems and their respective auxiliary support systems will be powered by on-site AC power (standby power) upon loss of off-site AC power. The on-site AC power consists of two divisional diesel generators. Also, the HPCS system has a self-contained independent diesel generator set designated division 3.

Question 9. Table 2.1-2a and 2.1-2b Column 11, manual actions required and how long they take is a short-term item that was not addressed.

Response The heading for Column 11 in Tables 2.1-2a and 2.1-b was intended to say the following:

> "Can manual initiation of the system be done in the control room? If not, what actions are required and how long do they take?"

Since all answers to the first question are yes, the second question is NA.

Question 10. Table 2.1-2a Column 12, there appears to be an inconsistency between note X-12 which states that logic system functional tests and surveillance testing of systems may impede systems for auto initiating and response as given in Column 12.

Response <u>System</u>

1. RCIC The system will temporarily be made inoperable during the implementation of test. Whenever the system is made inoperable because of a required test, the other systems or components that are required operable will be considered operable if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable.

2. Isolation Condenser - NA - WNP-2 does not have this system.

3. HPCS - Same response to RCIC (#1) is applicable to HPCS.

4. HPCI - NA - WNP-2 does not have this system.

5. LPCS - Same response to RCIC (#1) is applicable to LPCS.

6. LPCI - Same response to RCIC (#1) is applicable to LPCI.

7. ADS - ADS circuitry is capable of accomplishing its protective action with one operable trip system.

8. SRV - NA

9. RHR - including shutdown cooling, suppression pool cooling, containment spray modes - NA

10.-15 NA

NOTE: Functional testing of ECCS is administratively controlled such that only one ECCS system may be tested at one time.

Question 11. Table 2.1-2a Column 13, inconsistency between response and notes. Plants for which operation is performed should be identified. Also for ADS doesn't operator eventually have to close ADS valves?

Response

System	Operator Action Required Within Two Hours
1. RCIC	Once the reactor water level is recovered, the RCIC will be manually operated to maintain the water level.
2. Isolation Condenser	NA as WNP-2 does not have this system.
3. HPCS	Note 3-13 does not apply to WNP-2 as there is no throttle capability provided in the WNP-2 HPCS. No operator action is mandatory as the HPCS injection valve will automatically close when the reactor water level reaches the "high water level" and will reopen when the water level returns to the "low water level" However, once the reactor water level is recovered it is expected that the operator will exercise manual control of the HPCS, in conjunction with manual operation of other systems, to maintain reactor water level and to establish long term post-accident operation.
4. HPCI	NA as WNP-2 does not have this system.
5. LPCS	No operator action is mandatory. However, once the reactor water level is recovered it is expected that the operator will exercise manual control of the LPCS, in conjunction with manual operation of other systems, to maintain reactor water level and to establish long term post-accident operation.
6. LPCI	No operator action is mandatory. However, once the reactor water level is recovered it is expected that the operator will exercise manual control of the LPCI, in conjunction with manual operation of other systems, to maintain reactor water level and to establish long term post-accident operation. To obtain the full

[illegible]

suppression pool cooling capacity of the loop A and B LPCI flows the operator must close the RHR heat exchanger bypass valve (prohibited during the first ten minutes following automatic system initiation). Manual action is required if the operator decides to switch the A or B loops into the RHR suppression pool cooling or containment spray modes.

7. ADS — No operator action is mandatory. However, as a part of establishing long term post-accident operation, the operator would be expected to close the ADS Valves.

8 through 15: — NA

Question 12. Table 2.1-2a and 2.1-2b Column 17c. Identify size debris strainer will allow to pass instead of just stating strainer size is coarse.

Response

1. RCIC - 3/32"
2. Isolation Condenser - NA - WNP-2 does not have this system
3. HPCS - 3/32"
4. HPCI - NA - WNP-2 does not have this system
5. LPCS - 3/32"
6. LPCI - 3/32"
7. ADS - NA
8. SRV - NA
9. RHR - 3/32"
10. SSW - NA (Note - Column 17b should be changed to NA)
11. RPCCW - NA
12. CRDS - NA
13. CST - NA
14. Main Feed Water System - NA
15. Recirc. Pump Motor Cooling System - NA

Question 13. Table 2.1-2a, for note X-24 clarify what is meant by indirect indication on manual valves. Also identify which plants comments applicable to.

Response — All manual valves that have valve indication have direct indication.

Question 14. Table 2.1-2a, there appears to be an inconsistency between "no" responses in Column 25 and Note X-25.

Response

SYSTEM	REMOTE SHUTDOWN PANELS
1. RCIC	Yes
2. Isolation Condenser	NA - WNP-2 does not have this system
3. HPCS	No
4. HPCI	NA - WNP-2 does not have this system

5. LPCS	No
6. LPCI	No
7. ADS	No
8. SRV	Yes
9. RHR -	
Shutdown cooling	Yes
Supp. Pool cooling	Yes
Cont. spray modes	No
Stm. cond.	No
10. SSW	Yes
11. RBCCW	No
12. CRDS	No
13. CST	No
14. Main Fd. Wtr. System	No
15. Recirc. Pump/ Motor cooling system	No

Question 15. Table 2.1-2a, Column 26, identify other means of detecting leaking SRV.

Response Primary method - Temperature sensing elements on the SRV discharge piping.

Alternate methods - Operator observation of increasing water level and temperature in the suppression pool.

Question 16. Table 2.1-2a, and 2.1-2b, response incomplete. Would like to know plants that perform independent procedure verification and which do physical verification and if there is any significant differences in performance.

Response It is assumed that the question refers to column 28b.

NA as WNP-2 is not an operating plant. The subject of independent verification of system lineup for testing and return to normal condition will be addressed when the plant operating and testing procedures are developed.

Question 17. Table 2.1-1 shows Dresden I does not have ADS or FWCI. Table 2.1-2b indicates it does. Also Table 2.1-1 shows HB does not have LPCI, Table 2.1-2b indicates it does.

Response NA to WNP-2.

Question 18. Table 2.1-2b, note 2-8, how long can isolation condenser remove heat without makeup?

Response NA as WNP-2 does not have an isolation condenser.

Question 19. Table 2.1-2b, Column 9a, why does core spray have to be operating to use ADS for Humboldt Bay?

Response NA as question refers to the Humbolt Bay plant.

Question 20. Table 2.1-2b, Column 13 and 15b: What is meant by N/A for operator action under FWCI? Also note 4-15b missing.

Response NA as Table 2.1-2b applies only to BWR/I plants. Also, WNP-2 does not have FWCI.

Question 21. Table 2.1-2b column 15c, why is there no dèdicated capacity specified for HPCI?

Response NA as Table 2.1-2b applies only to BWR/I plants. Also, WNP-2 does not have HPCI.

Question 22. Table 2.1-2b, Column 13, isn't operator action required to close ADS valves? Also, why is operator action under FWCI for D-1 not applicable?

Response NA as Table 2.1-2b applies only to BWR/1 plants.

Question 23. Tables 2.1-4 for systems such as LPCI, LPCS and HPCS. Are there no trips on component malfunctions, i.e., high pump bearing temperatures or loss of coolant to pump bearing?

Response All component trips are inhibited when a "LOCA" is initiated. Therefore, Table 2.1-4a is correct with the exception of the HPCS Diesel Engine which will trip on overspeed and incomplete sequence (engine cannot start).

Question 24. Table 2.1-4d, what is source of auto isolation signal identified under trip conditions?

Response NA as WNP-2 does not have HPCI.

Question 25. Table 2.1-4m&n identify turbine and pump protection trips. Table 2.1-4m under degraded conditions, reduced capacity, what is significance of term "Open"?

Response Assume that 2.1-4 m&n means 2.1-4N-1 & 2.

TURBINE PROTECTION TRIPS

1. Manual (local and remote)
2. Overspeed
3. Low condenser vacuum
4. High H_2O level
5. Low bearing oil pressure
6. High vibration

PUMP PROTECTION TRIPS

1. Manual (local and remote)
2. Thrust bearing failure or excessive wear
3. Low bearing oil pressure
4. Low pump suction pressure
5. Pump suction valve not full open
6. High vibration

With respect to the second question - There is no term "open" in Table 2.1-4m.

Question 26. Table 2.1-5e, since Dresden I will not become operational without HPCI, shouldn't table reflect this? Also, FWCI should be included.

Response NA as the question addresses the Dresden I plant only.

Question 27. Table 2.1-5g and 2.1-5i. One diesel generator out of service missing from matrix.

Response NA as the question addresses the Duane Arnold and Hatch Unit I plants only.

Question 28. Table 2.1-5j. For Humboldt Bay does one or both ADS valves have to be out of service for plant to be shutdown?

Response NA as the question addresses the Humboldt Bay plant only.

Question 29. Table 2.1-5b. Under inoperable status failure to manually reset to start on low water level not included.

Response NA as the question addresses isolation condenser systems and WNP-2 does not have this system.

Question 30. Table 2.1-2a, Column 8. Isn't service air required as an auxiliary system to operate the main feedwater system?

Response Service air should be added to the list of auxiliary systems required for operation of the main feedwater system.

Question 31. Table 2.1-2b, Column 9. Does AC refer to on-site?

Response NA as Table 2.1-2b applies only to BWR/I plants.

Question 32. Table 2.1-2a, Column 16d. For small break is AWS required for manual operation of backup water source sufficient to prevent core uncovery if HPCI or HPCS not available?

Response It is assumed that "AWS" should be "ADS".

This question is currently being reviewed by General Electric and the owners group as a part of emergency procedure guidelines. Operator guidelines will be prepared regarding ADS initiation to ensure adequate core cooling.

Question 33. Rad monitor for isolation condenser.

Response NA as WNP-2 does not have an isolation condenser system.

ATTACHMENT 4

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NUMBER 2
DOCKET NO. 50-397

RESPONSE TO BULLETINS AND ORDERS TASK FORCE REQUESTS FOR ADDITIONAL INFORMATION CONCERNING NEDO-24708 QUESTION SET E

ITEM 1:

According to Section 3.1.1.1.2.1.6 of NEDO-24708, LPCS or LPCI must be throttled by the operator, for some plants, to insure adequate NPSH. Can these lines be orificed to achieve the same goal without compromising the adequacy of the system(s)? What are the consequences of not throttling?

RESPONSE:

The LPCS or LPCI pumps for WNP-2 do not need to be throttled to maintain adequate NPSH. The maximum suppression pool temperature following a LOCA is 220°F. The available NPSH as calculated per Regulatory Guide 1.1 for the LPCS and LPCI pumps exceeds the required NPSH for pool temperatures of 220°F or less.

ITEM 2:

Notes 5-8, 6-8, 9-8 for Table 2.1-2a state that some plants require lube oil and seal cooling. Which plants are these?

RESPONSE:

The LPCI and RHR have seal coolers and the source of water for cooling is from the Standby Service Water System.

The LPCS pumps use pump fluid for bearing and seal cooling.

Cooling is provided to the LPCS pump motor lube oil by the Standby Service Water System.

ITEM 3:

With regard to Tables 2.1.4a thru 2.1.4n which provide a description, in matrix form, of system initiation, permissives, manual valve lineups, etc., it is noted that additional valves installed by the AEs are not included. These tables must be complete. Are they administratively controlled?

RESPONSE

The statement that additional valves installed by the AE are not included is not applicable to WNP-2. All valves were included in the analysis for WNP-2. Furthermore, all valves are administratively controlled.

ITEM 4:

With regard to Items 1-4, 4-4, and 14-4 of Table 2.1-2a, it is noted that some plants require on-site AC power for small break protection. Prolonged operation of RCIC and HPCI can require AC powered space coolers. Provide the following information:

a) How long can these systems operate without space coolers?

b) What is the operating temperature limit without coolers?

c) What is the power source for coolers?

d) What specific components in each system require cooling and temperature limitation on components?

RESPONSE:

1. NA with respect to HPCI as WNP-2 does not have this system.

2. RCIC

   a. NA. Space cooling for the RCIC system is provided by the reactor building heating and ventilating system during normal operations and by the reactor building emergency cooling and critical electrical equipment area cooling systems. All components of the latter system are designed as engineered safety features and powered from the emergency power buses.

   b. Maximum allowable ambient temperature for the RCIC pump room - 148°F (Ref. - FSAR Vol. 16, Sec. 9.4.9).

   c. o Reactor building heating and ventilating system - 460V AC from normal on-site or off-site source.

      o Reactor building emergency cooling and critical electrical equipment cooling system -

         460 V AC from normal on-site or off-site source, backup off-site site or on-site standby source.

   d. Oil cooling - maximum oil temperature 160°F.

ITEM 5:

Items 1-8, 2-8, 3-8, 4-8, 5-8, 6-8 and 9-8 of Table 2.1-2a identify auxiliary systems that may require cooling for long-term operation. Answer questions 4a-d with regard to auxiliary systems.

RESPONSE:

A. 1. RCIC - No auxiliary systems requiring cooling.

2. Isolation Condenser - NA - WNP-2 does not have this system.

3. HPCS - If off-site power is lost, power is supplied by the HPCS diesel generator. Cooling for the diesel generator, the diesel generator building coolers and the HPCS pump room cooler is supplied by the HPCS service water system. The question is NA as the latter and the room coolers are designed as engineered safety features and powered from off-site AC or from the HPCS diesel generator.

4. HPCI - NA - WNP-2 does not have this system.

5. LPCS - The LPCS system requires standby service water for pump motor cooling. The question is NA as the standby service water pump house ventilating system is designed as engineered safety features system and powered from an essential power bus.

6. LPCI - The LPCI system (Mode A of RHR system) requires standby service water for pump seal cooling. See comments under 5 above.

9. RHR system - Same as 6 above.

B. 1. RCIC - NA

2. Isolation condenser - NA

3. HPCS - Standby service water pump house - maximum allowable ambient temperature - 114°F. HPCS diesel generator building maximum allowable ambient temperature - 120°F.

4. HPCI - NA

5. LPCS - Standby service water pump house maximum allowable ambient temperature - 114°F.

6. LPCI - Same as 5 above.

9. RHR system - same as 5 above.

C. 1. RCIC - NA.

2. Isolation condenser - NA.

ITEM 5 (continued)

3. HPCS - 460V AC from normal on-site or off-site source, back-up off-site source or on-site standby source.

4. HPCI - NA.

5. LPCS - Same as 3 above.

6. LPCI - Same as 3 above.

9. RHR - Same as 3 above.

D. 1. NA

2. NA

3. NA

4. NA

5. NA

6. NA

9. NA

ITEM 6:

With regard to Column 9b of Table 2.1-2a, the power source list is incomplete. Identify the AC requirements and whether on-site or off-site, i.e., the power sources for auxiliary systems are not identified.

RESPONSE:

Column 9b "Power Source" designates the power requirements for the "automatic startup logic" in Column 9a. The power sources required for system operation are listed in Column 4. Those safety systems and their respective auxiliary support systems will be powered by on-site AC power (standby power) upon loss of off-site AC power. The on-site AC power consists of two divisional diesel generators. Also, the HPCS system has a self-contained independent diesel generator set designated division 3.

ITEM 7:

With regard to Column 11 of Tables 2.1-2a and 2.1-2b, address the manual actions required and how long they take.

RESPONSE:

The heading for Column 11 in Tables 2.1-2a and 2.1-b was intended to say the following:

> "Can manual initiation of the system be done in the control room? If not, what actions are required and how long do they take?"

Since all answers to the first question are yes, the second question is NA.

ITEM 8:

With regard to Note 2-8 for Table 2.1-2b, how long can the isolation condenser remove heat without make up?

RESPONSE:

Not applicable - since WPPSS WNP-2 plant does not have the isolation condenser.

ITEM 9:

With regard to systems such as LPCI, LPCS, and HPCS, in'Tables 2.1-4, are there trips on component malfunctions, i.e., high pump bearing temperatures or loss of coolant to pump bearing?

RESPONSE:

All component trips are inhibited when a "LOCA" is initiated. Therefore, Table 2.1-4a is correct with the exception of the HPCS Diesel Engine which will trip on overspeed and incomplete sequence (engine cannot start).

ITEM 10:

One of the systems requests for information that has not been adequately addressed in NEDO-24708 is the loss of feedwater transient coupled with a stuck open SRV and loss of offsite power and diesels. From the information provided it is not possible to determine what the end result of this scenario would be. Since all the plants have various combinations of HPCI, RCIC and IC systems, SRV with varying relieving capacities, and varying stored energies, the results are plant specific. Therefore, for all the plants or plant types identified in NEDO-24708, provide the following time dependent plots for the above scenario:

a) steam and coolant inventory lost;
b) coolant temperature and pressure;
c) coolant makeup (where applicable);
d) reactor vessel water level relative to top of active fuel; and,
e) fuel and cladding temperatures.

The initial plant conditions assumed in the analyses, the time assumed for startup of the available systems and the time the RCIC and HPCI can operate before the system depressurizes below their operating conditions should be provided. In addition, identify when equilibrium conditions are achieved (core covered and water level maintained in normal operating range); if core uncovery occurs identify when, time duration, and extent of core damage (include basis).

RESPONSE:

Response to this item has been provided by G.E. in the long term information submitted on March 31, 1980 (Reference below).

Reference: Letter from R. H. Buchholz and T. D. Keenan to D. F. Ross, March 31, 1979, Subject: NUREG - 0578, Requirements 2.1.9 Implementation by BWR Owner's Group

RESPONSES TO

INSTRUMENTATION AND CONTROL SYSTEMS BRANCH (ICSB)

QUESTIONS 031.080 - 031.113

Q. 031.080 a)

Clarify the discrepancy between the definition of passive failures in electrical, instrumentation, and control systems in Sections 1.2.1.1.1.1.2.L and 3.11.2.3, 6.2.4.1.1, and 6.7.1.2.c of the FSAR.

Response:

There is no discrepancy intended. Systems designed in accordance with IEEE 279 include consideration of electrical passive failures as single active failures. Section 3.11.2.3 does not address single failure, only qualification of safety-related equipment.

Q. 031.080 b)

Clarify the discrepancy between your response to Item 031.001 (b) and Figures 7.2-1b and 7.2-1c of the FSAR.

Response:

There is no discrepancy. As stated in Question 031.001 (b) the reference to three trip logics has been revised. As you have noted by Figure 7.2-1 there are in fact two trip logics in each of the two RPS divisions, making a total of four logics.

Q. 031.080 c)

Revise Section 4.4.3.3.3 of the FSAR to provide the actual values of the measured parameters which are to be used in the WNP-2 facility. This section indicates "typical" values; at the OL stage of review, we require actual values.

Response:

Section 4.4.3.3.3 is revised as follows:*

4.4.3.3.3.a - "...if the difference between steam-line temperature and recirculation pump inlet temperature is less than a preset value (10.7°F)."

4.4.3.3.3.b - "...feedwater flow falls below a preset level (28 percent of rated) and the flow control valves are below a preset position (19 percent open)."

*Draft revised pages are attached.

<u>Region II</u> This region shows the area where the 25% pump speed and 100% pump speed operating regimes overlap. The switching sequence from the low frequency m-g set to 100% speed will be done in this region.

<u>Region III</u> This is the low power area of the operating map where cavitation can be expected in the recirculation pumps, jet pumps, or flow control valves. Operation within this region is precluded by system interlocks which trip the main motor from the 100% speed power source to the 25% speed power source.

<u>Region IV</u> This represents the normal operating zone of the map where power changes can be made by either control rod movement or by core flow changes through use of the flow control valve.

