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Forwards responses to questions from CSB, Core Performance Branch, & Geosciences Branch, as well as responses or updates to previous questions from CSB. (Only 1 copy of ref repts sent, Filed in Geosciences Branch, 259 pp.) (see reports)

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Washington Public Power Supply System
A JOINT OPERATING AGENCY

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November 21, 1978
G02-78-259

Docket Number 50-397 A

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

Attention: Mr. S. A. Varga, Chief
Branch No. 4
Division of Project Management

Subject: WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO CONTAINMENT SYSTEMS BRANCH,
CORE PERFORMANCE BRANCH, AND GEOSCIENCES
BRANCH QUESTIONS

Reference: Letter, SA Varga, NRC, to WPPSS, "Additional Acceptance Review Questions for the WPPSS Nuclear Plant No. 2", dated September 18, 1978.

Dear Mr. Varga:

Attached please find sixty (60) copies of the responses to the questions from Containment Systems Branch (CSB), Core Performance Branch, and Geosciences Branch submitted to WPPSS in the referenced letter. This submittal is consistant with the schedule provided you at the October 10th meeting (copy attached). Also included are responses or updates to previous questions from CSB where they were required.

These responses will be formally inserted in the FSAR as an amendment after submittal of the responses to the Instrumentation and Control Branch questions (scheduled for January 1).

Very truly yours,

D L Renberger

D. L. RENBERGER
Assistant Director
Technology

DLR:OKE:cph

attachment

cc: JJ Verderber - B&R, w/o responses
JJ Byrnes - B&R, w/o responses
RC Root - B&R Site, w/o responses
HR Canter - B&R, w/o responses
D. Roe - BPA. w/o responses
FA MacLean - GE, San Jose, w/o responses
I. Littman - WPPSS, NY, w/o responses
E. Chang - GE, San Jose, w/5 copies of responses
J. Ellwanger - B&R, w/5 copies of responses
NS Reynolds - Debevoise & Liberman, w/1 copy of responses

IV. SCHEDULE

Containment Systems Branch (18 questions)	Nov 15
I&C Branch (25 questions)	Jan 1
Reactor Systems Branch (4 questions)	Dec 15
Core Performance Branch (3 questions)	Nov 17
Geosciences Branch (4 questions)	Nov 15
Hydrology/Meteorology Branch (1 question)	Dec 15

Docket No. 50-397 Responses to Containment Systems Branch, Core Performance Branch, and Geosciences Branch Questions

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Generation and 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.

20 DLR

DATED November 17, 1978

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 20th day of November, 1978.

Reba B. Helgeson

Notary Public in and for the State of Washington
Residing at Richland

REVISIONS TO PREVIOUSLY SUBMITTED

CSB QUESTIONS

Q 22.7

Provide the secondary containment pressure time response for the design basis accident. List and discuss all assumptions made in this analysis.

Response:

The analysis is continuing along the outline stated in Amendment No. 1. However, the effort is more time consuming than originally contemplated. It is now estimated a reponse will be forthcoming in February 1979.

 78-57

Q 22.10
(6.2.6)

Provide the following information with regard to the leakage rate testing of the Type C containment isolation valve (as defined in Appendix J to 10CFR Part 50):

a. For each fluid line that penetrates the containment, show schematically the isolation valve arrangement and the design provisions that will permit the isolation valves to be leak tested. Indicate the direction in which the valves will be leak tested.

b. Identify the containment isolation valves that will not be subjected to Type C leak testing, and provide justification for not leak testing these valves.

Response

a. All fluid lines which penetrate the primary containment and the associated containment isolation valves which will be subject to Type C testing are illustrated in Figure 6.2-31(a-t). These figures also indicate how the Type C tests will be conducted.

b. The only lines which have not been identified as subject to Type C testing are:

1. All instrument lines (TABLE 6.2-16, NOTE 27)*
2. CRD insert and withdrawal lines (TABLE 6.2-16, NOTE 4)*
3. TIP lines (TABLE 6.2-16, NOTE 29)*
4. RRC hydraulic control lines (TABLE 6.2-16, NOTE 28)*

*TABLE 6.2-16 NOTES REFERENCED PROVIDE THE JUSTIFICATION FOR NOT TYPE C LEAK TESTING. (SEE QUESTION 22.027 FOR REVISED TABLE 6.2-16)

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.

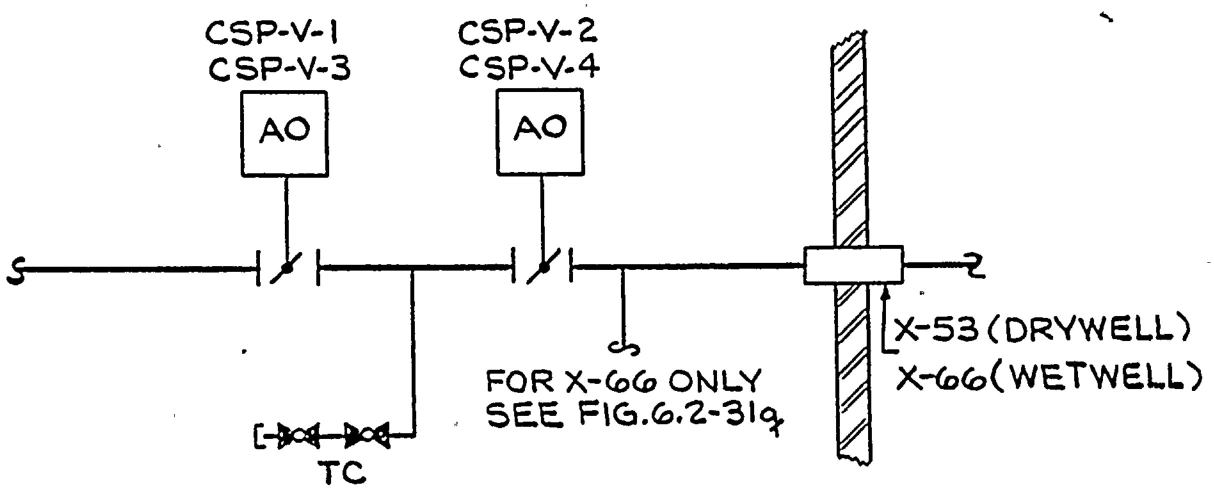
4. TYPE C TESTING IS PERFORMED BY PRESSURIZING BETWEEN THE ISOLATION VALVES. THE TEST YIELDS EQUIVALENT RESULTS FOR THE INBOARD GATE OR BUTTERFLY VALVE.*

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.