4.4.3.3.3 Design Features for Power-Flow Control

The following limits and design features are employed to maintain power-flow conditions to the required values shown in Figure 4.4-5.

a. <u>Minimum power limits at intermediate and high core flows.</u> To prevent cavitation in the recirculation pump, jet pumps and flow control valves, the recirculation system is provided with an interlock to trip off the 60Hz power source and close 15Hz power source if the difference between steam line temperature and recirculation pump inlet temperature is less than a preset value (~~typically 10°F~~). 10.7°F This differential temperature is measured using high accuracy RTDs with a sensing error of less than 0.2°F at the two standard deviation (2σ) confidence level. This action is initiated electronically through a 15-second time delay. The interlock is active while in both the automatic and manual operation modes.

b. <u>Minimum power limit at low core flow.</u> During low power, low loop flow operations, the temperature differential interlock may not provide sufficient cavitation protection to the flow control valves. Therefore, the system is provided with an interlock to trip off the 60Hz power source and close the 15Hz power source if the feedwater flow falls below a preset level (~~typically 30% of rated~~) and the flow control valves are below a preset position (~~typically 20%~~ open). The feedwater flow rate and recirculation flow control valve position are measured by existing process control instruments. The speed change action is electronically initiated. This interlock is active during both automatic and manual modes of operation.

28% of rated

19%

c. Pump bearing limit. For pumps as large as the recirculation pumps, practical limits of pump bearing design require that minimum pump flow be limited to 20% of rated. To assure this minimum flow, the system is designed so that the minimum flow control valve position will allow this rate of flow.

d. Valve position. To prevent structural or cavitation damage to the recirculation pump due to pump suction flow starvation, the system is provided with an interlock to prevent starting the pumps, or to trip the pumps if the suction or discharge block valves are at less than 90% open position. This circuit is activated by a position limit switch and is active before the pump is started, during manual operation mode and during automatic operation mode.

4.4.3.3.3.1 Flow Control

The principal modes of normal operation with valve flow control low frequency motor generator (LFMG) set are summarized as follows. The recirculation pumps are started on the 100% speed power source in order to unseat the pump bearings. Suction and discharge block valves are full open and the flow control valve is in the minimum position. When the pump is at full speed, the main power source is tripped and the pump allowed to coast down to 25% speed where the LFMG set will power the pump and motor. The flow control valve is then opened to the maximum position at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power-flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-5.

Q. 031.08C d)
(4.4.3)
(7.7-7c)

Clarify the discrepancy between the 25 percent pump speed interlock value described in Section 4.4.3.3.3.1 and the 20 percent value which is given in Figure 7.7-7c of the FSAR.

Response:

Due to a complete Chapter 7.0 rewrite in Amendment 10, Figures 7.7-7a through h have been moved to Appendix H.A.

There is no discrepancy; Section 4.4.3.3.3.1 describes the nominal speed at which the pump will be driven during steady-state conditions when powered from the LFMG set. Figure H.A-1c of the FSAR specifies speed ranges between which the functions described are permitted to occur during transient conditions existing when the pump is started, stopped, or transferred from one steady-state speed to another steady-state speed.

Typically, Section 4.4.3.3.3.1 describes the pump as being started on 100% power source, tripped when the pump approaches full speed, and allowed to coast down where the LFMG set then continues to drive the pump at 25% speed. Figure H.A-1c describes the same function but is more definitive in describing that the LFMG cannot start powering the pump until the pump speed is between 20 and 26% of rated. Since the LFMG output frequency is 25% of the pump motor rated frequency, the final steady-state speed will be 25% of rated.

Q. 031.080 e)

Describe the primary and secondary modes of operation of the isolation valves which are referenced in Section 6.2.4.2 of the FSAR.

Response:

The primary and secondary modes of operation for all containment isolation valves have been incorporated in Table 6.2-16 (see response to Question 022.044). The primary mode of operation is indicated under column headed "Isolation Signal" and the secondary mode under "Back Up".

Q. 031.080 f)

Clarify the discrepancy between the description of the solenoid valves in Section 6.2.4.2 of the FSAR and the design which is presented in your response to Item 031.001(h).

Response:

See 7.3.1.1.2.4, 6.2.4.2, and response to Question 031.001(h). Due to a complete Chapter 7.0 rewrite in Amendment 10, Section 7.3.1.1.2.4 has been changed to 7.3.1.1.2.B.*

*Draft FSAR page changes attached.

The main steam line isolation valves are spring loaded, pneumatic, piston operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the solenoid-operated pilot valves. Each valve has two independent pilot valves supplied from independent power sources. Each main steam line isolation valve has an air accumulator to assist in its closure upon loss of air supply, loss of electrical power to the pilot valves, and/or failure of the loaded spring. The separate and independent action of either air pressure or spring force is capable of closing an isolation valve.

Both solenoids for each isolation valve must be de-energized for the isolation valve to close automatically

It should be noted that all motor operated isolation valves remain in their last position upon failure of valve power. On the other hand, all air operated valves, (not applicable to air-testable check valves), close on loss of air except the butterfly valves on the reactor building-to-wetwell vacuum breaker lines.

The design of the isolation valve system includes consideration to the possible adverse effects of sudden isolation valve closure when the plant systems are functioning under normal operation.

6.2.4.3 Design Evaluation

6.2.4.3.1 Introduction

The main objective of the containment isolation system is to provide protection by preventing releases to the environment of radioactive materials. This is accomplished by complete isolation of system lines penetrating the primary containment. Redundancy is provided in all design aspects to satisfy the requirement that any active failure of a single valve or component does not prevent containment isolation.

Mechanical components are redundant, such as isolation valve arrangements, to provide back-up in the event of accident conditions. Isolation valve arrangements satisfy all requirements specified in General Design Criteria 54, 55, 56, and 57, and Regulatory Guide 1.11, 1971, Rev. 0.

The arrangements with appropriate instrumentation are described in Table 6.2-16 and Figure 6.2-31. The isolation valves have redundancy in the mode initiation with, generally, the primary mode being automatic and the secondary mode being remote manual. A program of testing, described in 6.2.4.4, is maintained to ensure valve operability and leaktightness.

Q. 031.080 g)

Clarify the discrepancy between the isolation valve arrangement which is described in Section 6.2.4.3.2.1.2.1 and that which is shown in Figures 5.4-9a and 7.4-1a of the FSAR.

Response:

The discrepancy is clarified in the revised text for Section 6.2.4.3.2.1.2.1.*

*See draft FSAR page change attached.

the design of the system. Consequently, a third valve is required to provide long-term leakage control. Should a break occur in the reactor water cleanup return line, the check valves would prevent significant loss of inventory and offer immediate isolation, while the outermost isolation valve would provide long-term leakage control.

6.2.4.3.2.1.1.8 Recirculation Pump Seal Water Supply Line

The recirculation pump seal water line extends from the recirculation pump through the drywell and connects to the CRD supply line outside the primary containment. The seal water line forms a part of the reactor coolant pressure boundary. The recirculation pump seal water line is 3/4 in. Class B from the recirculation pump through the outboard motor operated isolation valve. From this valve to the CRD connection the line is Class D. Should this line be postulated to fail, the flow rate through the broken line has been calculated to be substantially less than that permitted for a broken instrument line.

6.2.4.3.2.1.2 Effluent Lines

Effluent lines which form part of the reactor coolant pressure boundary and penetrate containment are equipped with at least two isolation valves; one inside the drywell and the other outside, located as close to the containment as practicable.

Table 6.2-16 also contains those effluent lines that comprise the reactor coolant pressure boundary and which penetrate the containment.

6.2.4.3.2.1.2.1 Main Steam, Main Steam Drain Lines and RHR Shutdown Cooling Lines

The main steam lines extend from the reactor pressure vessel to the main turbine and condenser system, and penetrate the primary containment. The main steam drain lines also penetrate the containment and the MSIV-LCS taps off these drain lines. The RHR steam supply line and RCIC turbine steam line connect to the main steam line inside the drywell and penetrate the primary containment. For these lines, isolation is provided by automatically actuated block valves, one inside the containment and one just outside the containment. The RHR shutdown cooling effluent line is provided with automatically actuated block valves.

common to both the RHR steam supply line and the RCIC turbine steam line

for each

6.2.4.3.2.1.2.2 Recirculation System Sample Lines

A sample line from the recirculation system penetrates the drywell. The sample line is 3/4 in. diameter and designed to

Q. 030.080 h)

Provide a cross-reference between GE diagram numbers (EXX-XXXX) which are used in the FSAR diagrams and are included in the list of references on these diagrams, and the WNP-2 figures.

Response:

See revised Section 1.8.*

*Draft FSAR page changes attached.

1.8 CROSS REFERENCE FOR PIPING AND INSTRUMENTATION DRAWINGS

Tables 1.8-1 ~~is~~ and 1.8-2 provide a cross reference between FSAR figure numbers and Burns and Roe, Inc. and GE drawing numbers for piping and instrumentation drawings.

TABLE 1.8-1

Burns and Roe

CROSS REFERENCE - PIPING AND INSTRUMENTATION DRAWINGS

BURNS & ROE DRAWING NO.	FSAR FIGURE NUMBER	FSAR DRAWING TITLE
M159	1.2-2	General Arrangement - Equipment List
M502	10.3-1 3.2-23	Main Steam Supply System
M503	10.4-6	Extraction Steam & Heater Vents-TG Bldg.
M504	10.4-5	Condensate & Feedwater System-TG Bldg.
M505	10.4-7	Heater Drain System
M506	10.3-7 3.2-24	Misc. Drains, Vents and Sealing Systems
M507	10.4-3	Circulating Water System
M508	9.2-1	Plant Service Water System
M510	9.3-1	Control & Service Air System
M511	10.4-1	Off-Gas and Air Removal System
M512	3.2-22 9.5-4	Diesel Oil and Misc. Systems
M515	9.5-1	Fire Protection System
M516	9.2-3	Plant Makeup Water Treatment System
M517	9.2-4	Demineralized Water System
M518	9.3-9	Non-Radioactive Floor Drains

TABLE 1.8-1 Page 2 of 4

Burns and Roe CROSS REFERENCE
PIPING AND INSTRUMENTATION DRAWINGS

BURNS & ROE DRAWING NO.	FSAR FIGURE NUMBER	FSAR DRAWING TITLE
M519	3.2-8	Reactor Core Isolation Cooling System
M520	3.2-7	HPCS and LPCS Systems
M521	3.2-6	Residual Heat Removal System
M522	3.2-5 9.3-13	Standby Liquid Control System
M523	3.2-11	Reactor Water Clean-up System
M524	3.2-13,9.2-10	Standby Service Water System
M525	3.2-14 9.2-2	Reactor Building Closed Cooling Water System
M526	3.2-12 9.1-4	Fuel Pool Cooling and Clean-up System
M527	9.2-9	Condensate Supply System
M528	3.2-4	Control Rod Drive Hydraulic System
M529	10.3-6	Main Steam Supply System Piping
	3.2-2	Nuclear Boiler System
M530	3.2-3	Reactor Recirculation System
M531	11.2-3	Floor Drain Subsystem

Burns and Roe

TABLE 1.8-1 Page 3 of 4

CROSS REFERENCE
PIPING AND INSTRUMENTATION DRAWINGS

BURNS & ROE DRAWING NO.	FSAR FIGURE NUMBER	FSAR DRAWING TITLE
M532	11.2-2	Equipment Drain Subsystem
M533	11.2-4	Chemical Waste Subsystem
M534	10.4-4	Condensate Demineralization
M536	11.4-1	Radioactive Waste Disposal, Solids Handling System
M537	3.2-9 9.3-5	Equipment Drain System Reactor Building
M538	9.3-6	Radioactive Equip. & Floor Drains-TG Bldg.
M539	3.2-10 9.3-8	Floor Drains - Reactor Bldg.
M540	9.3-7	Radioactive Equip. & Floor Drains, Radwaste Bldg.
P541	9.2-5	Potable Water System
P542	9.2-6	Plant Sanitary Drain System
M543	3.2-15 9.4-8	Primary Containment Cooling System
M544	3.2-16	Standby Gas Treatment System
M545	3.2-18	HVAC Reactor Bldg.

Burns and Roe

TABLE 1.8-1

CROSS REFERENCE
PIPING AND INSTRUMENTATION DRAWINGS

BURNS & ROE DRAWING NO.	FSAR FIGURE NUMBER	FSAR DRAWING TITLE
M546	9.4-6	Heating & Ventilating-TG Bldg.
M547	9.4-11	HVAC-Service Bldg.
M548	3.2-19	HVAC Control Room & Critical Switchgear
M549	9.4-3	HVAC-Radwaste Bldg.
M550	9.4-4	HVAC-Chilled Water System
M551	3.2-20 9.4-7	HVAC-CW,SW & MUW Pump Houses and DG Bldgs.
M552	9.4-9	HVAC-LABS & Office Area
M553	9.4-10	HW Heating and Chilled Water System
M554	3.2-17	Primary Containment Atmospheric Control System
M555	9.4-5	HVAC-Off-Gas Charcoal Absorber Vault
M556	3.2-21 9.3-2	Containment Instrument Air System
M557	3.2-25	Main Steam Isolation Valve Leakage Control System

Page . of 2

TABLE 1.8-12
CROSS REFERENCE - PIPING AND INSTRUMENTATION DRAWINGS

GE Drawing No.	FSAR Figure No.	FSAR Figure Title
761E712	3.2-1	Group Classification Diagram
761E952	4.6-5a	Control Rod Drive Hydraulic System (P&ID) (Sh. 1)
761E952	4.6-5b	Control Rod Drive Hydraulic System (P&ID) (Sh. 2)
732E103	5.1-30	Nuclear Boiler System P&ID (Sh. 1, 2, 3)
761E289AD	5.4-2a,b,c	Recirculation System P&ID (Sh. 1, 2, 3)
761E151AD	5.4-9a,b	RCIC P&ID (Sh. 1, 2)
731E961AD	5.4-13a,b	RHR P&ID (Sh. 1, 2)
732E129AD	5.4-16	Reactor Water Cleanup P&ID
761E549	5.4-18	Filter/Demineralization System P&ID
731E931AD	6.3-1	HPCS P&ID
921E868AD	6.3-5	LPCS P&ID
731E931AD	7.3-5	HPCS P&ID
921E868AD	7.3-12	LPCS P&ID
731E961AD	7.3-14	RHR P&ID
732E103	7.3-15	Nuclear Boiler System P&ID
732E151AD	7.4-1	RCIC's P&ID
732E187	7.4-3	Standby Liquid Control System P&ID
761E289AD	7.6-19	Reactor Recirculation System - P&ID
761E952	7.7-2	Control Rod Drive Hydraulic System P&ID
732E129AD	7.7-13	Reactor Water Cleanup System - P&ID
761E549	7.7-15	Filter/Demineralizer P&ID
791E908AD	11.3-2	Off-Gas System P&ID

~~ECC:gmm/102R1~~

Cross Reference - GE Piping and Instrumentation Drawings

GE Drawing No.	FSAR Figure No.	FSAR Figure Title
731E931AD	6.3-1	HPCS P&ID
731E931AD	7.3-5	HPCS P&ID
731E961AD	5.4-13a,b	RHR P&ID (Sh. 1, 2)
731E961AD	7.3-14	RHR P&ID
732E103	5.1-30	Nuclear Boiler System P&ID (Sh. 1, 2, 3)
732E103	7.3-15	Nuclear Boiler System P&ID
732E129AD	5.4-16	Reactor Water Cleanup P&ID
732E129AD	7.7-13	Reactor Water Cleanup System - P&ID
732E151AD	7.4-1	RCIC's P&ID
732E187	7.4-3	Standby Liquid Control System P&ID
761E151AD	5.4-9a,b	RCIC P&ID (Sh. 1, 2)
761E289AD	5.4-2a,b,c	Recirculation System P&ID (Sh. 1, 2, 3)
761E289AD	7.6-19	Reactor Recirculation System - P&ID
761E549	5.4-18	Filter/Demineralization System P&ID
761E549	7.7-15	Filter/Demineralizer P&ID
761E712	3.2-1	Group Classification Diagram
761E952	4.6-5a	Control Rod Drive Hydraulic System (P&ID) (Sh. 1)
761E952	4.6-5b	Control Rod Drive Hydraulic System (P&ID) (Sh. 2)
761E952	7.7-2	Control Rod Drive Hydraulic System P&ID
791E908AD	11.3-2	Off-Gas System P&ID
921E868AD	6.3-5	LPCS P&ID
921E868AD	7.3-12	LPCS P&ID

APPENDIX E

CROSS-INDEX OF FSAR FIGURES*

TO

ENGINEERING DRAWING NUMBERS

* Burns and Roe Figures only.
For GE Figs., see Table 1.8-2.

Q. 031.080 i)
(7.2.1)

Clarify the reference to Figure 1.2-5 of the FSAR which is contained in Section 7.2.1.1.4.2.e by specifying the exact physical location and arrangement of the turbine generator oil line pressure switches and their sensing lines.

Response:

The revised FSAR Chapter 7.0 no longer makes reference to FSAR Figure 1.2-5. Section 7.2.1.1.4.2.e has been replaced by 7.2.1.1(2)e. However, the turbine governor valve hydraulic line pressure switches C72-N005 A/B and C72-N005 C/D are mounted on local instrument racks IR-10 and IR-11, respectively. The location of these instrument racks is shown on General Plant Arrangement Drawing, FSAR Figure 1.2-5.

A routing drawing of the pressure switch sensing lines is not available at this time as they are field routed and have yet to be installed.

Q. 031.080 j)

Indicate in Section 7.2.1.1.4.4.5 of the FSAR the delay time before the reactor mode switch scram is automatically bypassed.

Response:

Due to a complete Chapter 7.0 rewrite in Amendment 10, FSAR Section 7.2.1.1.4.4.5 has been changed to 7.2.1.1(8). This section has been revised to indicate a 10-second time delay.*

~~*See draft FSAR page change attached.~~

Q. 031.080 k)

Clarify the discrepancy in the instrumentation range between Note 4 of Table 7.1-2 and line 3 of Table 7.2-1 of the FSAR (i.e., reactor vessel lower water line).

Response:

There is no discrepancy. Note 4 of the FSAR Table 7.1-2 states that the active range for the reactor vessel water level is -150/0/+50 with zero at the top of the active fuel which includes wide and narrow range instruments. This is not defining any specific water level instrumentation, but rather the overall range of the water level instrumentation for the plant. Line 3 of FSAR Table 7.2-1 specifies a range of 0-60" for the low water level 3 instrument, a narrow range instrument, which is a portion of the reactor vessel water level range used for the RPS. Note 2 of the table states that "The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored." Therefore, the upper limit of 60" was selected as being above the maximum 50" active water level. Measurement below the top of the active fuel is of no interest or value for a RPS trip function.

Amendment 10, revising Chapter 7.0, moved the entry to which Note 4 applied to Table 7.7-2 which references Zimmer 1 as identical in design. Note 4 has since been deleted.

Q. 031.080 l)

Clarify the discrepancy between the response to Item 031.001 s) and the content of Drawing 807E180TC, Sheets 1 through 9.

Response:

Drawing 807E180TC, Revision 12 (2/28/78), has deleted PRT and REVAB.

Q. 031.080 m)

Clarify the reference in Section 7.3.1.2.7 of the FSAR to Sections 8.2.1 and 8.3.1. This clarification should clearly state the range of voltage and frequency for which all Class 1E instrumentation and control equipment is qualified and the range of voltage and frequency to which it will be exposed in the WNP-2 facility.

Response:

Section 7.3.1.2.7 has been replaced in the Chapter 7.0 rewrite by 7.3.1.2.F which now reads as follows:

> "Refer to Tables 3.11-1 through 3.11-5 and paragraph 3.1.2.1.4.1 for environmental conditions. Refer to Sections 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to ESF instrumentation and controls; all ESF instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes."

Q. 031.080 n)

Clarify in Section 7.6.1.4.2 of the FSAR the divisional assignments which are made for the motor-generator sets of the reactor protection system. Specifically, justify the designation of these buses as "critical".

Response:

Due to a complete Chapter 7.0 rewrite in Amendment 10, FSAR Section 7.6.1.4.2 has been changed to 7.6.1.~~*~~ Reference to the RPS buses as "critical" has been eliminated.

~~*See draft FSAR page change attached.~~

Q. 031.080 o)

Clarify the discrepancy between Figure 5.2-6 and Tables 7.2-1, 7.3-2, 7.3-3, 7.3-4 and 7.3-5 of the FSAR with regard to the low level set point and range.

Response:

The only actual discrepancy existed in Table 7.3-2. This table did not list the confirmatory reactor vessel low water level (Level 3) used in the ADS initiation logic. Amendment 10, revising Chapter 7.0, revised the tables to clarify water level trips and renumbered Tables 7.3-2, 7.3-3, 7.3-4 and 7.3-5 to 7.3-3, 7.3-5, 7.3-7 and 7.3-9 respectively.

Q. 031.080 p)

Clarify the discrepancies in the pressure trip setting between the Amendment 1 revision of Table 6.3-2 and other submittals of information in the FSAR such as Table 7.3-3 for the spray valve differential pressure.

Response:

Due to a complete Chapter 7.0 rewrite in Amendment 10, FSAR Table 7.7-3 has been changed to Table 7.7-5. As stated in Notes (3), (5), and (6), the instrument setpoints are subject to change to agree with Chapter 16.0 Technical Specifications, which have not been submitted and are under development. The actual trip settings will be established when the Technical Specifications are submitted, after which Tables 7.3-1 (HPCS), 7.3-3 (ADS), 7.3-5 (LPCS), 7.3-7 (LPCI), 7.3-9 (PCRVCS), 7.2-1 (RPS) and 7.4-1 (RCIC) will be revised. The trip setpoints will take into account accuracy, calibration and drift allowances so that the required actuation will fall within the analytic or design basis limits.

Q. 031.080 q)

Clarify the discrepancy between your statement in Section 7.6.1.8.1.2 of the FSAR regarding the uniqueness of the RPT and the statement in Item 37 of Table 7.1-2 which claims your RPT is identical to that of Zimmer.

Response:

The statement in 7.6.1.8.1.2 has been removed from the revision to Chapter 7.0. Table 7.1-2 is correct as is.

Q. 031.080 r)

Clarify the reference to four RPS divisions in Section 7.6.1.8.3.2 of the FSAR. It is our understanding that there are only two RPS divisions.

Response:

The section discussing Recirculation Pump Trip (RPT) system instrumentation and controls has been rewritten to refer to Appendix H and Section 5.4.~~*~~ There are two RPS divisions.

~~*Draft FSAR page change attached.~~

Q. 031.080 s)

Clarify the discrepancy between the content of Sections 3.11 and 7.6.2.8.2.1.1.4 of the FSAR.

Response:

Due to a complete rewrite of Chapter 7.0 in Amendment 10, FSAR Section 7.6.2.8.2.1.1.4 has been changed to 7.6.2.3.A.4 and reads as follows:

> "Equipment Qualification (IEEE 279-1971, Paragraph 4.4)
>
> Vendor certification requires that the sensor associated with each of the systems required for safety trip variable, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.
>
> Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed at each project site during the preoperational test phase.
>
> For a complete discussion of equipment qualification for the safety-related systems described in Section 7.6, refer to Sections 3.5, 3.6, 3.10 and 3.11."

~~No discrepancy exists.~~

Q. 031.080 t)

Clarify the discrepancy between Sections 3.10 and 7.6.2.8.2.1.5 of the FSAR. (Note that the RPT system is not listed in Table 3.10-1.)

Response:

WNP-2 is presently reviewing qualification requirements of Class 1E equipment. A composite list of Class 1E equipment, including RPT initiation sensors, logic, breakers, etc., will be entered in Tables 3.10-1 and 3.10-4 or equivalent tables generated. The revision to Chapter 7.0 is removing the cross-reference between 7.6 and 3.10.

Q. 031.080 u)

Clarify the references in Section 7.6.1.7.8 of the FSAR to Table 3.11-4 for the reactor and control building environments.

Response:

The Rod Sequence Control System referred to in 7.6.1.7.8 is a non-safety-related system. Environmentally-related statements have been removed from the text and the text has been moved to Section 7.7.

Q. 031.081
(3.10)
(3.11)

Identify each type of relay in the WNP-2 facility which must be energized or which must remain energized, during a seismic event. For each of these relay types, provide the following information: (1) the minimum voltage at which it must operate; (2) the voltage at which it was seismically qualified; (3) the normal operating voltage; and (4) the locations and functions of this type of relay. Where a particular relay was not qualified by test or was not tested in both the energized and de-energized state, justify the seismic qualification of the relay.

Response:

WPPSS has established a safety-related equipment re-evaluation program. This program will re-evaluate the equipment's original qualification. Its intent is to address the elements of qualification identified in new seismic and new environmental requirements. This program of re-evaluation is scheduled for completion during December 1980, at which time the requested information will be provided to your SQRT personnel. Table 031.081-1 provides a list of the relays identified to date.

MANUFACTURER	MODEL No. OR TYPE	NOMINAL VOLTAGE RATING	OPERATING MINIMUM VOLTAGE ~~RATING~~	VOLTAGE AT WHICH SEISMIC QUALIFIED	PICK-UP VOLTAGE	DROP-OUT VOLTAGE	LOCATION	FUNCTION
Asea	RXMK1 ~~RK225-052CP~~	120 VAC	LATER ~~96V~~	LATER	96V	36V	Reactor Building	Containment Isolation Valve
Struthers-Dunn	219 BBXP	120 VAC			80% DC 85% AC	None Specified	~~Radwaste Building~~ Control Room	Containment Atmosphere Control Isolation Valve Containment Supply Purge Indication
	~~GPI~~	~~125 VDC~~			~~100V DC~~	~~12-50VDC~~	~~LATER~~	~~LATER~~
Agastat	GPI	120 VAC			92V @ 20°C	20-40% of Rated	Reactor Building	Containment Atmosphere Isolation Interlock
	7012PH	125 VDC	~~100V~~		100 VDC	10% of Rated	control Room ~~Radwaste Building~~	RHR HEAT EXCH. BYPASS INTERLOCK
Denison	WE-74/EX-2	120 VAC			N/A	N/A	Reactor Building	Containment Supply Purge Indication
Sylvania-Clark	7305-PM-~~544~~ 5U4	120 V	~~85V~~		85V	50V	Diesel Building	DMA & DEA Interlock with Div. 1 or Div. 2 Diesel Generator
GE	HEAG1	120 VDC	~~75V~~		N/A	N/A	Diesel Building	HPCS Diesel Generator Lockout Relay ~~Energizes Exciter of HPCS DG~~
	HMA	120 VAC 125 VDC			90VAC 75VDC	6-12VAC 6-12VDC*	control Room ~~Radwaste Building~~	Multi-system logic interlocks ~~LATER~~
	HFA	120 VAC 125 VDC			90VAC 75VDC	88VAC 3-13VDC*	control Room ~~Radwaste Building~~	~~LATER~~ Ditto
	12CFD22B2A	N/A			N/A	N/A	Diesel Building	HPCS DIESEL GENERATOR CURRENT ~~LATER~~ DIFF RELAY.
	CR2820	120 VAC 125 VDC			98VAC 100VDC	75VAC 82VDC	Control Room Radwaste-Building	Minimum Flow Valve Time Delay
	CR105	115V 60Hz			98V	75V	Control Room Radwaste-Building	~~LATER~~ Reactor Protection System Auto Scram
								*After being continuously energized, pick-up & drop-out V increased by 10-20%.
	7012AC	120 VAC			102 V	60 V	Diesel Building	DEA VENT AFTER HPCS, DIV I & II DG OFF
	7012AD	120 VAC			102 V	60 V	Diesel Building	LOW DIFF. PRESS. TRIP DHA FAN FOR CABLE COOLING SYSTEM.
	7022AI	120 VAC			102 V	60 V	Diesel Building	DMA COOLING AFTER HPCS, DIV I & II DG OFF

Table 031.080-1

Q. 031.082
(7.6.2)

Demonstrate that the safety-related equipment discussed in Section 7.6.2.3.2.1 of the FSAR satisfies the requirements of General Design Criteria 1, 2, 3, 4, 13, 14, 16, 19, 23, and 55. Provide this demonstration of compliance with the requirements of Appendix A to 10CFR Part 50 in other sections of the FSAR where it is missing.

Response:

The applicable General Design Criteria that apply to the high pressure/low pressure interlocks are discussed in revised 7.6.2.2 and 7.1.2.2.

Q. 031.083
(3.11.3)
(3.11A)
(031.006)
(031.056)
(031.059)

Neither your response to Item 031.006 nor Appendix 3.11A of the FSAR satisfy our need for additional information on equipment qualification. In order to ensure that your environmental qualification programs conform with General Design Criteria 1, 2, 4, and 23 of Appendix A and Section III and XI of Appendix B to 10 CFR Part 50, and to the national standards (e.g., IEEE Standard 323-1971) mentioned in the Acceptance Criteria contained in Section 3.11 of the Standard Review Plan, NUREG-75/087, provided an amended response to Item 031.006 for:

a. The logic equipment for the standby gas treatment system (031.006, item (d) of the second paragraph).

b. The following sensors: (1) the rod block monitor flow transmitters; (2) the main steam line tunnel temperature thermocouple; and (3) B22-N024A.

c. All items listed in Questions 031.056 and 031.059.

Response:

a. The requested information for logic equipment for the standby gas treatment system will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

b. The requested information will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

c. 031.056 - See response to Question 031.006.

See revised response to Question 031.059.
031.059 - ~~The main steam line isolation valve logic is the same for WNP-2 as that supplied for previously reviewed and accepted for licensing BWRs.~~

~~(1) ASCO Valve #832320 is a fail safe valve and closes by current de-energizing. This valve's application does not require it to function following a LOCA. The valve has been qualified for normal ambient conditions (VPF #3680-1) as follows:~~

* Revised response attached.

Q 31.059
(031.001)
(11.6-1)

Your response to Item 031.001(h) presents a new design for the logic of the main steam line isolation valves which is different from that reviewed and accepted for licensing on similar boiling water reactors. Provide the manufacturing drawings for ASCO Valve No. 832320. Additionally, provide the results of the engineering analysis and the test results which demonstrate the ASCO Valve No. 832320: (1) is qualified for the environment in the drywell following a loss-of-coolant accident; (2) is seismically qualified; (3) meets the physical separation and the required electrical independence in accordance with the staff positions contained Regulatory Guide 1.75; (4) satisfies the single failure criterion (previous designs accepted for licensing have used two separate valves in a one-out-of-two logic for a reactor trip). Note that Table 1.6-1 of the FSAR states that the GE Topical Report, APED-5750, is applicable to the WNP-2 facility and that Table 7.1-1 indicates the main steam line isolation valves are designed and supplied by GE. Accordingly, provide justification for the change to the design which was previously reviewed and approved by the staff in our evaluation of the GE Topical Report, APED-5750.

Response:

The main steam line isolation valve logic is the same for WNP-2 as that supplied for previously reviewed and accepted for licensing BWR's.

Attachment - ASCO Dwg. #HVA-166-265

(1) ASCO Valve #832320 is a fail safe valve and, closes by current deenergizing. This valves application does not require it to function following a LOCA. The valve has been qualified for normal ambient conditions (VPF #3680-1) as follows:

Cycle tested at temperature (172-198^{o}F)

100 cycles at 4 minute intervals
10 cycles at 12 hours intervals
5 cycles at 120 hours intervals

Total time at temperature 941 hours

(2) Qualification of valve for seismic. The solenoid valve was qualified for original seismic requirement when tested with complete valve (Wyle Laboratories -- Seismic Simulation Test Report #42610-1, dated 2/27/74).

The solenoid valve remained functional during all phases of the testing. [INSERT]

(3) The protection system criteria of IEEE 279-1971 are met with this design; the requirements of Regulatory Guide 1.75 were not committed for this plant.

(4) The ASCO valves in question are not used in generating a reactor trip. The ASCO valves are used in a two-out-of-two logic for each MSIV. That is, in order for each MSIV to be isolated both ASCO solenoids must deenergize. The ASCO valves themselves are not single failure proof. Single failure criterion is preserved since each main steam line contains two valves in series. If a single failure occurs in one valve scheme the second will provide isolation.

There is no deviation from the commitments made in APED-5750.

Re-evaluation of the equipments original qualification will be performed as part of the overall seismic and environmental reevaluation effort. See the response to Question ~~Q~~. 031.006.

Q. 031.084
(3.11A)

The specification requirements of Table 3.11A-1 of the FSAR are incomplete since they do not address the maximum and minimum values of all of the parameters which are cited in Section 3(7) of IEEE Standard 279-1971. Accordingly, provide the required data for all Class 1E components.

Response:

See response to Question 031.006.

Q. 031.085
(7.6.1)
(7.7.1)

Several systems (e.g., the safety/relief valve discharge line temperature monitoring system and the reactor vessel head leak detection system) are listed in both Sections 7.6 and 7.7 of the PSAR. However, Section 7.6 should describe only those systems required for safety while Section 7.7 should describe only those systems not required for safety. In conformance with the guidance contained in Section 7.7 of Regulatory Guide 1.70, Revision 2, safety-related systems should not be listed in Section 7.7 of the FSAR. Accordingly, revise your FSAR to eliminate such ambiguous design descriptions of safety-related systems and non-safety-related systems.

Response:

FSAR Sections 7.6 and 7.7 have been revised as part of the Chapter 7.0 rewrite effort. Section 7.6 now describes only "all other systems required for safety" and Section 7.7 now describes only "control systems not required for safety" or non-safety-related instrumentation and controls of systems not described anywhere else in the FSAR.

Q. 031.086
(7.4.1)
(9.3.5)

The standby liquid control system (SLCS) is designated in Section 7.4.1.2.3.1 of the FSAR as a special plant capability event system in the WNP-2 facility. To assure the availability of the SLCS, you have provided in parallel, two sets of those components required to actuate this system. However, our review indicates that you have not provided redundant heating systems. Additionally, the heating equipment supply emergency bus is neither identified nor is it redundant. We have concluded, therefore, that your statement in Section 9.3.5.3 of the FSAR that "...a single failure will not prevent system operation..." is not true. (Note that this matter has been resolved in similar facilities by providing the redundant heating systems.) Accordingly, provide a modified design of the SLCS which satisfies the single failure criterion. Alternatively, justify your present design.

Response:

See also the response to Question 031.073.

The standby liquid control system is neither designed nor required to satisfy single failure criteria. The injection controls have redundant circuits, but only to enhance availability insofar as practical. The SLCS is a backup to CRD in the event that the rods cannot be inserted during normal plant operation by operator control of the Reactor Manual Control System. Hence, the SLCS by itself is not required to meet single failure criteria. Therefore, the system heaters are not required to be redundant nor does the power supply under normal conditions need tc be powered from an emergency bus. Section 9.3.5.3 of the FSAR reads: "The SLC system is required to be operable in the event of a station power failure; therefore, the pumps, heaters, valves, and controls are powered from the standby A-C power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure will not prevent system operation."

Due to a complete Chapter 7.0 rewrite in Amendment 10, Section 7.4.1.2.3.1 has been changed to 7.4.1.2.

Q. 031.087
(3.11.2)

With regard to Section 3.11.2.3 of the FSAR, provide the following additional information and clarifications:

a. Provide a copy of the procedures for the following aging simulations: (1) thermal; (2) radiation; (3) operation; and (4) seismic.

b. Provide justification for the aging temperature which was used with respect to the maximum normal environmental conditions which are listed in Table 3.11-1 of the FSAR.

c. Indicate the thermal aging acceleration rate and provide the basis for this rate.

d. Indicate the thermal aging time used for each plant location listed in Table 3.11-1 of the FSAR which contains a valve that has been qualified in accordance with IEEE Standard 382-1972. Identify the valves which are so qualified.

e. Provide information similar to that requested in Items (b) through (d) above for radiation aging. In addition, describe how the effect of the neutron fluences was considered.

f. Provide your criteria for determining the limits of an actuator family including: (1) the definition of the limits of an actuator family; (2) the criteria which were used to assure that the sample valve operator is a valid representative of the family; and (3) a demonstration of how the criteria were applied.

g. Provide a table of the following information for all Class 1E valve actuators in the WNP-2 facility: (1) the equipment specifications in accordance with Section 3 of IEEE Standard 382-1972; (2) an identification of the family membership; and (3) an identification of the samples.

h. Indicate the number of operating cycles to which each test specimen was subjected.

i. Indicate the frequency range which was used in the seismic qualification and aging of the samples. (Note that the frequency range permitted by IEEE Standard 382-1972 does not agree with our acceptance criteria contained in Paragraph II.1.a of Section 3.10 of our Standard Review Plan, NUREG-75/087. We will require conformance with our positions in this latter document.)

j. Describe how you assure that equipment not qualified for all service conditions, will not spuriously operate during exposure to service conditions, including excessive exposure times during which this equipment is not required to function to mitigate the effects of accidents on other events.

Response:

a. The procedures for the following aging simulations are as specified in IEEE 382-1972:

(1)	Thermal:	Refer to IEEE 382-1972, Part II, Section 2, Page 10.
(2)	Radiation:	Refer to IEEE 382-1972, Part II, Section 1, Page 10.
(3)	Operation:	Refer to IEEE 382-1972, Part II, Section 3, Page 10.
(4)	Seismic:	Refer to IEEE 382-1972, Part I, Section 4, Paragraph 4.3, Page 8.

These IEEE-382 procedures provided the outline for valve actuator qualifications. Actual valve test parameters are discussed in the following sections.

b. The actual thermal aging qualification test parameters which were imposed for the NSSS safety-related actuator applications at WNP-2 were based on Part II, Section 2, Page 10 of IEEE 382-1972. This basis (i.e., 140°F for 40 years at 55% relative humidity) envelopes the normal average temperature of 3.11.1 for the worst case location of safety-related valve actuators on WNP-2. Note (6) of Table 3.11-1 states that the maximum (abnormal) temperature and humidity will occur less than 1% of the time and, therefore, this temperature is not used as the basis for the aging of valve actuators. During refueling and maintenance times (more than 1%) the temperature of the primary containment will be much less than the average temperature (135°F) which balances the maximum temperature conditions and durations. Therefore, the aging program basis of IEEE 382-1972 envelopes the WNP-2 requirements.

c. The NSSS actuators are qualified for thermal aging in accordance with IEEE 382-1972, with the aging acceleration rate justified by the application of the 10°C rule (standard industry practice).

d. All plant locations listed in FSAR Table 3.11-1 have the same aging time to qualify NSSS valves. The aging time is derived by using the 10°C rule and the accelerated aging temperature. This aging time/temperature is based on the Part II, Section 2 of IEEE 382-1972 which envelopes the WNP-2 requirements.

e. For radiation aging, air equivalence of neutron dose to gamma dose was determined so that the actual gamma dose used in aging is the summation of gamma dose and neutron/gamma dose equivalence.

f. As required by IEEE 382-1972, a type test demonstrated that the performance characteristics of the actuator adhered to the equipment specifications and met all functional test requirements of IEEE 382-1972. The sample valve actuator was constructed using normal manufacturing processes and was then subjected to the test program. The test program for the sample valve actuator consisted of subjecting the actuator to the following sequence of conditions to simulate the design basis service conditions of the actuator: (a) aging, (b) seismic, and (c) accident. These test conditions are detailed in subparagraph 2 below. No maintenance was performed during this type test.

The manufacturer's equipment test specifications for the qualification sample are presented below and encompass the most severe conditions of equipment service.

(1) The valve operator was required to operate and remain operable during plant normal, test, design basis event, and post-design basis event conditions.

(2) The valve operator was required to provide rated mechanical force for the following conditions:

a. range of voltage - 230 to 460 volts;
b. range of frequency - 1 to 35 hertz @ 1.0g, (including seismic forces) 4 to 34 hertz @ 3.0g, 35 hertz @ 5.0g;
c. thermal conditions - see figure;
d. mechanical aging - 500 cycles, open and close; and
e. radiation exposure.

(3) The mounting configuration for the valve and operator was specified as mounted in a nominally horizontal run of pipe with the valve stem nominally vertical.