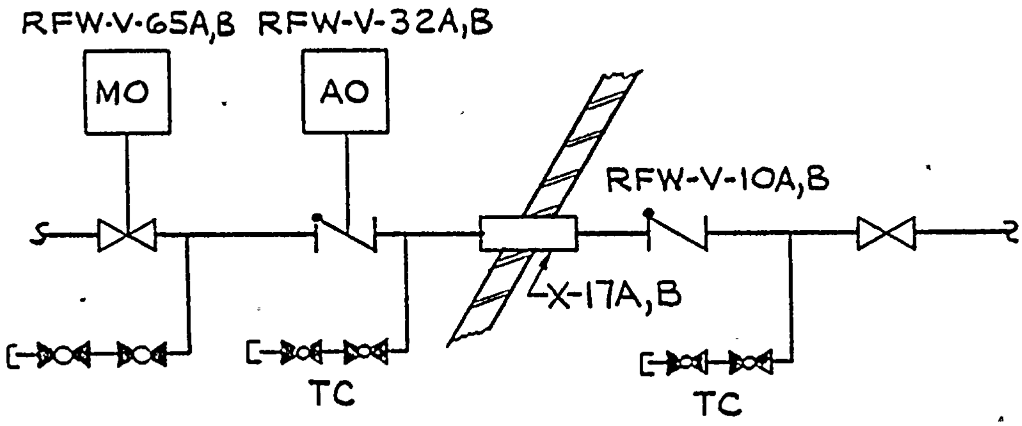
* THE GATE AND BUTTERFLY VALVES ARE EQUALLY LEAK TIGHT IN EITHER DIRECTION BECAUSE OF SYMMETRY OF DESIGN AND BECAUSE OF CONSTRUCTION. THIS FACT HAS BEEN CONFIRMED BY REVIEW OF LEAKAGE TEST DATA AND OTHER INFORMATION SUPPLIED BY THE VALVE MANUFACTURERS.

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.

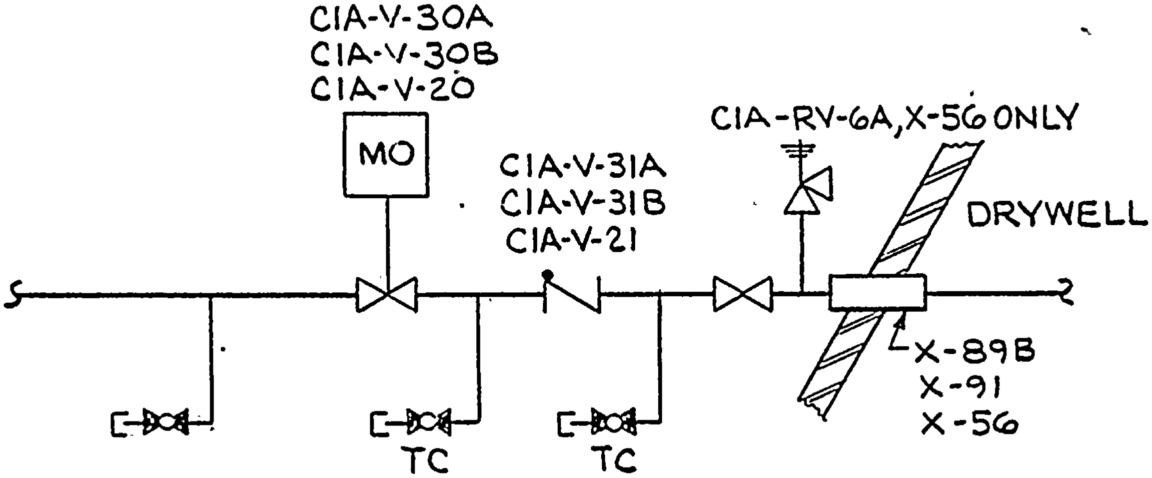
X-53 DRYWELL PURGE SUPPLY
X-66 WETWELL PURGE SUPPLY



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

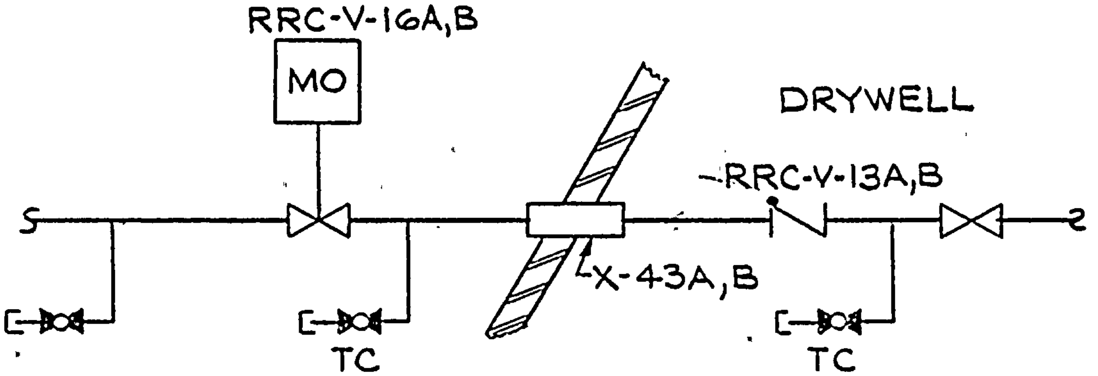
REACTOR FEEDWATER LINES

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2.	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-53, X-66, X-17A & X-17B	FIGU 6.2-31