(4) Lubricants and seals have a minimum design life of 5 years.

(5) The design life of the valve operator is 40 years.

(6) Control and indicating devices contained on the valve operator include a torque switch and a limit switch.

All electrical valve actuators used in the [illegible] NSSS design are in one family. The designation for this family has been established by the vendor as the "SMB" family. The sample valve actuator which was used to qualify this family was designated as follows:

Manufacturer:	Limitorque
Type:	SMB
Size:	0
Order No.:	360943A
Serial No.:	144068

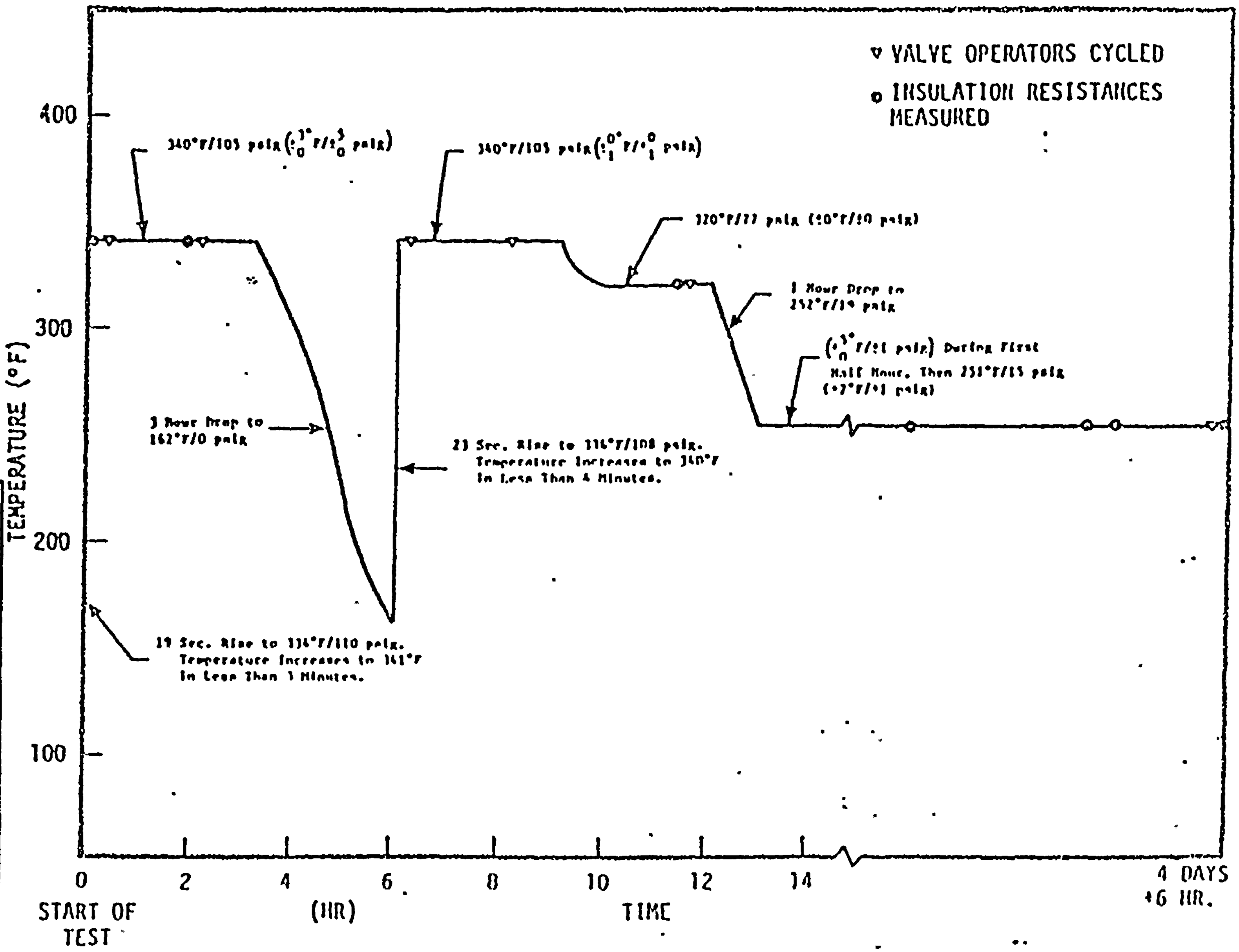
g. Same as f above.

h. The test actuator was subjected to 500 cycles as indicated in IEEE 382-1972.

i. A search for resonance was performed by scanning the test specimen in the three major axes. Scanning was done in the range from 1 to 35 hertz at a maximum acceleration of 1g. This testing identified no resonance. Next, the test specimen was vibrated at even-integer frequencies from 4 to 34 hertz for a period of 10 seconds at an excitation of 3g in each of the three major axes. The test specimen was actuated at each dwell for one complete cycle (open and close). The test specimen was then vibrated at 35 hertz for 10 seconds in each of the three major axes at an excitation of 5g and was actuated for one complete cycle.

Specific quantification of actuator qualification is embodied in the qualification test reports which are available for review at GE-NED (San Jose).

j. Equipment qualification is conducted on the safety-related actuators to assure that equipment will not operate spuriously. Safety-related NSSS valve actuators are temperature qualified to IEEE 382-1972 by test for the equivalent active 40-year plant life plus LOCA conditions. Also, these valve actuators are qualified for radiation on the basis of integrated radiation doses from LOCA plus 40 years life conditions.



FINAL SAFETY ANALYSIS REPORT

FIGURE ./.

ACTUAL STEAM EXPOSURE PROFILE

Q. 031.088
(6.2.4)
(7.6.1)
(7.7.1)
(9.3.5)

Clarify the discrepancies between the following sections and figures of the FSAR with regard to isolation of the reactor water cleanup system when the standby liquid control system is initiated: (1) Sections 6.2.4.3.2.1.1.7, 7.7.1.3, 7.6.1.4.3.6, and 9.3.5.2; (2) Figures 7.3-11a, 7.4-3, 7.7-14, and (3) Table 7.3-13.

Response:

Isolation of the RWCU system by SLCS initiation is not described in 6.2.4.3.2.1.1.7 and 7.6.1.4.3.6 or in Figure 7.3-11a because these sections and figure discuss provisions for assuring primary containment integrity following a RWCU system line break outside the containment. RWCU trip with SLCS initiation is provided, not to assure containment integrity, but to assure proper operation of the SLCS and is, therefore, only discussed in 9.3.5.2 and shown in Figure 7.4-3 as part of the SLCS system description.

The discussion of the RWCU system has been removed from 7.7 since RWCU is not a major plant control system.

Note that Amendment 10, revising Chapter 7.0, renumbered Figures 7.3-11a and 7.7-14 to 7.3-10a and 7.3-1, respectively. The discussion in 7.6.1.4.3.6 has been moved to 7.6.1.3.B and Table 7.3-13 has been removed from Chapter 7.0 and all information incorporated into Table 6.2-16.

Q. 031.089
(1.7)
(6.7.3)

The single failure analysis presented in Section 6.7.3.1 of the FSAR is inadequate. Accordingly, revise this section to include single failures of electrical components such as the spurious closing of relay contacts on K4. (Refer to GE Drawing 851E708TD.) Provide the electrical schematic and one-line drawings of this system for our review.

Response:

See response to Question 031.076.

Q. 031.090
(7.2.1)
(F7.2-1a)
(031.032)

In Section 7.2.1.1.2 of the FSAR, you state that the reactor protection system (RPS) is Class 1E. However, based on our review of similar facilities (e.g., Zimmer), we believe that this statement is incorrect since the WNP-2 motor-generator sets of the RPS are probably not Class 1E equipment. Accordingly, correct the discrepancy between Sections 7.2.1.1.1 and 7.2.1.1.2 of the FSAR regarding the qualification of the RPS motor-generator sets. Alternatively, demonstrate that all components of the RPS are Class 1E equipment.

Additionally, describe how the design and implementation of your RPS satisfies the requirements of Section 6.6 of IEEE Standard 379-1972, with special emphasis on the last paragraph of this section.

Response:

Section 7.2.1.1.2 has been deleted from the recently revised Chapter 7.0. The RPS motor-generator sets are now discussed in 7.2.1.1.

Class 1E protective devices (over/under voltage and frequency) have been included in the design to preclude adverse influence to the protection system from the motor-generator sets in the event a failure occurs.

Q. 031.091
(F7.2-9)

In facilities similar to WNP-2, the wiring from the RPS relay contacts 14A and 14C, via cabinet penetration Y, and 14E and 14G, via cabinet penetration Z, appears on terminal strip CC. (Refer to the GE drawing 807E-166TU.) This wiring is powered from two separate Class 1E d-c busses. Insufficient physical separation was provided between these busses on terminal strip CC, the associated cables, and in penetrations Y and Z, which also serve the plant process computer system. Our concern is that there may be insufficient physical separation in the RPS cabinets of the WNP-2 facility since it is our understanding that they are being manufactured by the same vendor. Accordingly, if this same problem exists in the RPS cabinets of the WNP-2 facility, we will require you to provide an acceptable design for the routing of Class 1E circuits inside the RPS cabinets. Alternatively, demonstrate that our concern on this matter is not applicable to the WNP-2 facility.

Response:

The separation review performed on WNP-2 designs resulted in the rework of the Reactor Protection System Cabinets. This rework included redesign of cabinet penetrations and contactor enclosures eliminating the problems described.

Q. 031.092
(F7.2-9)

In facilities similar to WNP-2, the cabinet lighting circuit which is not treated as an associated circuit, crosses cabinet penetration 187 in RPS cabinet A, and as a result, becomes associated with the containment isolation system wiring going to penetration 187. Our concern is that the physical separation provided in the RPS cabinets of the WNP-2 facility may not satisfy either the requirements of IEEE Standard 279-1971 or the WNP-2 separation criteria. Accordingly, if this problem of physical separation exists in the RPS cabinets of the WNP-2 facility, we will require you to take the following corrective actions:

a. Provide a modified design for the routing of non-Class 1E circuits in RPS cabinet A which satisfies the separation criteria.

b. Review the design of all other Class 1E cabinets for similar defects and indicate the cabinets which you reviewed.

c. Advise us of your findings and plans for the modifications necessary to satisfy the separation criteria.

d. Identify and justify all exceptions which you may take to items (a), (b) or (c) above.

e. Provide panel layout drawings and one line diagrams which show the routing and physical separation between the reactor trip sensors and: (1) the high level cut-offs for the HPCS and RCIC; and (2) the post-accident reactor vessel level indication system.

Response:

a. WNP-2 has recently completed a review of separation criteria versus actual installations. The review included the control room cabinets and panels as well as PGCC. See response to 031.100 for a description of the WNP-2 separation criteria. The RPS cabinets were reviewed and several modifications have been made. Cabinet lighting wiring has been separated from safety-related wiring. However, it should be noted that neither the WNP-2 separation criteria nor IEEE 279-1971 requires that non-Class 1E circuits be separated in any way from Class 1E circuits unless the non-Class 1E is supplied from a redundant division Class 1E power source.

b. Same as (a) above.

c. Same as (a) above.

d. Same as (a) above.

e. There is no criteria requiring physical separation between reactor trip sensors and HPCS/RCIC high level trips or reactor trip sensors and the reactor vessel level indication system, unless two or more non-compatible electrical divisions exist on a particular device. As part of the separation review, WNP-2 local instrument racks were evaluated and design changes incorporated to eliminate separation problems. However, two divisionally noncompatible systems (which are not electrically or functionally redundant), RPS and HPCS level trips, remain on a single instrument. Contact-to-contact separation has been employed with circuit wiring leaving the instrument housing via different feed-throughs and conduits to separated terminal boxes. The drawings showing these revisions are in the process of being updated.

Q. 031.093
(7.2.1)
(7.2.2)

Provide in Sections 7.2.1.1 and 7.2.2.1 of the FSAR, the design criteria and a description of the scram discharge volume switches and their qualification testing, including the following information: (1) the manufacturer; (2) the type of float (i.e., whether it is self-equalizing or sealed); (3) the float material and the magnet material; and (4) the qualification test conditions including the water temperature; the pressure; the duration of the test conditions; the number of test cycles; the period between test cycles; the extremes of external temperature, pressure, and humidity; and the radiation source, strength, and dose.

Response:

The scram discharge volume switches are Quality Class 1E magnetic switches which use non-self-equalizing floats. They are manufactured by the Magnetrol Company and are identified as Model 5.0-751. The floats are made of 347 stainless steel, while the magnet material is Alnico 5.

Test units were subjected to ten 64-hour exposure cycles, each consisting of 16 hours at 300°F dry heat, and 48 hours at 95 to 100% relative humidity between ambient and 100°F. Ten thousand cycles of on-off operation were performed. The units were exposed to 4.4 x 10^4 RAD integrated dosage to simulate 40 years of background radiation.

Any other information concerning qualification testing will be available to your SQRT or environmental review personnel when the qualification re-evaluation review has been completed. See responses to Questions 031.006 and 031.023.

Q. 031.094
(7.2.2)
(7.3.2)
(031.033)

Your response to Item 031.033 is incomplete since it does not indicate that the individual system level indicators can be actuated from the control room by the operator. Accordingly, revise Sections 7.2.2.1.2.1.5, 7.3.2.1.2.1.6.2, and 7.3.2.2.2.1.5.1.2 and all other similar sections of the FSAR, to describe the provisions you have incorporated into the design of the WNP-2 facility to satisfy Position C.4 of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems", May 1973. (Note that this position is not intended to address the testing of annunciators, but is intended to provide manual initiation of system level indication of inoperable and bypassed status.)

Response:

The WNP-2 FSAR Chapter 7.0 has been modified to include a generic discussion ~~[illegible]~~ in 7.1 of the conformance to Regulatory Guide 1.47. All other discussions in 7.2, 7.3, 7.4, 7.5 and 7.6 have been deleted.

The generic discussion describes the capability of each system level bypass indication to be manually actuated. This provides system level indication for those bypass and inoperability conditions which are not automatically indicated.

Q. 031.095
(7.3.2)

Your discussion of how the instrumentation and controls satisfy the requirements of Section 4.1 of IEEE Standard 279-1971 is inadequate. Indicate the pick-up and drop-out voltage values of the bus voltage relay.

Response:

The manufacturer, voltage characteristics, and seismic test criteria for WNP-2 relays is provided in the response to Question 031.081.

Q. 031.096
(7.3.2)

Provide justification in Section 7.3.2.1 of the FSAR for indicating a loss of power to the motor starters by de-energizing the indicating lamps. In your response, discuss how the reactor operator can distinguish between a failed lamp and a system bypass. Demonstrate how your design of these indicating lights satisfies the requirements of IEEE Standard 279-1971 with regard to providing the reactor operator with timely and unambiguous information.

Response:

The WNP-2 system level bypass indication system includes inputs from safety-related motor starter power monitoring relays. The de-energization of these relays results in annunciation of a safety system bypass condition.

The operator will then scan the main control board section involved looking for equipment with both indicating lamps extinguished. The use of these indicating lamps to identify valve power loss is only an aid to the operator in identifying the actual component causing the system bypass after receiving annunciation.

Q. 031.097
RSP
(031.026)

Your response to Question 031.026 is unacceptable. It is our position that isolation devices which are used to provide electrical independence between Class 1E and non-Class 1E equipment must: (1) be designed, qualified, and implemented in accordance with all of the requirements for Class 1E equipment; and (2) be an integral part of the system which they are intended to protect. Accordingly, we require you to revise your response to Question 031.026 and to provide all the information requested in this previous request. (Note that this matter has been resolved in similar facilities by modifying the design.)

Response:

Question 031.063 also requested an update of the response to 031.026. The updated response was submitted in Amendment 3 in March, 1979, and responds to the concerns of this question.

Q. 031.098
RSP
(7.6)
(15.4.1)

It is our position that the use of the rod worth minimizer (RWM) is unacceptable for safety-related functions since it does not satisfy the requirements of IEEE Standard 279-1971. Accordingly, we require you to delete this system from Section 7.6 of the FSAR. (Note that you claim credit for the RWM in Section 15.4.1.2.2.1 of the FSAR which implies that the RWM is a safety-related system.

Response:

The Rod Worth Minimizer (RWM) does not provide a safety function, nor is it required for power generation. The FSAR will be modified to move the RWM description from Section 7.6 to Section 7.7.

Section 15.4.1.2.2.1 does describe the function of the RWM and the Rod Sequence Control System (RSCS) as the means of preventing an approach to criticality at low power levels by blocking the withdrawal of an out of sequence rod.

Credit is taken for these systems because the RWM and RSCS are redundant for this extremely unlikely event. However, the analysis provided in NEDO 23842 shows that the failure to block rod withdrawal by the RWM and the RSCS is backed-up by the scram which will be initiated by the Neutron Monitoring System (either IRM or APRM) inputs to the Reactor Protection System. Both these systems are safety-related. The analysis shows that the licensing basis criterion for fuel failure is still satisfied even when the RWM and RSCS fail to block rod withdrawal.

Q. 031.099
(3.4)
(7.3.1)
(031.030)

The response to Item 031.030(c) is incomplete since you do not discuss the consequences to electrical equipment in the event of internal flooding. Section 7.3.1.2.8.1 of the FSAR is similarly incomplete. Accordingly, provide a revised response to Item 031.030(c) which discusses the protection of Class 1E equipment from internal flooding (e.g., a failure of either the main condenser cooling line or of the fire protection system).

Response:

The protection of Class 1E equipment from internal flooding is discussed in FSAR Sections 3.4.1.4.1.2 and 3.4.1.5.2, which were submitted in Amendment 5. The ECCS equipment in the reactor building basement, where the building sumps are located, are protected from internal flooding due to post-LOCA, ECCS passive failures by a Class 1E leak detection system is discussed in the response to FSAR Question 212.003. The passive failure is isolated before it has any additional effect on ECCS operation.

The potential flooding and environmental effects from postulated through-wall leakage cracks in moderate energy fluid piping systems, and postulated rupture of high energy fluid pipings are currently being re-evaluated as stated in FSAR Section 3.6.

The effects of the internal flooding on electrical equipment are being taken into account in the re-evaluation. The results of this analysis will be furnished by amendment to FSAR Section 3.6. At that time a change to Question 031.030(c) and this question will be provided.

Q. 031.100
(7.3.2)

In Table 7.1-2 of the FSAR, you indicate that many of your instrumentation and control systems are identical to those of LaSalle and Zimmer. During the course of our review of these facilities, which are similar to the WNP-2 facility, we encountered a number of errors in the implementation of the basic GE design. Our concern is that these same errors, or similar errors, could occur in implementing the electrical design of the WNP-2 facility. In particular, we find that your analyses in 7.3.2.1.2.3.1 and 7.3.2.2.3.1.1 of the FSAR, to determine compliance with the requirements of IEEE Std. 279-1971, are too general in content. We provide guidance for the information we need in Section 7.2 of the Standard Review Plan, especially in Appendix 7.2.A. Specific examples of areas where we require additional information are presented in Items 031.081, 031.084, 031.091, and 031.092 of this enclosure. Accordingly, provide more specific analyses of how you have implemented, in detail, the basic GE electrical design in the WNP-2 facility. References to other sections of the FSAR are acceptable in lieu of repeating this information in 7.3.2.1.2.3.1.

Response:

The GE separation criteria has been integrated into the WNP-2 separation criteria. A copy of the criteria is attached.

Cable Separation Criteria

Objective

The installation of electrical cables shall be in accordance with the following design criteria. The purposes of these criteria are as follows:

a. To preserve the independence of redundant safety related electrical systems.

b. To minimize the influence of a non-safety related cable on safety related cables.

c. To minimize the influence of various types of cables (instrumentation, power, etc.) on each other.

d. To give design and installation guidance to assure that separation and identification requirements are met.

Definitions and General Requirements - Balance of Plant and Nuclear Steam Supply Safety Related Systems

Definitions

Power Cable

Power cables are defined as those cables that provide electrical energy for equipment motive power and heating requiring 14.4 kv, 6.9 kv, 4.16 kv, 480 volts, 240 volt, 120/208 volt, a-c, 250 and 125 volts d-c. (See Page 031.100-24 for further information.)

Power cables of different voltage ratings must be routed in different cable trays except as follows: (a) Common tray is permitted for 480 volt, 120/208 volt ac, 125 volt and 250 volt dc of compatible divisions; (b) Common tray is permitted for 4160 and 6900 volt power cables of compatible divisions. 480, 4160 and 6900 volt power cables are not to be installed in cable trays in the spreading area beneath the control room. If a run through this area is unavoidable, the power cable shall be installed in conduit.

Power cables shall be installed in raceways separate from control cables and low-level signal cables and where vertically stacked, the power cables shall be placed in the tray with the highest position in the tray tier. Stacking of multiple power trays shall be such that the voltage levels decrease sequentially from the top to the bottom tray in the stack.

Control Cable

Control cables are those cables using voltages 120 volts ac (or below) or 125 volts dc (or below), with normal current not in excess of 30 amperes, whose circuits are designed to supply control power for the plant systems. Included in the category of control cables are those cables used for intermittent operation to change the operating status of a utilization device of the plant system. Control cables include all cables which have any of the following functions: (See page 031.100-24 for further information.)

a. 125 volts dc or 120 volts ac feeds to switchgear, panel and local panel control buses. Wire types are to be power cable, type G2.

b. 125 volt dc or 120 volts ac feeds to solenoids.

c. 125 volts dc or 120 volts ac control and interlock circuits.

d. Annunicator circuits.

Instrument Cable (Low-level signals)

Instrumentation cables are those cables used to carry low-level analog or digital signals. Low-level signal cables require a specific degree of separation or segregation to perserve the accuracy of the transmitted signal. Low-level signal cables are run in raceways separate from all power and control cables, except within the Control Room Power Generation and Control Complex (PGCC) and as noted below. Instrument (signal) trays shall be of the enclosed (solid bottom and covers) type.

Analog and digital signal input cables shall be routed as follows:

Digital computer signals in the reactor building shall be run in Divisional control trays as applicable by the device being serviced. Non-Class 1E digital signals in other areas shall be run in instrumentation trays of Division B, unless they are routed through the reactor building.

Analog computer signals in the reactor building shall be run in Divisional instrumentation trays as applicable by the device being served. Non-Class 1E analog signals in other areas shall be run in instrumentation trays of Division A, unless they are routed through the reactor building.

Safety Related Electrical and Instrumentation Systems and Equipment

Those electrical and instrumentation systems and equipment which are relied upon to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary. Safety related systems and equipment are limited in this document to the Reactor Protection System and the Engineered Safeguards Systems.

Reator Protection System (RPS)

The Reactor Protection System is the overall complex of instrument channels, trip system and trip actuators, and wiring which generates a reactor trip (scram) signal to initiate a reactor trip when a monitored parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. The complete RPS is a Class 1E safety related system.

The Reactor Protection System Power System, consisting of MG sets, distribution panels, etc., is a separate, non-safety related system which supplies power to the RPS itself.

Engineered Safeguards Systems (ESS)

This includes that combination of subsystems which take automatic action to provide the cooling necessary to limit or prevent the effects of fuel cladding melting, maintain the integrity of the containment, and insure that the exposure of the public to radiation will be below the limits of 10CFR100 in the event of a design basis reactor accident.

Nuclear Steam Supply Shutoff System

The instrument channels (except those common to RPS), power supplies, trip systems, manual controls, and interconnecting wiring involved in generating a NSSS system function. Instrument channels for the isolation functions which are shared with the Reactor Protection System are considered a part of the RPS as far as separation is concerned.

Instrument Channel

An arrangement of sensory and intermediate components as required to generate a single trip signal related to a particular plant parameter and introduce this trip signal into a trip system. The channel loses its identity upon combination of its trip signal with others.

Trip System

An interconnected arrangement of components making use of instrument channel outputs in the generation of a trip function when appropriate logic is satisfied.

Trip Actuator

The mechanism which carries out the final action of the protection system.

Redundant System

A system or sub-system whose function can be provided by another system or sub-system.

Standby Power Sources

Emergency "on-site" power sources designed for use when offsite power is not available. These include engine-driven generators and station batteries.

Single Failure

A single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be part of the single failure. Systems providing safety functions are considered to be designed against an assumed single failure, if a single failure of any component does not result in a loss of capability of the systems to perform their safety function.

a. Active Failure: An active component failure is defined as the malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, phnematic, or electrical malfunction, but not the loss of component structural integrity.

b. Passive Failure: A passive component failure in the sense utilized in Section 3.6.1.21a refers to the failure of:

   1) passive electrical equipment such as shorts in cables,

   2) pump or valve seals for long term cooling requirements.

Isolation Device

A device in a circuit which limits the effects of events in one section of a circuit from causing unacceptable consequences in other sections of the circuits or other circuits. Some examples of isolation devices are relays, buffer amplifiers, isolation transformersm fuses, circuit breakers and fire stops.

Raceway

Any channel that is designed and used expressly for supporting wire, cables or bus bars. Raceways consist primarily of, but are not restricted to, cable trays, wireways and conduits.

Potential Hazardous Area

This is any area in the vicinity of potential missile and external fire risk, pipe whip, and jet impingement.

General Area

This is an area from which potential hazards of missiles, external fires and pipe whip are excluded.

General Requirements

Segregation of Cables

Outside of the Main Control Room separate cable trays shall be installed for the five types of cables, i.e., high voltage power, power, control, low-level signal and RPS, with not more than one of these types of cable permitted in any tray.

Separation Details for Raceways

The degree of isolation and/or separation between raceways varies with the potential hazards within a particular area of the station. These areas are classified as follows:

a. General Areas

b. Mechanical Damage (Missile) Area

c. Fire Hazard Area

d. Cable Spreading Room

e. Control Room

Minimum separation distances are for open ventilated trays providing the following is observed:

a. Cable splices in raceways are to be prohibited.

b. Cables and raceways are to be flame retardant.

c. Design basis is that cable trays will not be filled such that cables extend above tray side rails (this approximates a 50% tray fill on a random basis).

d. Hazards to be limited to failures or faults internal to electric cables.

General Areas

a. The minimum separation distance between open cable trays of redundant divisions or between an open tray of one division and a conduit of a redundant division routed above the tray shall be three feet free air space (horizontally) and five feet free air space (vertically). However, if no automatic area fire detection and extinguishing system exists, and the lower tray is the highest tray in a tier of more than three, the minimum vertical free air space for separation shall be eight feet. The minimum separation distance between an open cable tray of one division and a conduit of a redundant division where the conduit is routed below the open tray shall be one inch. Where equipment arrangement precludes maintaining the minimum separation distance, covers or barriers are to be provided between trays of redundant divisions, as shown on pages 031.100-15 thru 031.100-18. Circuits of redundant divisions can also be run in solid enclosed raceways, such as totally enclosed trays or rigid steel conduit, where the minimum established distance for open trays is not maintained.

b. In cases of crossover of one open tray over another of a redundant division where the minimum vertical separation criteria established in a. above is not maintained, barriers consisting of solid steel covers on bottom trays and solid bottom in top trays shall be provided. These covers shall extend to each side of both tray edges by a minimum distance equal to three times the width of the widest tray involved in either division. The length of the protective covers is taken along the tray centerline. At cross-overs, a minimum vertical separation of one inch is to be provided between the top of the bottom tray and the bottom of the top tray.

c. In cases of crossovers of enclosed raceways and open trays of a redundant division, the minimum separation distance shall be one inch when the enclosed raceway is below the open tray. Otherwise, vertical separation established in a. above shall be maintained.

d. Fire stops shall be used where any raceway penetrates the slab into the control room, where any raceway penetrates designated fire areas, or where any raceway penetrates areas where an ambient pressure difference exists. In addition, fire stops shall be provided where any open vertical raceway penetrates floor or ceiling slabs. Both the penetration and the trays themselves shall be sealed with fire resistant material.

Mechanical Damage (Missile) Area

a. An ananalysis shall be performed to assure that routing, arrangement and/or protective barriers are such that no credible locally generated missile pipe whip or jet impingement can damage a sufficient amount of safety related cabling or equipment to cause loss of safe shutdown/accident mitigation capability when taken with a single active or passive failure.

b. Class I Electrical Systems Cables shall not be routed through other than Class I structures to protect against earthquake damage and exposure to tornado or flooding conditions unless analysis is performed to demonstrate that loss of such cables does not negate a safety function.

c. Installation of non-Seismic Category I equipment in areas containing Seismic Category I equipment should be avoided where practicable, or adequate barriers shall be provided to protect Category I equipment, or analysis shall be performed to demonstrate that failure of the non-Category I equipment will not lead to degradation of a plant safety function.

Fire Hazard Areas

a. Routing of cables and conduits for safety related redundant systems through an area where there is potential for accumulation of large quantities of oil or other combustible material shall be prohibited.

b. Fire stops shall be provided for all tray and conduit (at the first available junction) penetrations passing through fire rated barriers at fire hazard area boundaries (both sides).

Cable Spreading Room

a. The cable spreading room is the area under the control room where cables leaving the panels are dispersed into their various raceways for routing to all parts of the plant.

b. The minimum separation distance between open trays of redundant divisions is to be one foot between trays separated horizontally and three feet between trays separated verticially assuming a fire detection and extinguishing system is present. If these distances cannot be maintained, fire barriers shall be installed.

c. The minimum separation clearance between conduits and open trays of redundant divisions is 6 inches free air space when the conduit is below or to the side of the open tray and 3 feet free air space when the conduit is located above the open trays.

Control Room

a. In general, no single control panel should include wiring essential to more than one safety related redundant function. If cabling of redundant functions must be terminated in the same panel or if cables of redundant divisions run through the same panel, a minimum separation of 6 inches shall be maintained between cables and components to prevent common damage, unless separated by a barrier or an isolation device. A sheet metal enclosure and/or conduit around the intruding division wiring or component is an adequate barrier. The enclosure(s) shall include the cables, terminal blocks and the actual device (e.g., switch, light) if required.

b. For PGCC see G. E. NEDO 10466.

c. In the area behind the PGCC termination cabinets and near the Control Room walls, cables will be routed in grounded flexible conduit and the area provided with a silicone foam fill or halon fire suppression system, or an alternate method of providing electrical separation/fire protection shall be furnished.

Identification of Panels, Racks, Junction and Pull Boxes, Cable, Cable Trays and Conduit

a. General

Equipment associated with the RPS, NSSS and ESS shall be identified so that two facts are physically apparent to the operating and maintenance personnel: first, that the equipment is part of nuclear safeguards system; and second, the grouping (or division) of enforced segregation with which the equipment is associated.

b. Panels and Racks

Panels and racks associated with the nuclear safeguards system shall be labelled with marker plates which are conspiciously different in color or color of engraving-fill from those for other similar panels. The marker plates shall include identification of the division of the equipment included.

c. Junction or Pull Boxes

Junction and/or pull boxes enclosing wiring for the Nuclear Safeguards System shall have identification similar to and compatible with the panels and racks considered in b. above.

d. Cable

Safety related cables (Divisions 1 thru 7) shall be uniquely identified by number and color code. Each cable listed in the cable schedule shall be assigned a number for identification purposes. The number shall appear on the electrical installation drawing and on the wiring diagrams on which the terminations of the cable are shown.

Cable identification tags shall be made of a permanent material and permanently attached to all cables. Tags shall indicate the individual cable number, and the particular separation division to which the cable is assigned according to the marking characteristics shown in f. below.

Cables shall be tagged at fifteen foot intervals and at their terminations. This identification requirement does not apply to individual conductors or to cables which run in conduit only.

e. Identification of Cable Tray and Conduit

Each cable tray section shall be assigned an identification node number which is made of a plastic material and applied to the sides of the tray. Moreover, those sections that are assigned a separation code corresponding to the codes assigned to each safety system cable grouping shall have their respective color numbers marked on their sides in color.

Conduits shall be tagged in a manner similar to that used for cable identification.

All trays and conduits shall be identified at entrance and exit points of each room they pass through. Conduits shall be identified at the beginning and at the end, at all boxes, and at all discontinuities.

Tray/conduit marking characteristic code is shown in f. below.

f. Marking Characteristic Code

Division	Application	Tray/Conduit Characters	Inscription Characters	Background
1	P, C, I	Div. 1	Black	Yellow
2	P, C, I	Div. 2	Black	Orange
3	P, C, I	Div. 3	Black	Red
4	RPS-A1 NSSSS-A1 NMS-A	R Ch. A1	Red	Lt. Blue
5	RPS-A2 NSSSS-A2 NMS-C	R Ch. A2	Red	Green
6	RPS-B1 NSSS-B1 NMS-B	R Ch. B2	Red	Dk. Blue
7	RPS-B2 NSSSS-B2 NMS-D	R Ch. B2	Red	Brown
A	P, C, I	Div. A	Black	Silver or Silver/Yellow Stripe
B	P, C, I	Div. B	Black	Gold or Gold/Orange Stripe

P - Power
C - Control
I - Instrumentation

Non-Class 1E circuits receiving power from Class 1E power sources which are not shed by an accident signal shall be identified by the addition of checkered black/silver or black/gold markers indicating the Class 1E division (Division 1 or 2 respectively) from which the circuit receives its power and identified as A'1 or B'2 (respectively) in the computerized cable schedule.

Specific Requirements for Separation of Cables for Nuclear Safeguards Systems

Reactor Protection System (RPS, NSSSS and NMS)

Reactor Protection System (RPS, NSSS and NSSS, and NMS fail-safe wiring:

a Fail-safe wiring outside of the main protection system cabinets shall be run in rigid or flexible conduits and/or totally enclosed trays used for no other wiring and shall be conspiciously identified at all junction or pull boxes. IRM, LPRM input, and RPS Scram Group output cables may be combined in the same wireway provided that the four divisional separation is maintained.

b. Wires from both RPS trip system trip actuators to a single group of scram solenoids may be run in a single conduit; however, a single conduit shall not contain wires to more than one group of scram solenoids. Wiring for two solenoids on the same control rod may be run in the same conduit.

c. Cables through the primary containment penetrations shall be so grouped that failure of all cabling in a single penetration cannot prevent a scram. (This applies specifically to the neutron monitoring cables and the main steam isolation valves position switches.)

d. Power supplies to systems which de-energize to operate (so called "fail-safe" power supplies) require only that separation which is deemed prudent to give reliability (continuity of operation). Therefore, the protection system flywheel motor generator (MG) sets and load circuit breakers are not required to comply with the separation requirements of this Specification for safety reasons even though the load circuits go to separate panels.

e. Wiring for the four RPS scram group outputs and the NSM LPRM inputs must be routed as four separate divisions.

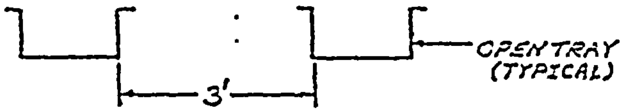
Non-Class 1E Circuits

Non-Class 1E circuits which receive power from Class 1E power sources shall be uniquely identified and comply with the same separation requirements placed on Class 1E circuits. For example, a Division A non-Class 1E circuit whose power source origin is a Division I critical bus must be separated from a Division B non-Class 1E circuit whose power source origin is a Division II critical bus.

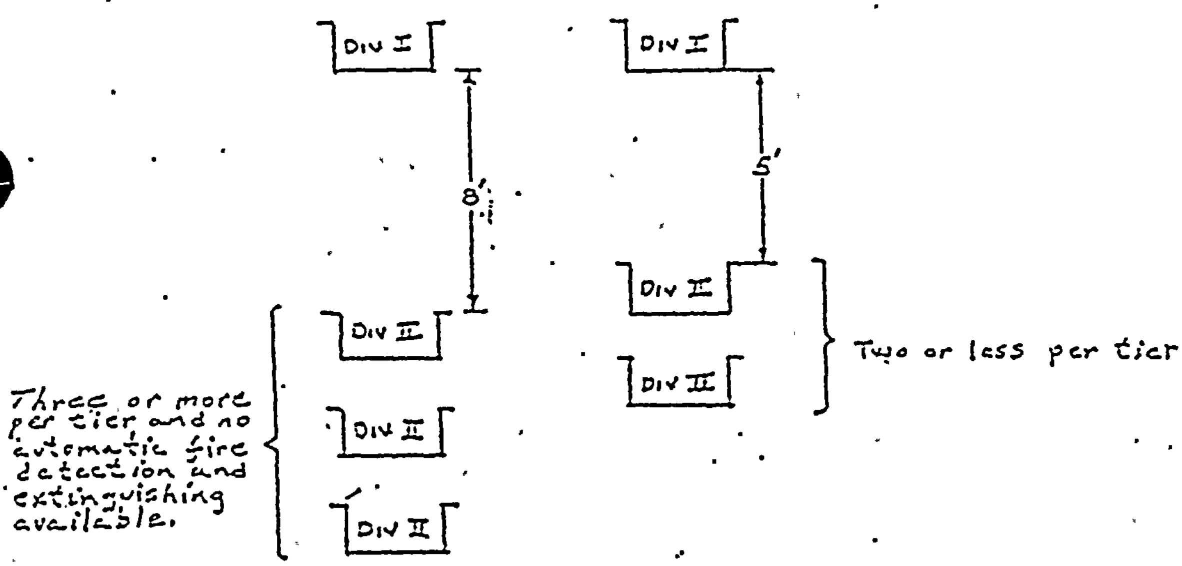
All other non-Class 1E circuits require no separation.

See Table IV for a description of acceptable non-Class 1E circuit routing.

A. Minimum horizontal separation requirements between any two redundant divisions.



B. Minimum vertical separation requirements between any two redundant divisions.



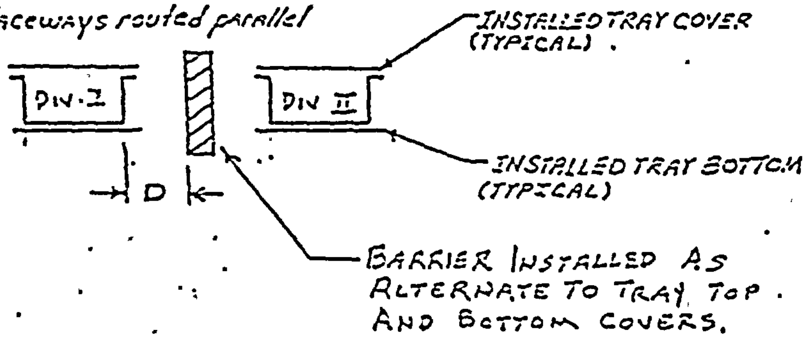
Note: Distances shown consider the ideal arrangement of two (2) raceways only. If more than two (2) trays exist in any particular arrangement, physical separation distances chosen must be based on the complete configuration.

GENERAL AREAS/OPEN TRAYS (See note, p. 031.100-15)

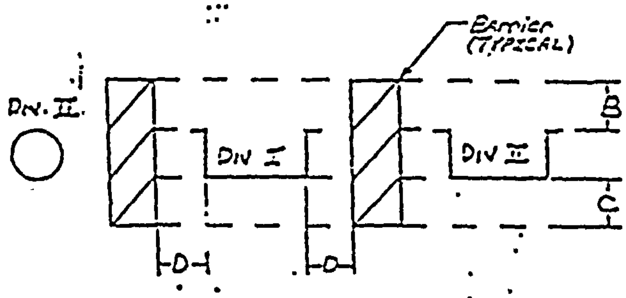
C. Where minimum separation requirements, between two raceways of redundant divisions are not met, the appropriate solution depicted in the following illustrations shall be implemented.