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

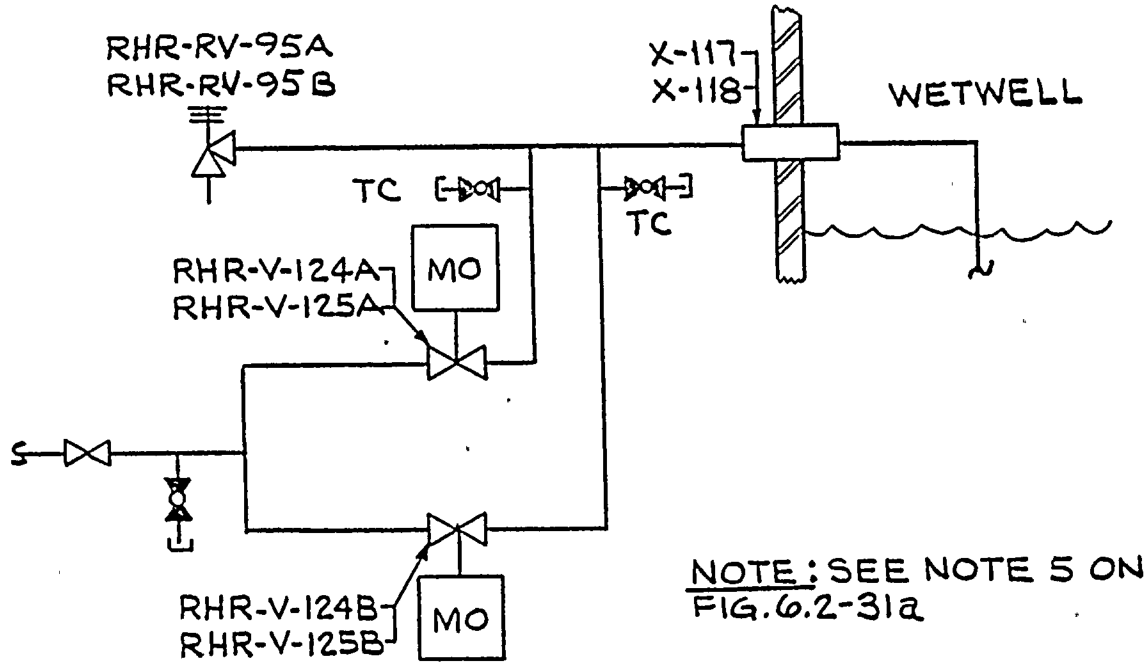
CONTAINMENT INSTRUMENT AIR



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

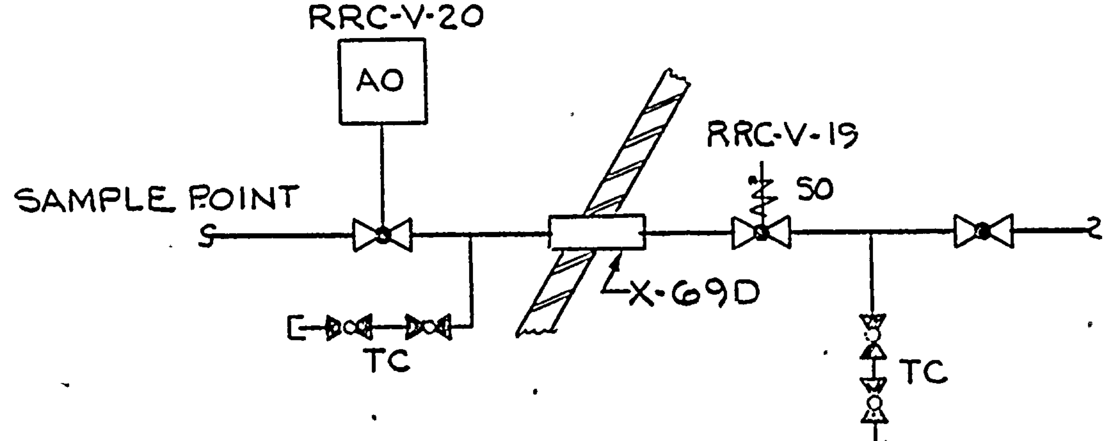
RRC PUMP SEAL PURGE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS - X-89B, X-91, X-56 & X-43A, X-43B	FIG 6.2-



NOTE: SEE NOTE 5 ON FIG. 6.2-31a

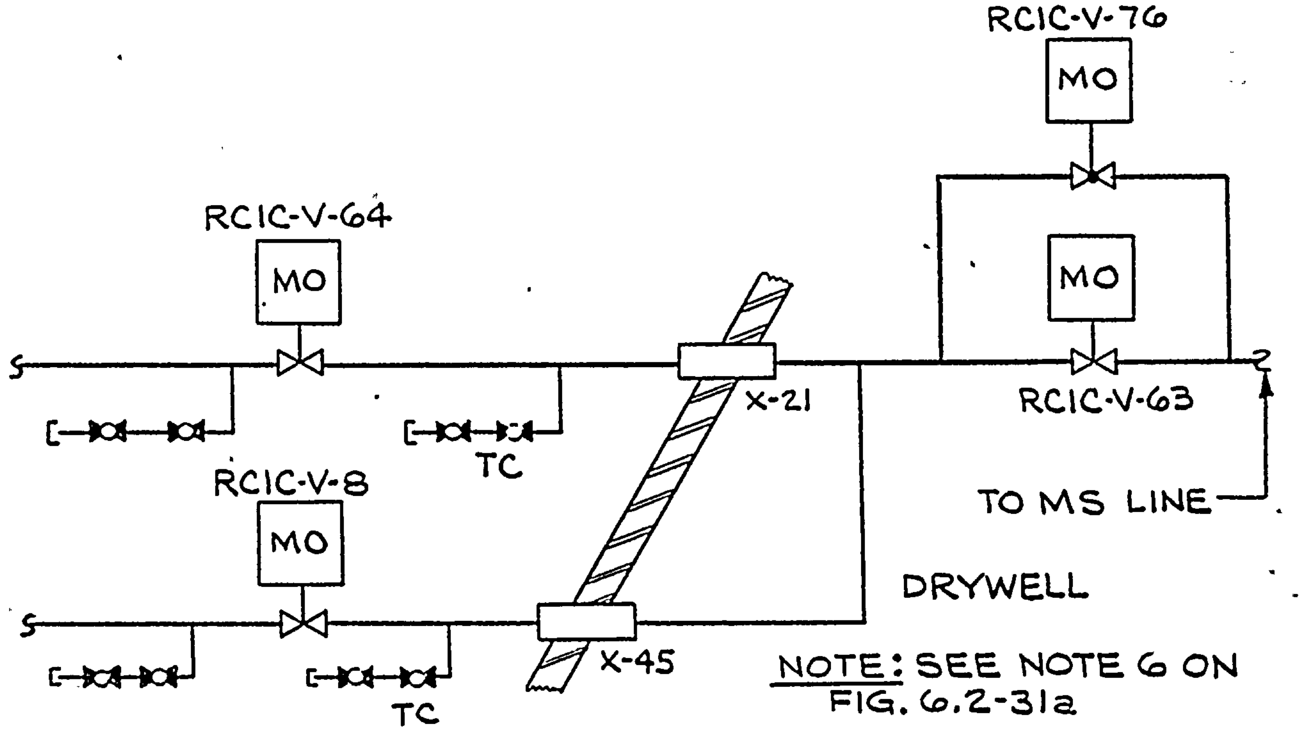
RHR STEAM RELIEF LINES



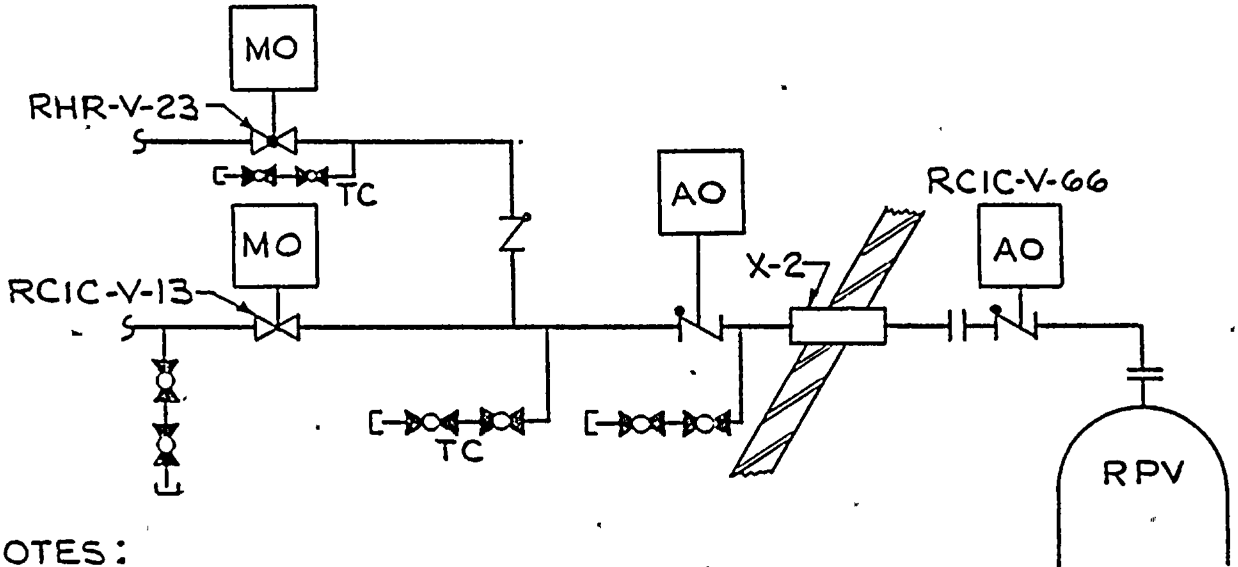
NOTE: SEE NOTE 1 ON FIG. 6.2-31a

RRC SAMPLE LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-117, X-118 & X-69D	FIGU 6.2-3



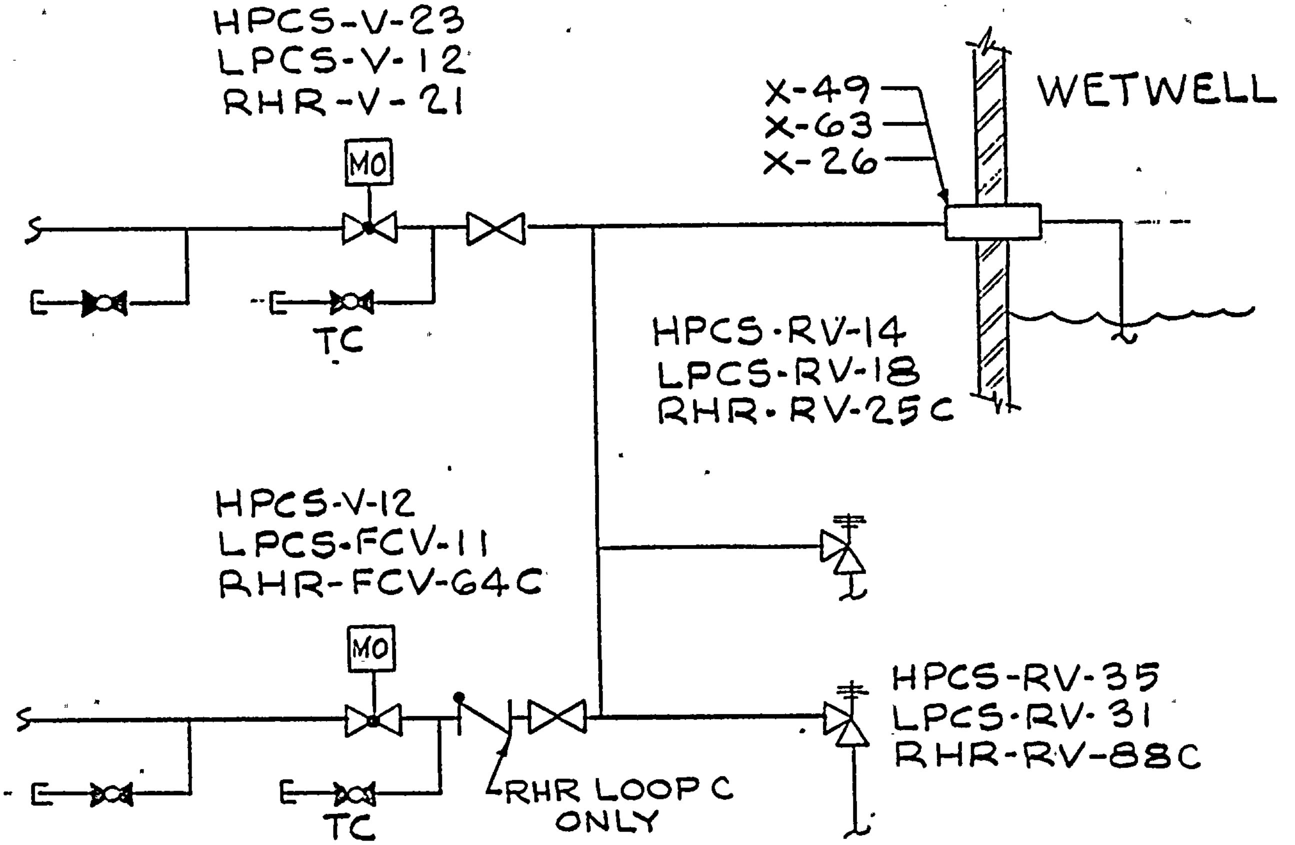
STEAM TO RCIC TURBINE & RHR HEAT EXCHANGER