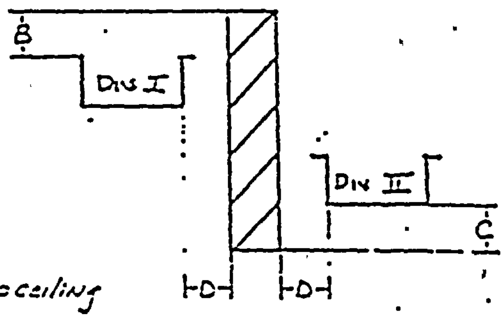
HORIZONTAL

1) Control & Instrumentation Raceways routed parallel



2) Power Raceways routed parallel



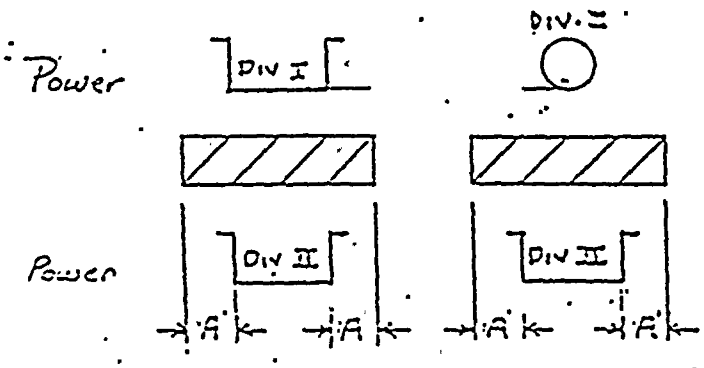


B = 12" Minimum or flush to ceiling
C = 12" Minimum or flush to floor
D = 1" Minimum

GENERAL AREAS/OPEN TRAYS
VERTICAL

See Note, P.031.100-15

Barriers & Tray Covers - Where 2 or more Power raceways, of redundant divisions are routed parallel above or below each other and where Control & Instrumentation raceways are routed above or below Power raceways of redundant divisions.

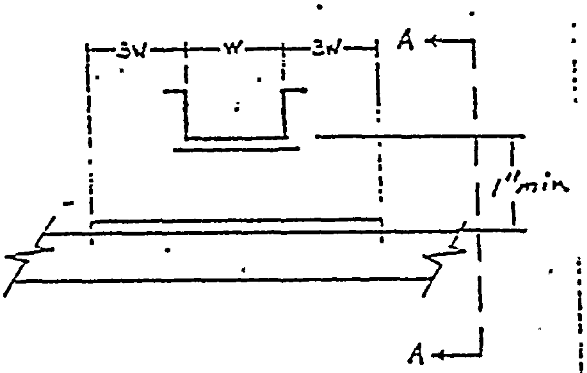


A = 12". minimum or flush to wall.

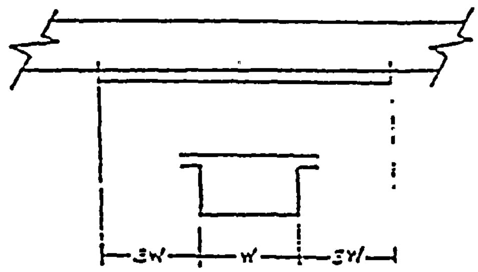
GENERAL AREAS/OPEN TRAYS

CROSSOVERS

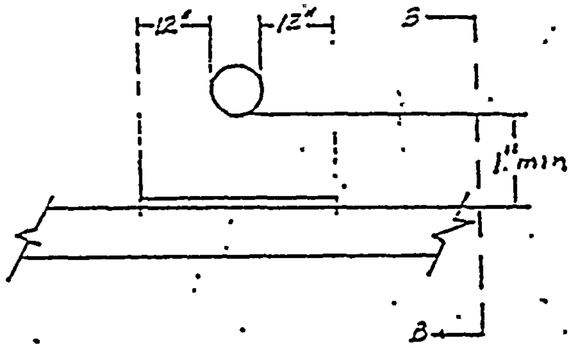
Tray Covers shall be used for all crossovers of redundant division raceway systems. The schemes shown below shall be used regardless of the voltage level of the cables in a crossover raceway system.

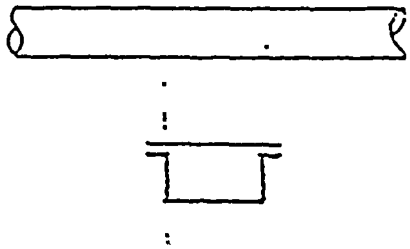


'W' is defined as the nominal tray width of the widest tray involved.
3W is 3 times the nominal tray width or flush to a wall

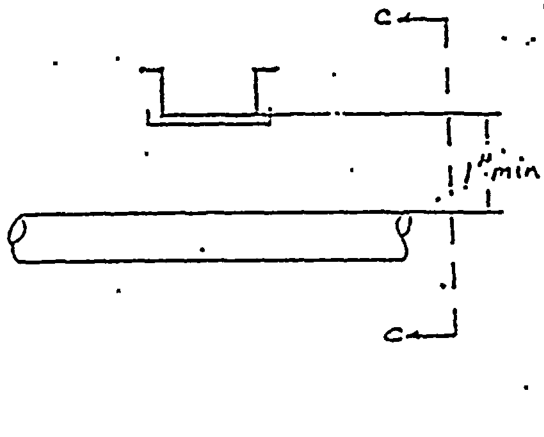


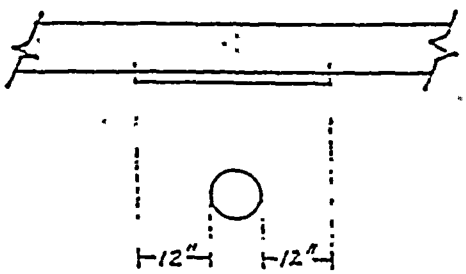
SECTION A-A





SECTION B-B





SECTION C-C

TABLE I
CABLE ROUTING CRITERIA
(Excluding of Redundant Channels)

Group (Note 1)	System or Service	Type: None	Type: Class IE	Type: Control	Type: Power	Type: H.V. Pwr.	Type: Signal	Raceway: Enclosed	Raceway: Conduit or Open Tray	Remarks
1	RPS Controls		X	X				X		
2	RPS Scram Solenoids		X		X			X		
3	RPS Neutron Monitoring		X				X	X		
4	RTD's		X				X	X		
4	TC's		X				X	X		
4	Transducers		X				X	X		
4	Supervisory Inst.		X				X	X		
4	Radiation Monitoring		X				X	X		Note 2
4	Other Low Level Non-Dig.		X				X	X		
5	RTD's	X					X	X		
5	TC's	X					X	X		
5	Transducers	X					X	X		
5	Supervisory Inst.	X					X	X		
5	Radiation Monitoring	X					X	X		Note 2
5	Other Low Level-Non Dig	X					X	X		
6	Digital		X				X	X		
6	Control Circuits.		X	X					X	Note 3
7	Digital Ckts.	X					X		X	Note 4
8	Control Ckts.	X		X					X	Note 3
8	Communication Ckts.	X		X					X	
9	125-250 VDC, 120-480 VAC		X		X				X	
10	125-250 VDC. 120-480 VAC	X			X				X	
11	4.16. 6.9 KV		X			X			X	
12	4.16 - 6.9 KV	X				X			X	

NOTES:
1. Same Group Number indicates a common raceway. Different Group number indicates separate raceway.
2. T.I.P. Cables may be combined with control cables.
3. See Par. 3.6.1.2 for control cable definition.
4. Non IE Digital Ckts. inside the reactor bldg. may be mixed with Control Ckts.

TABLE II

ASSIGNMENT OF SYSTEMS TO DIVISION OF SEPARATION

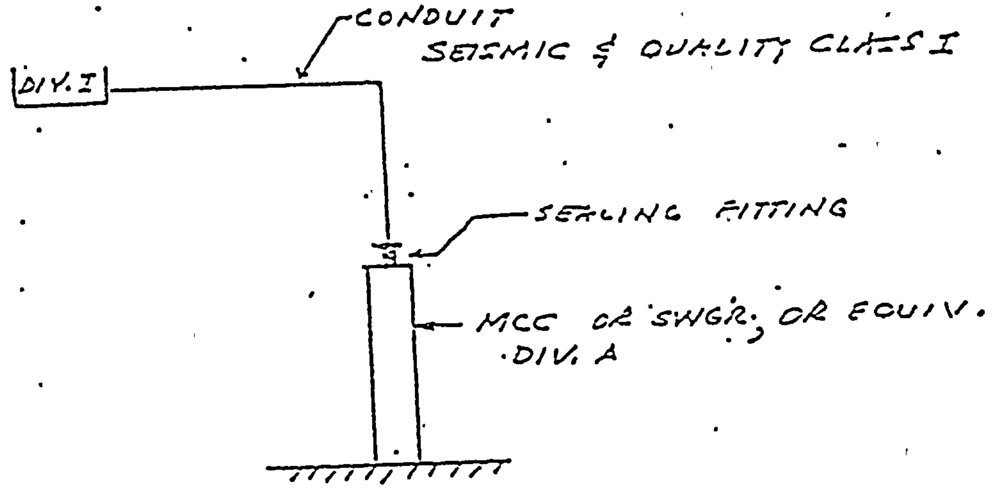
Division 1	Division 2	Division 3
RHR A	RHR B	HPCS
LPCS	RHR C	HPCS Diesel-Generator
Outboard Isolation Valves	Inboard Isolation Valves	125 VDC Battery 3
Standby Emergency Power 1	Standby Emergency Power 2	Standby Service Water C
RCIC		Safety Related Display Instr. 3
Automatic Depressurization Div. 1 Controls	Automatic Depressurization Div. 2 Controls	
Standby Gas Treatment (Loop 1)	Standby Gas Treatment (Loop 2)	
250 volt dc Battery		
125 volt dc Battery 1	125 volt dc Battery 2	
24 volt dc Battery 1	24 volt dc Battery 2	
Standby Service Water Pump A	Standby Service Water Pump B	
MSIV-LCS (Inboard)	MSIV-LCS (Outboard)	
Leak Det. System 1	Leak Det. System 2	
CAC 1	CAC 2	
Cont. Inst. Air 1	Cont. Inst. Air 2	
SLCS 1	SLCS 2	
Mn. Cont. Rm. HVAC 1	Mn. Cont. Rm. HVAC 2	
Reactor Shutdown 1	Reactor Shutdown 2	
RPT 1 Output	RPT 2 Output	
Safety Related Display Instr. 1	Safety Related Display Instr. 2	
Suppression Pool Temp. Monit. 1	Suppression Pool Temp. Monit. 2	

TABLE III

ASSIGNMENT OF RPS, NSSSS AND NMS TO DIVISIONS OF SEPARATION
(FAIL-SAFE WIRING)

Division 4	Division 5	Division 6	Division 7
(PGCC Div. 1)	(PGCC Div. 2)	(PGCC Div. 1)	(PGCC Div. 2)
RPS A1	RPS A2	RPS B1	RPS B2
NSSSS A1	NSSSS A2	NSSSS B1	NSSSS B2
NMS A	NMS C	NMS B	NMS D

CASE No. 3



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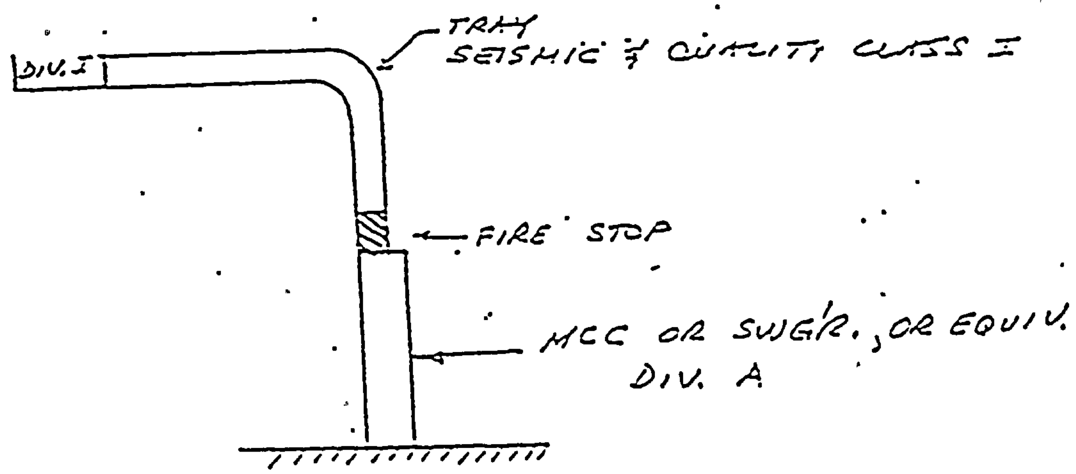
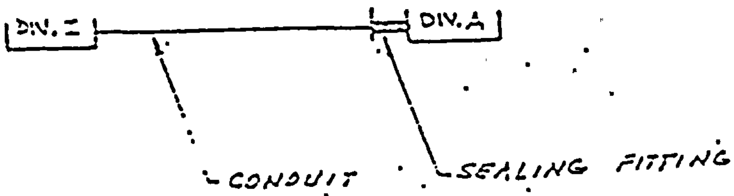


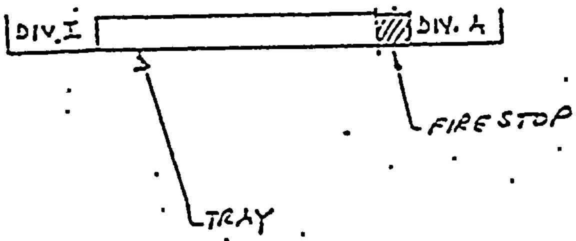
NON-CLASS 1E CIRCUITS

CASE NO. 1



SEISMIC
AND QUALITY
CLASS I.

CASE NO. 2



SEISMIC
AND QUALITY
CLASS I

	1	3	4	5	6	7	8	9	10	11	12
		LOAD TYPE									
CABLE SIZE (AWG)	SERVICE VOLTAGE (VOLTS)	MOTORS (ALL EXCEPT COL. 11)	MOTOR OPTD. VALVES	SOLENOID VALVES	SPACE HEATER (INC. MOTOR HEATER)	PROCESS HEATER	TRANSF. (INCL. PWR & LIGHT'G.)	FDR'S TO SWGR & LOC. CONT.	METERING, PROTECTION & CONTROL CKTS.	SMALL MOTORS (* *)	
#10 AND SMALLER	120 VAC 125 VDC	P	P	C	SEE NOTE 2 C (UP TO 900 W.)	P	P	SEE NOTE 2 C (UP TO 30A. CIRCUITS)	C	C	
LARGER THAN #10	120 VAC 125 VDC	P	P	C	P	P	P	P	C	N.A.	
ANY	ABOVE 120 VAC, 125 VDC	P	P	N.A.	P	P	P	P	C	N.A.	

031.100-24

NOTES:

1. (* *) INCLUDED ARE: ELECTRO HYDRAULIC OPERATORS (EHO'S), HVAC DAMPERS, REACTOR START-UP RANGE DETECTOR DRIVE MOTOR, MOTORS UP TO 1/3 HP.
2. CONTROL DESIGNATION IS TO BE RETAINED FOR CABLES REQUIRING SIZES LARGER THAN #10 AWG FOR VOLTAGE DROP REDUCTION.

LEGEND

P - POWER

C - CONTROL

N.A. - NOT APPLICABLE

Prime designator (i.e. A', B' etc.) indicates a non-Class IE circuit connected to a Class IE power source.

TABLE IV
DIVISIONAL COMPATIBILITY

DIVISIONS	1	2	3	4	5	6	7	A'	A'(9000)	B'	B'(9000)	XXX I'	XXX II'	XXX III'	XXX I	XXX II	XXX III	A	B
1	X			X		X		X	X			X			X	X	X	X	X
2		X			X		X			X	X		X		X	X	X	X	X
3			X											X	X	X	X	X	X
4	X			X		X		X	X			X			X	X	X	X	X
5		X			X		X			X	X		X		X	X	X	X	X
6	X			X		X		X	X			X			X	X	X	X	X
7		X			X		X			X	X		X		X	X	X	X	X
A'	X			X		X		X	X			X			X	X	X	X	X
A'(9000)	X			X		X		X	X			X			X	X	X	X	X
B'		X			X		X			X	X		X		X	X	X	X	X
B'(9000)		X			X		X			X	X		X		X	X	X	X	X
XXX I'	X			X		X		X	X			X			X	X	X	X	X
XXX II'		X			X		X			X	X		X		X	X	X	X	X
XXX III'			X											X	X	X	X	X	X
XXX I	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
XXX II	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
XXX III	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
A	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
B	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

X – DENOTES NO SEPARATION REQUIRED (ie, COMPATIBLE DIVISIONS)

NOTE; CABLES SPECIFICALLY ASSIGNED TO NMS LPRM INPUTS, AND THE RPS SCRAM GROUP OUTPUTS, SHALL STRICTLY ADHERE TO FOUR DIVISIONAL SEPARATION

Q. 031.101
(7.3.2)

In Section 7.3.2.2.2.3.1.4 of the FSAR, you state that: "All components used in the isolation system have demonstrated reliable operation in ... industrial applications". Vague or general statements like this are unacceptable without the supporting basis. Accordingly, identify in Section 7.3.2 of the FSAR, the equipment which has been environmentally qualified by previous operating experience and, for each item, provide the basis for the extrapolation in accordance with the requirements of IEEE Standard 323-1971.

Response:

With the Chapter 7.0 rewrite, FSAR Section 7.3.2.2.2.3.1.4 has been removed and reference made to Sections 3.10 and 3.11. However, in selecting such components, priority is given to seleching those which have in the past demonstrated reliable operation in nuclear applications. If such components do not exist, only those which have demonstrated reliable "industrial applications" are used. Furthermore, regardless of whether or not the component selected has a nuclear or industrial history, the above section states that, "...qualification tests or analyses will be conducted (on all components) to qualify the items for this application. See 3.10 and 3.11."

Records of such qualification tests are available at GE, San Jose for NRC staff audit.

In addition, WPPSS has an environmental qualification review program underway. The results of that program will be factored into revised 3.10 and 3.11 which will respond to concerns such as those expressed in this question.

Q. 031.102
(7.3.2)

Describe in Section 7.3.2.1 of the FSAR, your proposed methods to provide for emergency operation of emergency switches and valves which are locked. (Refer to your discussion on locked safety equipment in Section 7.3.2.1.2.3.1.14(d) of the FSAR.)

Response:

The ~~forthcoming~~ revision of Chapter 7.0 discusses this subject in paragraph 7.3.2.1.2.14, which states:

> "Access to means of bypassing any safety action or function for the ESF systems is under the administrative control of the control room operator. The operator is allerted to bypasses as described in Section 7.1.2.4, Regulatory Guide 1.47.
>
> Control switches which allow safety system bypasses are keylocked. All keylock switches in the control room are designed such that their key can only be removed when the switch is in the "accident" or "safe" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Further, the key locker will be audited once per day by the shift supervisor. Should a key be required to change a valve position, it will be obtained from the shift supervisor via approved key control procedures."

Q. 031.103
(7.3)
(7.6)
(F7.6-5)

Your description of the area temperature monitoring system in Sections 7.3 and 7.6 of the FSAR, is insufficient. Accordingly, provide the following additional information:

a. Identify the interfaces between the Class 1E and non-Class 1E parts of this system.

b. Describe how redundant components are electrically isolated and physically separated.

c. Describe how the electrical isolation devices were qualified and indicate the range of this qualification in terms of voltages and currents.

d. Provide the schematics for the Class 1E portions of this system, including the isolation devices.

e. Provide the bases and methods which were used to select the samples which were tested in accordance with the criteria identified in Item (c) above.

Response:

a. Interfaces of the area temperature monitoring system are shown on Elementary Diagram 807E154TC Revision 9, Leak Detection System, which were provided to you per Chapter 1.7 of the FSAR. The local temperature elements and control room temperature trip switches are Class 1E. The output of each temperature trip switch interfaces to a common non-1E meter module and meter and to the non-1E annunciator and computer circuitry. There is one set (meter module and meter) per control room panel, consequently one set per division of instrumentation. The separation interface with the annunciators and computer are protected by relays (contact-to-contact) or temperature switches (contact-to-contact).

b. Power sources, sensors, and wiring for essential circuits are physically separated and electrically isolated, as described in FSAR Section 8.3.1.4.