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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-21, X-45 & X-2	FIGURE 6.2-31e

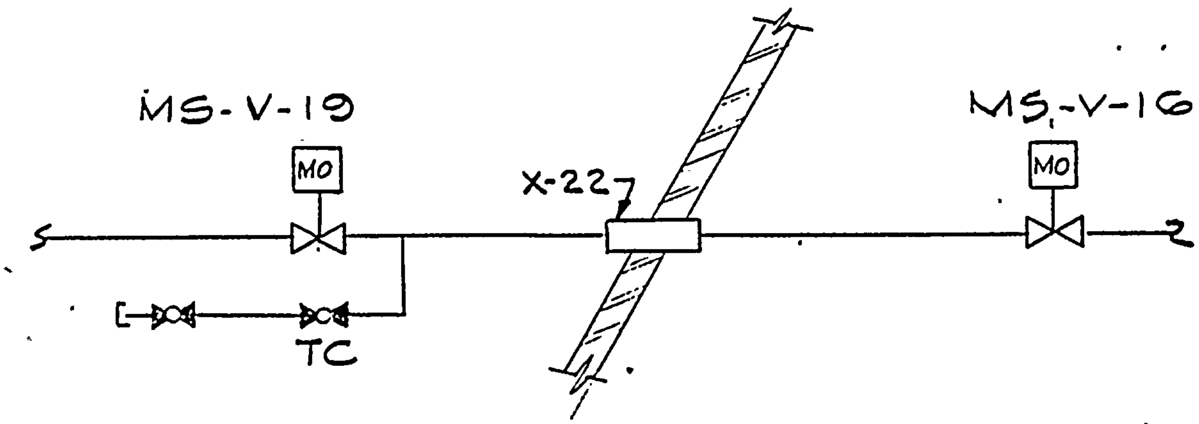


NOTE: SEE NOTE 1 ON FIG. 6.2-31a

X-49 HPCS TEST LINE

X-63 LPCS TEST LINE

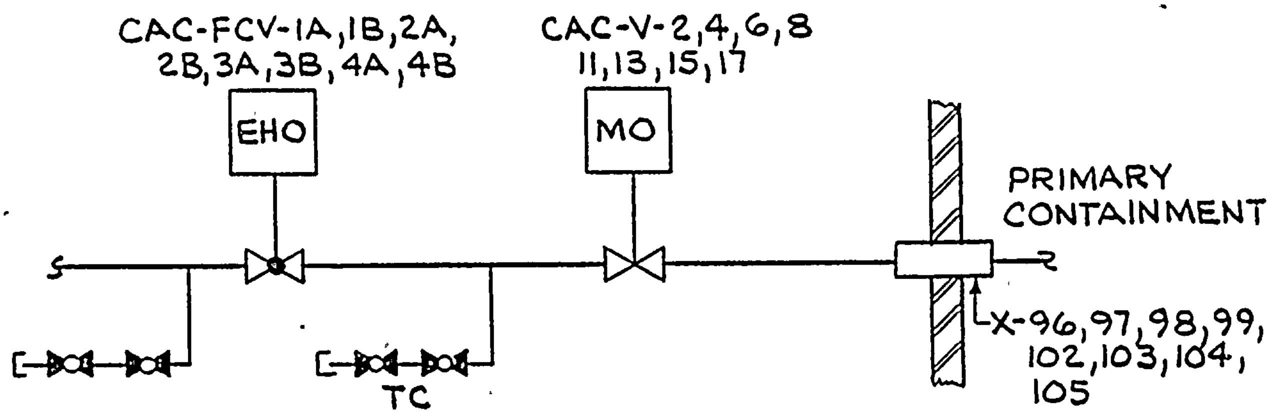
X-26 RHR LOOP C TEST LINE



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

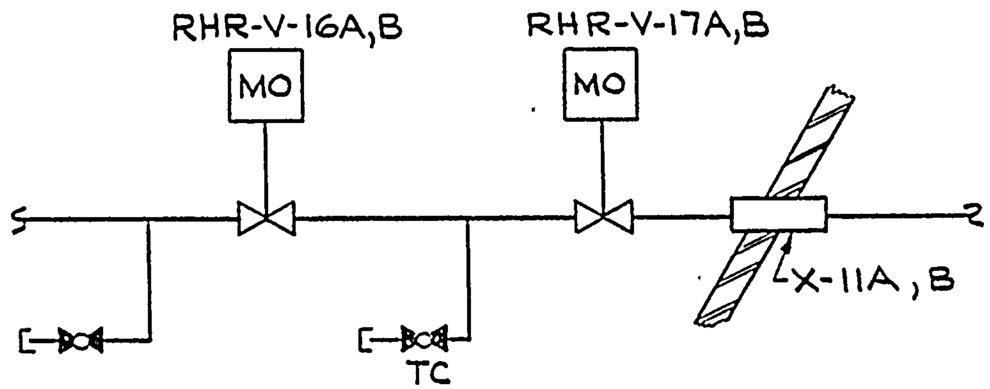
MS DRAIN LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-49, X-63 X-26 & X-22	FIGUR 6.2-31



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

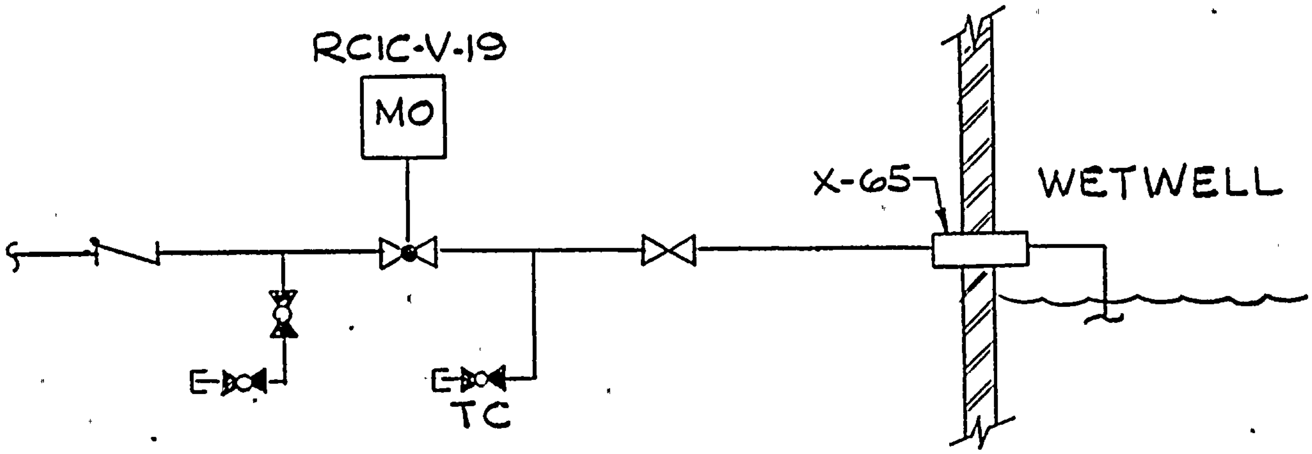
CAC SYSTEM



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

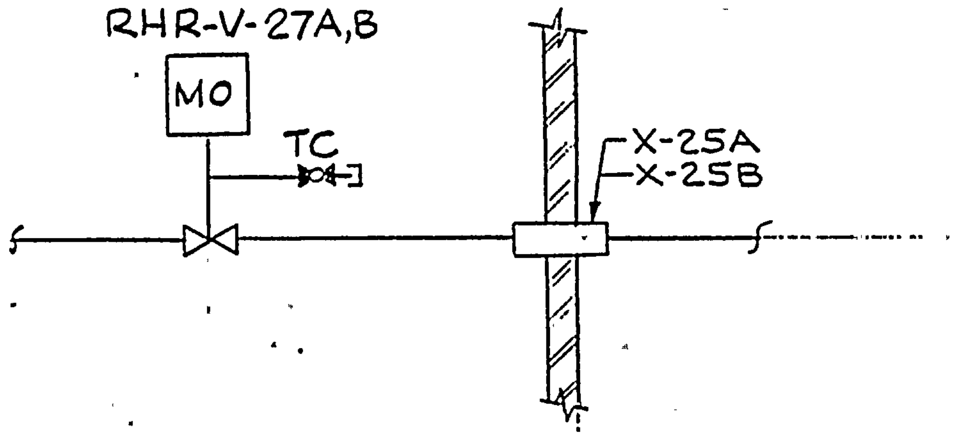
RHR DRYWELL SPRAY

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	FIGU 6.2-3



NOTE: SEE NOTE 1 ON FIG.6.2-31a

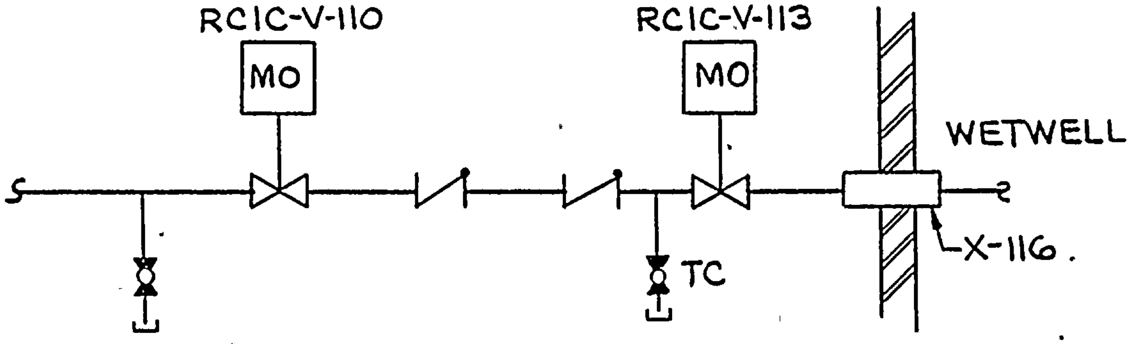
RCIC PUMP MIN. FLOW



NOTE: SEE NOTE 2 ON FIG.6.2-31a

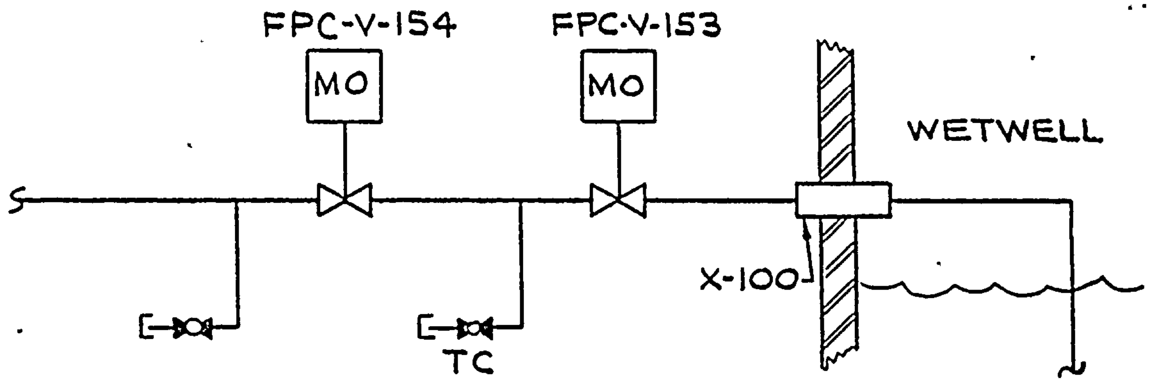
RHR WETWELL SPRAY

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-65, X-25-A & X-25B	FIGU 6.2-3