Even though separation of Class 1E and non-Class 1E circuits was not a design requirement for WNP-2, electrical isolation of power sources between essential and non-essential equipment is provided by relays (coil to contact). Refer to elementary diagram 807E154TC, Revision 9.

Redundant control room components are physically separated, where possible, by placing them in separate control room logic cabinets. Where it is not possible to place redundant control room components in separate cabinets, separation is achieved by surrounding redundant wiring and equipment in metal encasements or providing 6" physical separation as far as practicable.

b. Sensor devices are separated physically such that no single failure (open, closure, or short) can prevent the safety action. By the use of conduit and separated cable trays, the same criterion is met from the sensors to the logic cabinets in the control room. The logic cabinets are so arranged that redundant equipment and wiring are not present in the same bay of a cabinet (a bay is defined by adequate fire barriers). Redundant equipment and wiring may be present in control room bench boards, where separation is achieved by surrounding redundant wire and equipment in metal encasements, or by 6" physical separation as far as practicable. From the logic cabinets to the isolation valves, separated cable trays or conduit are employed to complete adherence to the single-failure criterion.

c. No electrical isolation problem exists between the redundant components of the temperature monitoring system because of physical and electrical separation of the redundant channels.

The trip relays providing electrical isolation between essential and nonessential equipment were not originally required to be qualified as Class 1E isolation devices on the WNP-2 design.

However, the relays used in the WNP-2 design are qualified as Class 1E. The qualification was completed to IEEE 323-1971 and IEEE 344-1971 using GE Qualifications Specifications. The contact-to-contact rating of the Agastat GPI is as follows: 1000 MEGOHMS @ 500-Vdc insulation resistance and 1200 VRMS @ 60 hertz dielectric.

The rating for the GE type HMA relay was tested to ANSI 37.9 and has a contact-to-contact dielectric strength of 1500 VRMS @ 60 hertz.

The rating for the relay used in the temperature point modules is hermetically sealed with a dielectric strength of 500 VRMS between open contacts; 500 VRMS between coil and case; and 1000 VRMS between all other combinations.

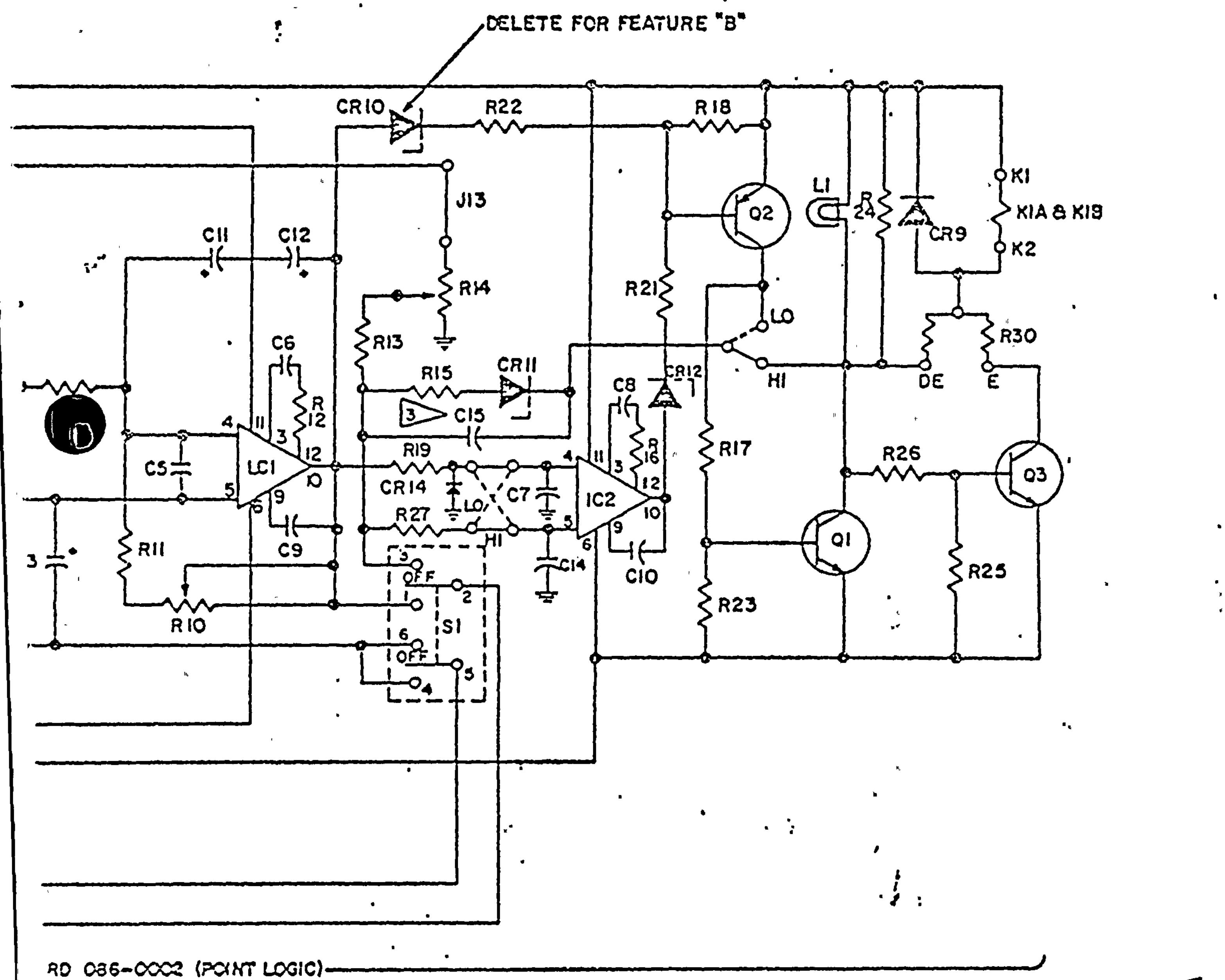
The interface between the Class 1E temperature trip switch and the non-1E meter module takes place within the Class 1E temperature switch. An individual momentary toggle switch is contained in each temperature trip switch. Manual initiation of the toggle switch is required before any temperature trip switch electronics could interface with a signal going to the meter module. This manual toggle switch would be initiated during testing, calibration, or surveillance monitoring, and then only one switch at a time would be interfaced to the meter module. No specific qualification test has been conducted to classify this device as an isolation element, and it is not felt necessary due to the nature of the application. Switch vendor information identifies this switch as having a 1000-volt (RMS) dielectric strength, which is well above any credible power source available to this system.

d. Refer to the response to part "a" of this question and see attached Figures 1 and 2.

e. The bases and methods used to select the samples which were tested as noted in "c" above are specified in 225A6634, Qualification Specification for Essential Components, which can be made available if desired.

REVISION 39
JANUARY 1978

Figure 1a

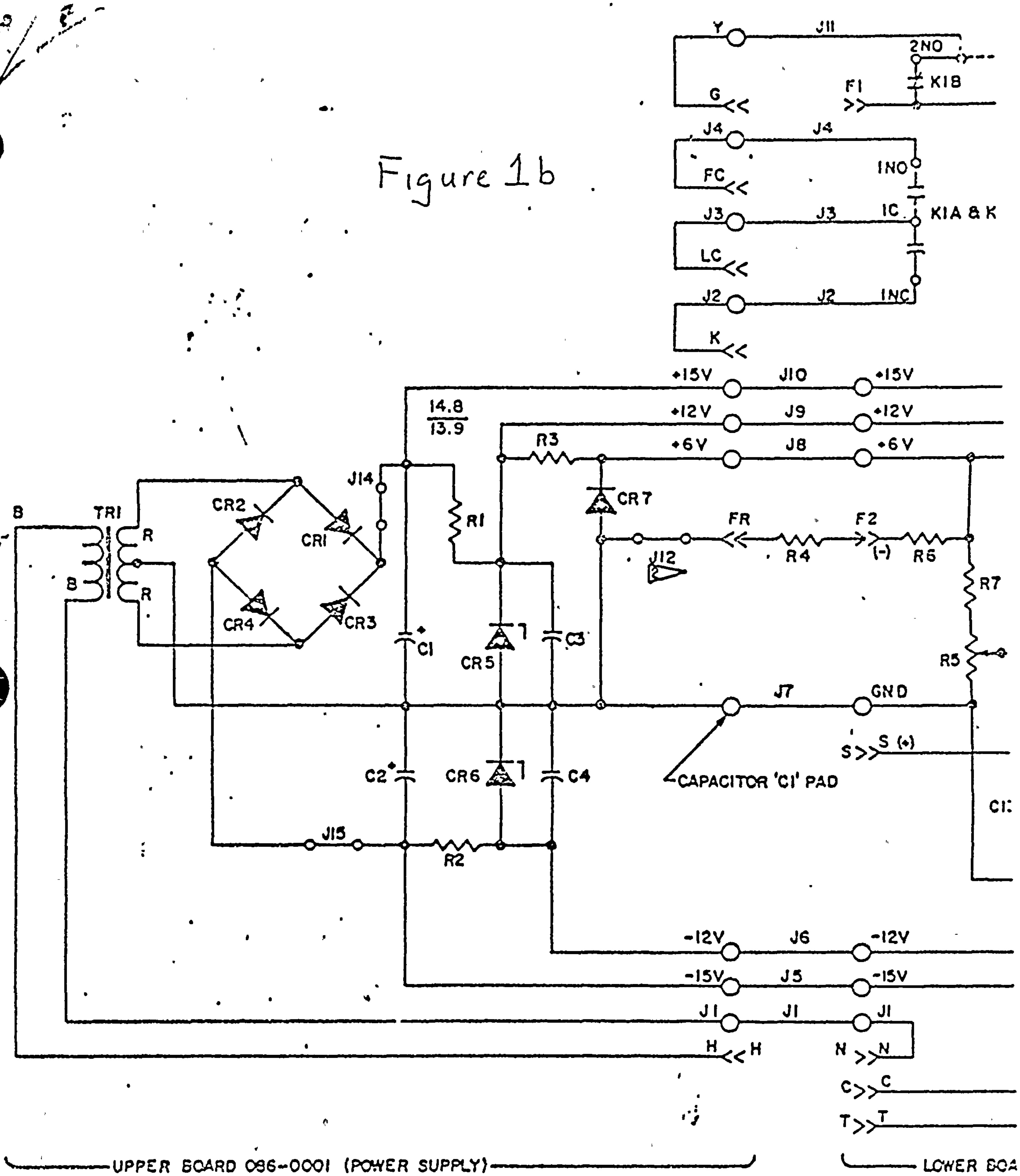


WM. H. ZIMMER NUCLEAR POWER STATION UNIT 1
FINAL SAFETY ANALYSIS REPORT

FIGURE Q221.254-1
AREA TEMPERATURE MONITORING DEVICE
SCHEMATIC
(SHEET 1 of 2)

Q221.354-2

Figure 1b

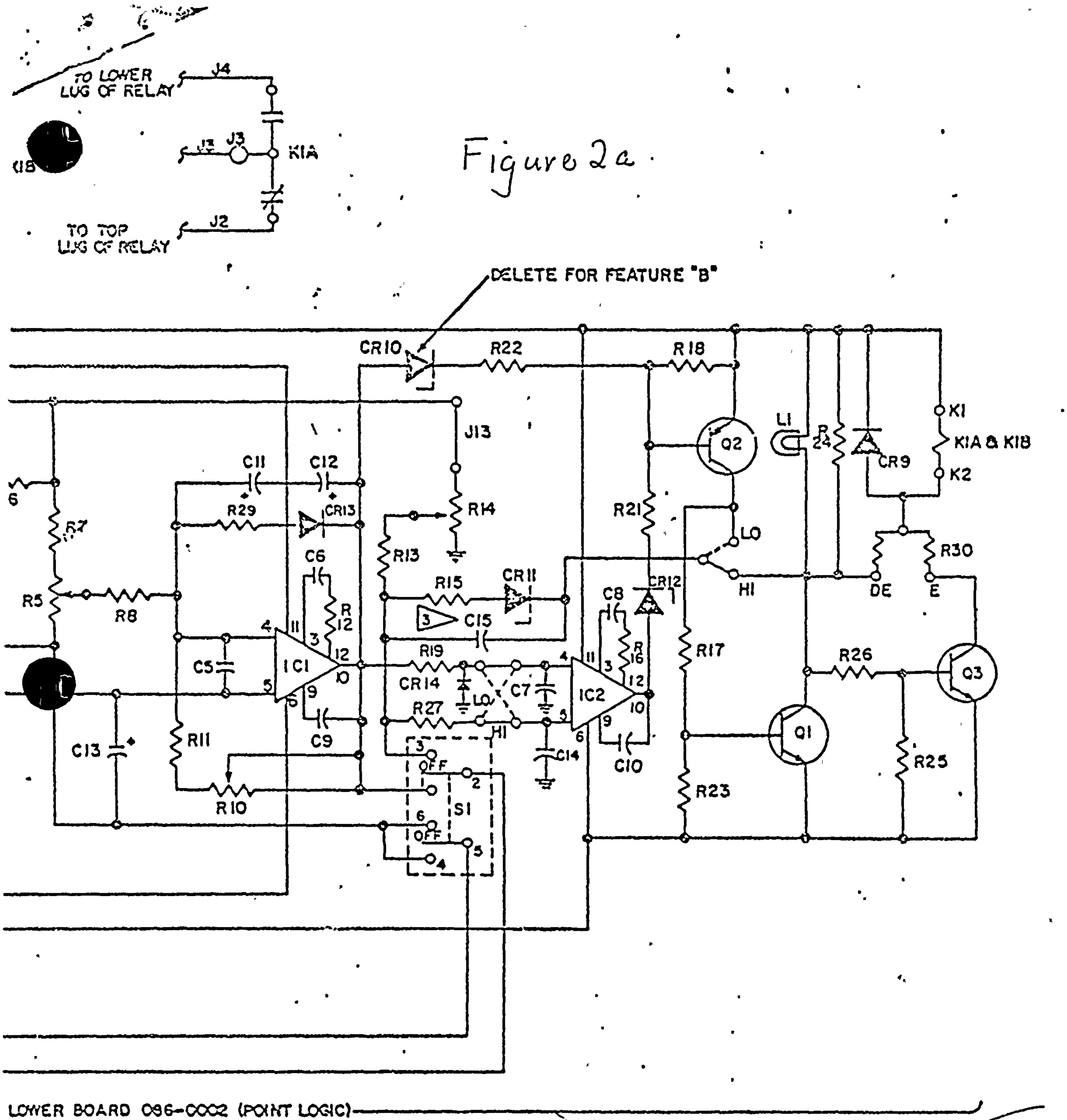


NOTES: 1 ADDITIONAL "FORM C" CONTACT FOUND ON OPTIONAL K1B RELAY

2 FOR "ISA DESIGNATIONS" R OR S J12 BECOMES R29

3 ADD FOR FEATURE "G"

Figure 2a



WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
FINAL SAFETY ANALYSIS REPORT

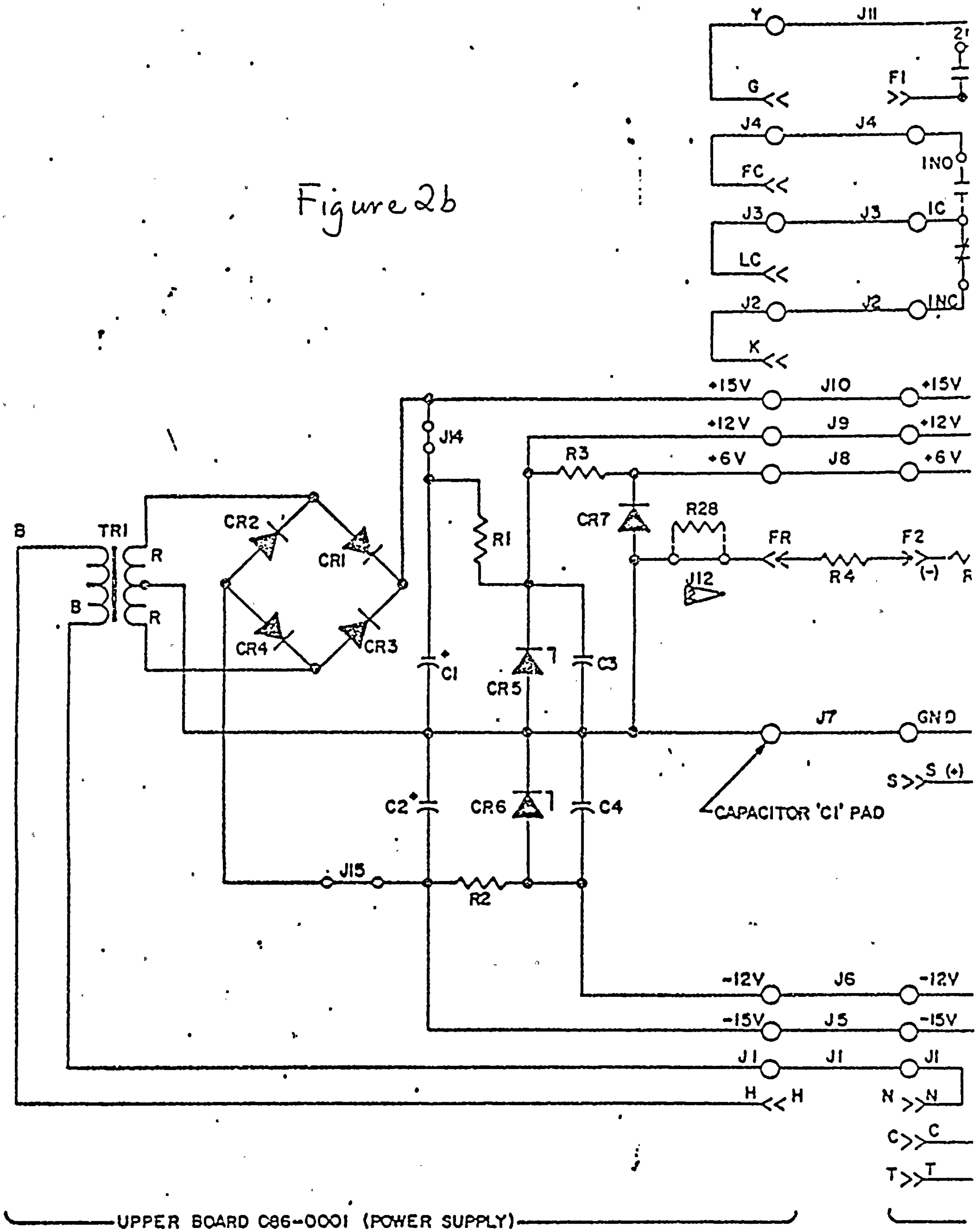
FIGURE Q221.254-1

AREA TEMPERATURE MONITORING DEVICE SCHEMATIC

(SHEET 2 of 2)

Q221.354-3

Figure 2b



NOTES: ADDITIONAL "FORM C" CONTACT FOUND ON OPTIONAL K1B RELAY

FOR "ISA DESIGNATIONS" R OR S J12 BECOMES R2G

ADD FOR FEATURE "G"

Q. 031.104
(7.5)
(031.034)

The description of the control room in Section 7.5 of the FSAR is incomplete as are the figures in this section. (Refer to Item 031.034.) Accordingly, provide a layout drawing of the control room showing in sufficient detail, the following information:

a. The location and identification of each cabinet and panel.

b. The location and routing of each conduit and cable tray and pan.

c. The location and field of each emergency light.

d. The location and identification of each indicator and relay contact which satisfies the requirements of Sections 4.19 and 4.20 of IEEE Standard 279-1971.

Response:

a. The following WNP-2 drawings provide main control room panel and cabinet location and master parts list (MPL) identification numbers: E775 and 127D1625TC.