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

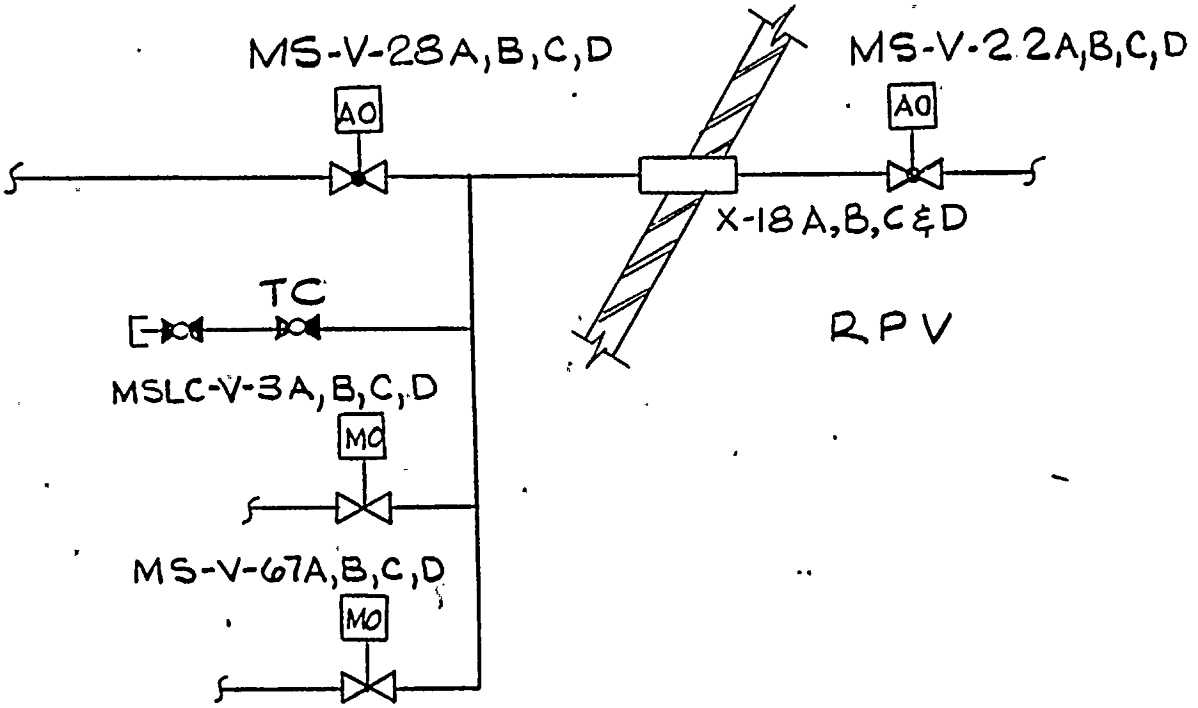
RCIC TURBINE EXHAUST VACUUM BREAKER



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

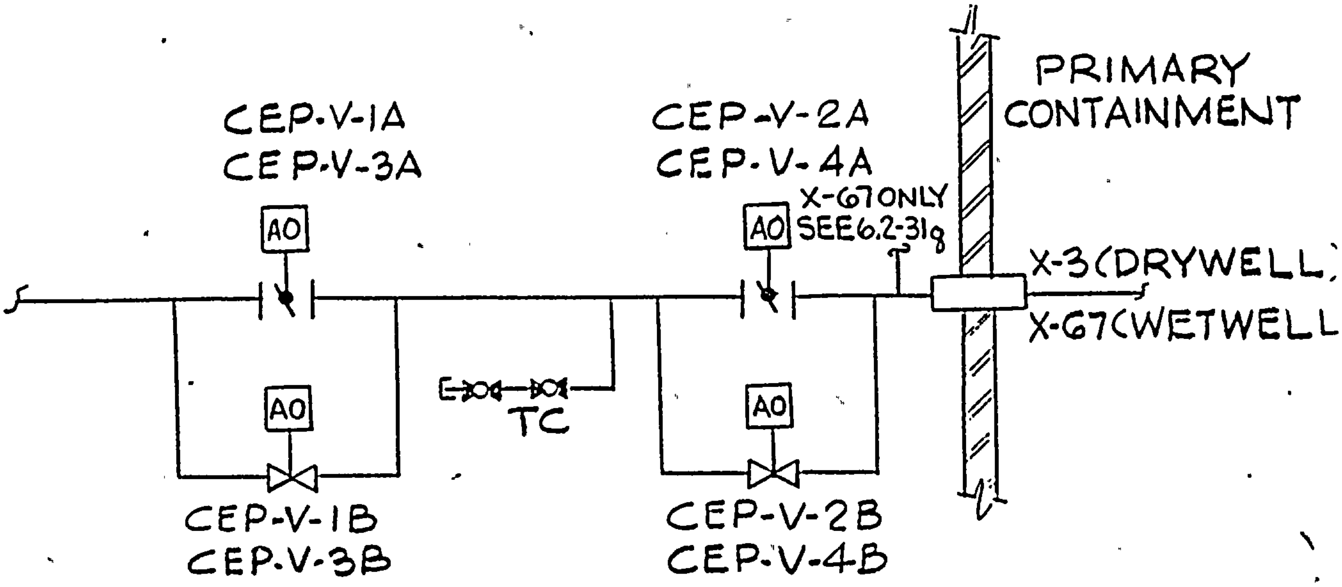
SUPPRESSION POOL CLEAN-UP SUCTION LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2.	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-116 & X-100	FIG 6.2-