   Panel MPL idents may be associated to a particular system identification through drawing 238XL93AD.

b. Location and routing of conduits within the main control room are shown on the following WNP-2 drawings: E751, E765, and E766.

   Location of PGCC floor ducts within the main control room is shown on WNP-2 drawing 127D1625TC (see Item a.).

c. The location and field of the WNP-2 main control room emergency lighting are shown on drawing E733.

d. All indications necessary for the plant operator to assess status of protective actions and status of protective functions are located in system level groupings, along with manual system controls, on front row main bench boards. These front row boards are identified on WNP-2 drawings E775 as P601, P602, P603, P800, P820, and P840.

Relays or analog outputs used to actuate indications of protective actions and status of protective functions are located within control room back row cabinets. These cabinets were identified by part a. above.

Copies of the above referenced drawings have been submitted to you under separate cover. The drawings are for information only as they are under continuous review and update.

Q. 031.105
(7.4.1.4)

Your description of the procedure for reactor shutdown from outside the control room is inadequate. Accordingly, provide the following additional information:

a. Provide plant layout sketches which show where the switches are located.

b. Describe the method which will be used to seal the transfer switches.

c. Describe the consequences of an inadvertent actuation of one or more of the switches.

d. Identify and justify each transfer switch which is not wired to the bypass and inoperable status indication system.

e. Describe the methods and indications available outside of the control room by which the operator can: (1) verify relief valve operation; (2) determine the reactor pressure, the coolant level and the coolant temperature; (3) determine the suppression pool level and the temperature of the water in the pool; (4) determine the containment pressure; and (5) determine the service water flow rate and the change in the coolant temperature through the RHR heat exchangers.

Response:

a. Please see revised 7.4.1.4.*

1. Radwaste and Control Building Control Room and Remote Shutdown Room Arrangement Sheet - 1
   B&R Drawing No. E775

2. BOP Remote Shutdown Board
   B&R Drawing No. M618, Sheets RS1 and RS2

3. Remote Shutdown Vertical Board and Legends
   GE Drawing No. 828E466TC and 163C1327TC, Sheets 1 and 2

*Seven (7) copies of the draft figures are submitted separately.

b. The transfer switches will not be "sealed" as such. However, the Remote Shutdown Room will be locked and access administratively controlled. In addition, movement of the transfer from the normal to the "transfer" position will result in an annunciator actuation in the Main Control Room as described in paragraph d below.

c. Inadvertent actuation of one or more transfer switches will result in loss of manual and automatic control, of associated equipment, from the Main Control Room. However, as described in paragraphs b above and d below, access to the transfer switches is administratively controlled and actuation of any transfer switch will result in annunciation in the Main Control Room.

d. Actuation of any Remote Shutdown Transfer Switch which controls safety-related equipment will result in actuation of the associated system level bypass and inoperability indication.

e. Methods and/or indications available to a remote shutdown operator are described in 7.4.1.4.

1. Verification of relief valve operation by observation of the reactor pressure indicator;

2. Determination of reactor pressure as noted in 1 above; determination of coolant level by the level indicators; and determination of coolant temperature by reference to saturated steam table curves;

3. Determination of suppression pool level and water temperature by associated level and temperature indicators;

4. Determination of containment presure by associated pressure indicators;

5. Determination of the RHR heat exchanger service water flow rate by associated flow indicators. Reactor pressure vessel parameters can be used to determine the operability of the RHR Heat Exchanger.

Q. 031.106
(7.6.1)

Confirm in 7.6.1.13 of the FSAR that the primary containment atmosphere monitoring system, including its associated sensors, will be seismically and environmentally qualified. Identify and justify all exceptions. Describe your methods to seismically qualify the drywell hydrogen and oxygen monitoring system. Indicate the required response spectra for which the system is qualified and identify the limiting component.

Response:

The response to this question will be provided upon completion of the seismic and environmental re-evaluation. See response to Question 031.006.

Due to a complete rewrite of Chapter 7.0 in Amendment 10, the information in 7.6.1.13 has been moved to 7.5.1.5.

Q. 031.107
(7.6.2)

In Section 7.6.2.13.3.5.19 of the FSAR, you indicate that the post-LOCA containment monitors are in continuous operation. This appears to be a change from previous boiling water reactor (BWR) designs in which the hydrogen and oxygen subsystems were activated by an accident signal. Discuss this new feature of your design.

Response:

WPPSS recognizes that the hydrogen and oxygen analyzers are only required to perform their function after a LOCA. However, WPPSS has chosen to have this system on line during normal plant operation thereby assuring availability after a LOCA. The discussion of hydrogen and oxygen analyzers has been moved from 7.6 to 7.5.

~~*Draft FSAR page change attached.~~

Q. 031.108
(3.11)
(4.4.3)
(5.4.1)

Figures 5.4-2, 7.7-7, and 7.7-8 and Sections 3.11, 4.4.3.3, 5.4.1, 7.6.1.8, and 7.6.2.8 of the FSAR contain many discrepancies and are, therefore, unacceptable. Provide a consistent set of drawings and other information which represents the design of the reactor recirculation system for the WNP-2 facility. In your response to this item:

a. Provide setpoint information (i.e., the range of the instrument, its accuracy and its setpoint) for the following items: (1) the low total feedwater permissive (H13-P634); (2) the steam line recirculation pump differential temperature (K634); (3) C001A rated speed permissive for CB3A Trip 2; (4) a pump speed greater than 15 percent but less than 40 percent; (5) C001A less than rated voltage permissive for closing CB2A; (6) the generator protective trip voltage; and (7) the reactor power permissive for low speed start.

b. Identify which control option of note 7 in Figure 7.7-8 of the FSAR is applicable to the WNP-2 facility.

c. Identify the inputs which are received from Reference Document No. 4 of Figure 7.7-7a of the FSAR (i.e., C12-1050). Provide this reference and clarify the function of these trips and the ATWS trips shown on Sheet f of Figure 7.7-7.

d. Clarify the discrepancy between the ATWS trips shown in Figure 7.7-7 of the FSAR and the logic description given in Section 7.6.1.8.1.

e. Indicate the signal source for the "permissive when low speed auto start sequence is not activated". (Refer to Figure 7.7-7 of the FSAR.)

f. Indicate the signal source for the "transfer to high speed initiated" auxiliary device. (Refer to Figure 7.7-7 of the FSAR.)

h. Clarify the discrepancy between the setpoint stated in Sections 4.4.3.3.3.a and 5.4.1.3 of the FSAR.

Response:

A response to this question will be provided in a future amendment.

Q. 031.108
(3.11)
(4.4.3)
(5.4.1)

Figures 5.4-2, 7.7-7, and 7.7-8 and Sections 3.11, 4.4.3.3, 5.4.1, 7.6.1.8, and 7.6.2.8 of the FSAR contain many discrepancies and are, therefore, unacceptable. Provide a consistent set of drawings and other information which represents the design of the reactor recirculation system for the WNP-2 facility. In your response to this item:

g. Describe the initiating circuitry for, and the location of, the hydraulic line containment isolation valves.

i. Provide justification for not environmentally qualifying the 6.9 Kv switchgear.

Response:

g. For the initiating circuitry and the location of the hydraulic line containment isolation valves please see Table 6.2-16.

i. The 6.9 Kv non-Class 1E switchgear supplies the non-Class 1E reactor recirculation pumps, cooling tower substations and auxiliary substations and therefore is not required to be environmentally qualified. (Reference FSAR Section 8.3.1.1.1.)

Q. 031.109
(7.7.1)

Provide in Section 7.7.1.2 of the FSAR, the results of a failure mode and effects analysis for the reactor manual control system analyzer. Identify the design features which are provided to detect these failures. Describe the test procedures, including the test frequency, which will be used to detect these failures.

Response:

The limiting failure mode of the reactor manual control system (RMCS) leads to a continuous rod withdrawal. The rod worth minimizer (RWM) and rod sequence control system (RSCS) provide rod block inputs to the RMCS to mitigate this event in the start-up range (low power). However, as described in the response to 031.098, an analysis provided in NEDO 23842 shows that the neutron monitoring system scram (either IRM or APRM), which is independent of the RMCS, adequately terminates the event assuming RWM and RSCS failure to block rod withdrawal. In the power range, the rod block monitor (RBM) provides the block to the RMCS (see the response to 031.052).

The operation and design features of the RMCS are described in FSAR Chapter 7.0, paragraph 7.7.1.2.B.1. Briefly, the system is designed to continously monitor both the plant status for the presence of rod motion inhibits (rod blocks) and the operability of its various functions. In the event of any system failure, the default condition is a rod motion inhibit. The cause of any failure must be corrected before rod motion can proceed.

The ability of the control rod drive portion of the RMCS (Control Rod Drive Control System - CRDCS) to apply rod motion inhibits is demonstrated in conjunction with normal plant instrumentation surveillance on the neutron monitoring system (NMS). When the various NMS functions are tested they provide rod block inputs to the CRDCS. The generation of rod blocks indirectly confirms the operability of the CRDCS. The surveillance requirements for the NMS are described in Chapter 16.0, Technical Specifications.*

*Draft revised page change attached (change to revised Chapter 7.0).

In either rod motion direction, the A and B messages are formulated and compared each millisecond and, if they agree, the A message is transmitted to the HCU selected by the operator. Continued rod motion depends on receipt of a train of sequential messages because the HCU insert, withdraw, and settle valve control circuits are ac-coupled. The system must operate in a dynamic manner to effect rod motion.

Any disagreement between the A and B formulated messages or the responding echo message will prevent ~~further~~ rod motion. Electrical noise disruptions will have only a momentary effect on system operation. Correct operation of the system will resume when the noise source ceases. In Figure 7.7-5, the three action loops of the solid-state reactor manual control system are depicted:

Operator Follow Mode:

a) The high speed loop (0.0002-sec duration) services the control rod selected by the operator to transmit action commands and receive status indications, i.e. presence of rod blocks.

SCAN Mode:

b) The medium speed loop (0.045-sec duration) continuously monitors the other control rods in the reactor, one at a time, to update their status display.

Self Test Mode:

c) The low speed loop (on the order of 20 to 100-sec duration) automatically exercises one HCU at a time to ensure correct execution of actions commanded. This provides for a continuous, periodic self-test of the entire reactor manual control system.

INSERT

The rod selection circuitry is arranged so that a rod selection is sustained until either another rod is selected or separate action is taken to revert the selection circuitry to a no-rod-selection condition. Initiating movement of the selected rod prevents the selection of any other rod until the movement cycle of the selected rod has been completed. Reversion to the no-rod-selected condition is not possible (except for loss of control circuit power) until any moving rod has completed the movement cycle.

The direction in which the selected rod moves is determined by the position of four switches located on the reactor control panel. These four switches, "insert", "withdraw", "continuous insert" and "continuous withdraw", are pushbuttons which return by spring action to an off position.

The following is a description of the operation of the reactor manual control system during an insert cycle. The cycle is described in terms of the insert, withdraw, and settle commands from the reactor manual control system. Figure 7.7-3 can be used to follow the sequence of an insert cycle.

With a control rod selected for movement, depressing the "insert" switch and then releasing the switch energizes the insert command for a limited time. Just as the insert command is removed, the settle command is automatically energized and remains energized for a limited time. The insert command time setting and the rate of drive water flow provided by the control rod drive hydraulic system determine

Insert A:

In the event that any discrepancy is detected in one of these three modes of operation, a rod motion inhibit is applied. This situation is alarmed and annunciated on the reactor control console as an "activity disagree" condition. The control rod drive control system is also designed to produce a rod motion inhibit condition should any failure of the system occur.

The cause of the discrepancy or failure must be corrected before rod movement can proceed. Note, however, that this system cannot affect normal shutdown capability via the reactor protection system.

Q. 031.110
(7.7.1)
(15)

The information presented in Sections 7.7.1.5 and 15 of the FSAR is incomplete with regard to load following operations. Accordingly, describe the interfaces between the system dispatcher and the WNP-2 control systems (e.g., the turbine-generator and the recirculation flow control systems).

Response:

See revised 7.7.1.5.B.4.a).

Q. 031.111
(9.5.2)

The description of the intra-plant radio system inspection and testing in Section 9.5.2.2 of the FSAR is inadequate. Describe the preoperational and periodic testing which assures that radio transmissions will not cause spurious operation of relays and, as a result, negate the protective function of Class 1E equipment. (This question is similar to Item 031.123 on the LaSalle docket.)

Response:

During the preoperational test program any systems or components that are affected by radio transmissions during the normal course of testing and operations will be identified on a Startup Problem Report. The Test and Startup personnel and Operating staff will be briefed on the possible interaction between radio transmission and electronic gear, including solid state relays, and each preoperational test will include precaution statement in Section 3.0 of the test procedure to alert personnel to possible interaction with protective relays. This will ensure that the appropriate action will be taken to assure that protective function of Class 1E equipment is not negated or degraded during plant operation. The preoperational testing is described in FSAR Section 14.2.12.1. Security-related radio communications equipment will be surveillance tested periodically as required by Site Physical Security Plan.

Q. 031.112
(7.6.1)
(7.6.2)

Describe in Sections 7.6.1.1 and 7.6.2.1 of the FSAR, how the power cables and the refueling interlock circuits are separated on the refueling crane.

Response:

The refueling interlocks are not safety-related and no separation is required or provided between the power cables and the interlock circuits. See revised Chapter 7.0, Section 7.7.1.13.

Q. 031.113
RSP
(15.4.1.2)

It is our position that the rod sequence control system does not satisfy the requirements of IEEE Standard 279-1971 and, therefore, is unacceptable for the prevention of a control rod withdrawal accident. Accordingly, we require you to provide a modified design for the WNP-2 facility.

Response:

See the response to Question 031.098.

REVISIONS TO PREVIOUSLY SUBMITTED
ICSB QUESTIONS

Q. 031.001(j)

Provide justification for not seismically qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Seismic qualification documentation for the feedwater isolation valve actuators is presently under examination as part of an overall qualification review with results to be provided to your SQRT personnel. The control rod drive excess flow isolation valve has been deleted along with the CRD Return Line.

Q. 031.001(k)

Provide justification for not environmentally qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Environmental qualification documentation for the feedwater isolation valve actuators is presently under examination as part of an overall qualification review to determine degree of compliance with NUREG-0588, the results of which will be provided to NRC personnel. The control rod drive excess flow isolation valve has been deleted along with the CRD Return Line.

Q 31.006

The staff requests that the following information regarding the qualification test program be provided for Class 1E equipment: (a) the equipment design specification requirements; (b) the test plan; (c) the test set up; (d) the test procedures; (e) the acceptability goals and requirements; and (f) the test results.

Provide this information for each of the following Class 1E components: (a) the 4.16 kV switchgear SM 7; (b) the damper operator for WMA-V-52C; (c) the fan WMA-FN-52B; (d) the logic equipment for the standby gas treatment system; (e) the diesel-generator control equipment; (f) the 480 V ESS switchgear MC-7A-A; and (g) the solenoid valve for the main steam line isolation valves.

Response:

An extensive seismic and environmental review program is presently underway encompassing BOP and NSSS scope, with a planned completion date in ~~the second quarter of~~ December ~~1979.~~ 1980

Within the BOP scope, the equipment documentation has been extracted from the contract files, copied and categorized for easy retrieval. Within the NSSS scope, contract negotiations are underway with GE to perform a similar function.

A list of all Class 1E equipment including splices, terminal blocks, termination cabinets and connectors is presently being compiled. This list will contain the following information:

1. Equipment location
2. Safety functional requirement
3. Manufacturer & Model No.
4. Qualification Method (test-analysis)
5. Environmental Extremes
6. Identification and location of qualification documents

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

A Composite ~~The completed~~ list will be included in the FSAR as equipment tables in 3.10 (seismic) and 3.11 (environmental).

The extensive review program underway will also satisfy the requirements of IE circular 78-08, address the degree of compliance with NUREG 0588, and establish the conservativeness of seismic tests and analysis performed to IEEE-344, 1971.

The detailed results of this review will be made available to NRC SQRT and Environmental review personnel during their site documentation reviews.

Q 31.026
(7)

Describe the installation, operation, and removal of the "Startrec" computer system which is used for startup testing of GE boiling water reactors, including the following topics: (a) specifications and qualification testing of electrical isolators; and (b) separation criteria for permanent and temporary wiring.

Response:

Insert - next page

~~This system known as "Startrec" will be provided by General Electric on a loan basis for Startup Testing. This equipment will be located in the main control room and will consist of multiplexed data input terminals, data reduction; and data recording equipment.~~

~~The WNP-2 "Startrec" design implementation consists of both permanent and temporary equipment. Signal inputs will be permanently wired to test jacks located on the signal originating panel for non-safety-related input sources and routed to a central divisionalized panel for signals originating in safety-related equipment. This centralized panel will house isolation devices as well as output test jacks. The non-divisional wiring from all output test jacks will be temporary and routed overhead in the control room to the "Startrec" equipment.~~

~~All signals originating from safety-related divisionalized equipment will be physically and electrically isolated such that faults occurring in the "Startrec" equipment cannot propogate back into safety-related circuits. The isolation devices will be optical in nature, qualified to the standards of Class 1E equipment and meet the intent of Regulatory Guide 1.75 concerning isolation devices. These isolation devices will be mounted in divisional centralized panels where all safety-related equipment inputs will converge. There will be a central isolation panel for each division as required. The output of each isolated input point will be routed through test jacks to the "Startrec" equipment as non-divisional cabling.~~

~~In order to preserve plant availability all analog signal inputs to "Startrec" originating in non-safety-related plant control system equipment will be electrically isolated through isolating amplifiers. This will prevent faults in the "Startrec" equipment from disturbing sensitive control system signals. The output of these amplifiers will be routed directly through test jacks to the "Startrec" equipment.~~

Insert to Page 031.026-1:

The Transient Data Acquisition System (TDAS) to support startup transient testing will no longer be the GE STARTREC computer system. A re-evaluation of the WNP-2 data acquisition needs has led to the ~~proposal~~ implementation of a permanent installation. KDC

The WNP-2 TDAS control unit and analysis computer will be located in the main control room. Analog and digital signals will be isolated and conditioned in divisionalized remote units in the cabinet where they originate. Multiplexing and digitizing of all data will take place in these remote units before the data is transmitted over fiber-optic links to the control unit.

All signals originating from safety-related divisionalized equipment will be physically and electrically isolated such that faults occurring in the TDAS equipment cannot propagate back into safety-related circuits. The isolation devices will be qualified to the standards of Class 1E equipment and meet the intent of Regulatory Guide 1.75 concerning isolation devices.

Q. 031.055

Identify all Class 1E equipment which was not qualified by test. For each such item, provide the basis for assuming that it will not be spuriously operated, or fail to operate when required, during and after a seismic event.

Response:

A program is currently underway to re-evaluate the effects of vibratory loads on Class 1E equipment. This re-evaluation will consider the effects (including spurious operation) of seismic as well as hydrodynamic loads produced for Class 1E equipment.

We anticipate completing the re-evaluation phase of this program in calendar year 1980.

Q 31.056
(3.11)

Describe the environmental qualification procedures and the environmental extremes of qualification for the following specific passive Class 1E components inside the drywell: (1) splices; (2) terminal blocks; (3) termination cabinets; and (4) connectors.

Response:

See response to ~~Q. 31.055~~. Question 031.006.

Q. 031.057
(3.11)

Identify all Class 1E equipment inside the drywell, except the equipment cited in Item 031.056, and summarize the environmental qualifications for this equipment. The identification and summary for each item should include: (1) the safety function and functional requirement; (2) the manufacturer, model number, type number, and any other identifying numbers; (3) the specific location of this equipment the drywell; (4) the method of environmental qualification; (5) the environmental extremes, including the time period of testing, for which it is qualified; and (6) the identification and the location of the documents which are available so as to permit an independent evaluation of the adequacy of the environmental qualification.

Response:

See response to Question 031.006.