NOTE: SEE NOTE 3 ON FIG. 6.2-31a

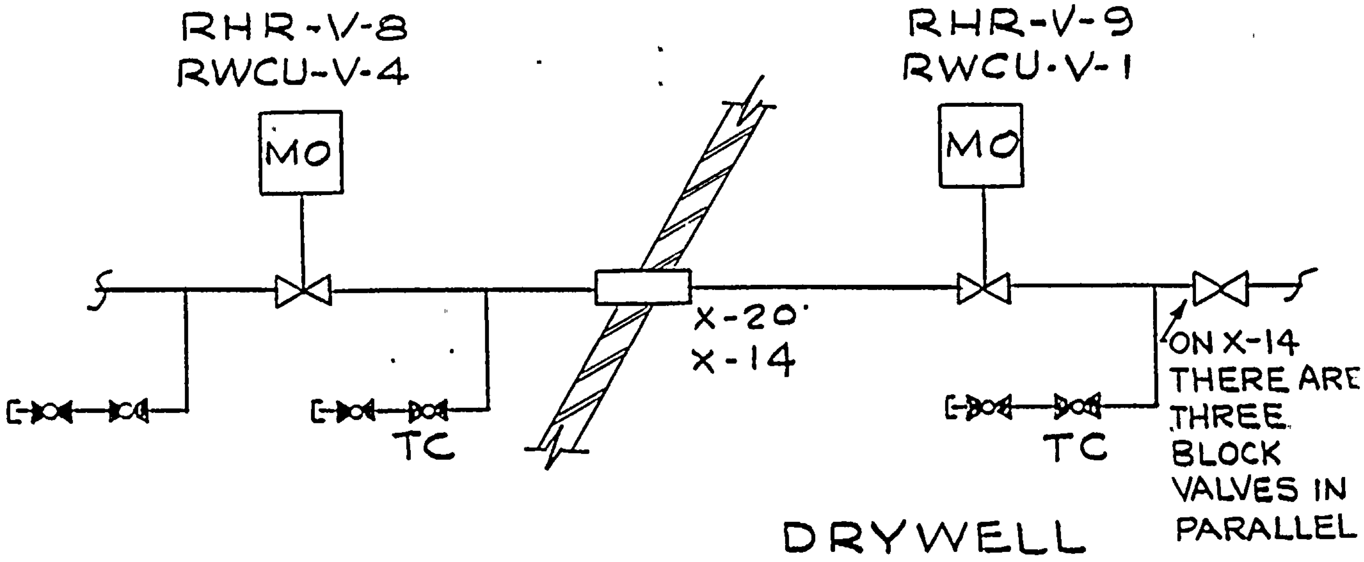
MAIN STEAM LINES



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

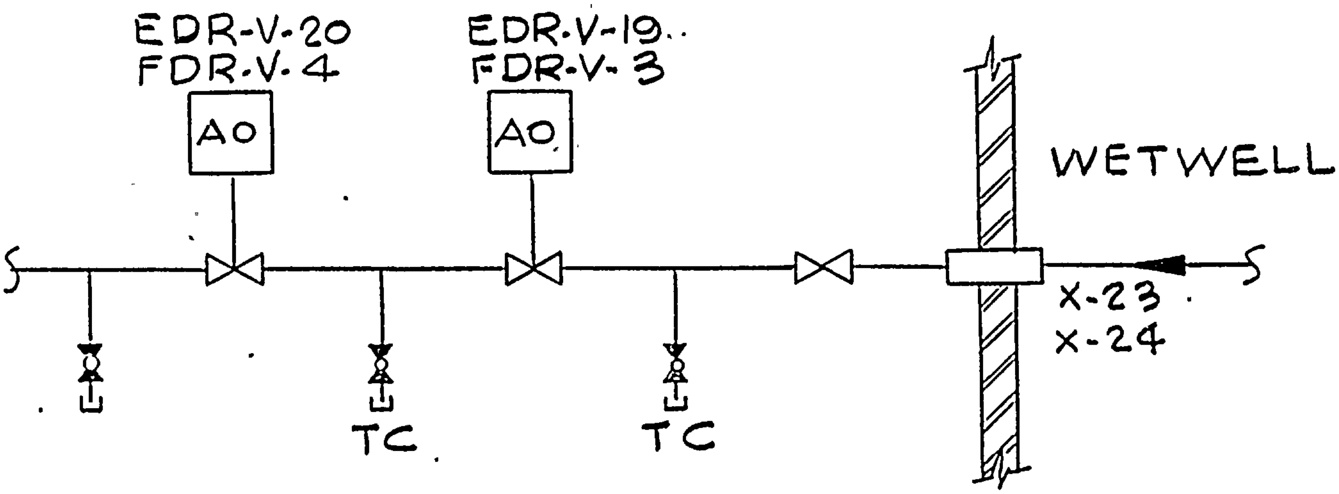
X-3 DRYWELL PURGE EXHAUST
X-67 WETWELL PURGE EXHAUST

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	FIGUR 6.2-31



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

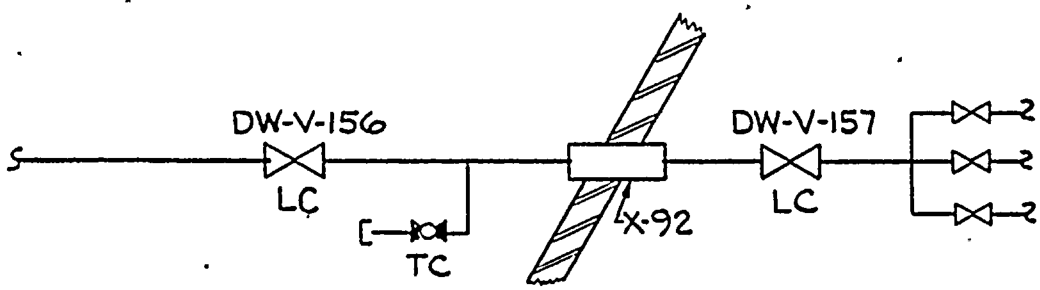
X-20 RHR SHUTDOWN COOLING SUPPLY
X-14 RWCU SUCTION



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

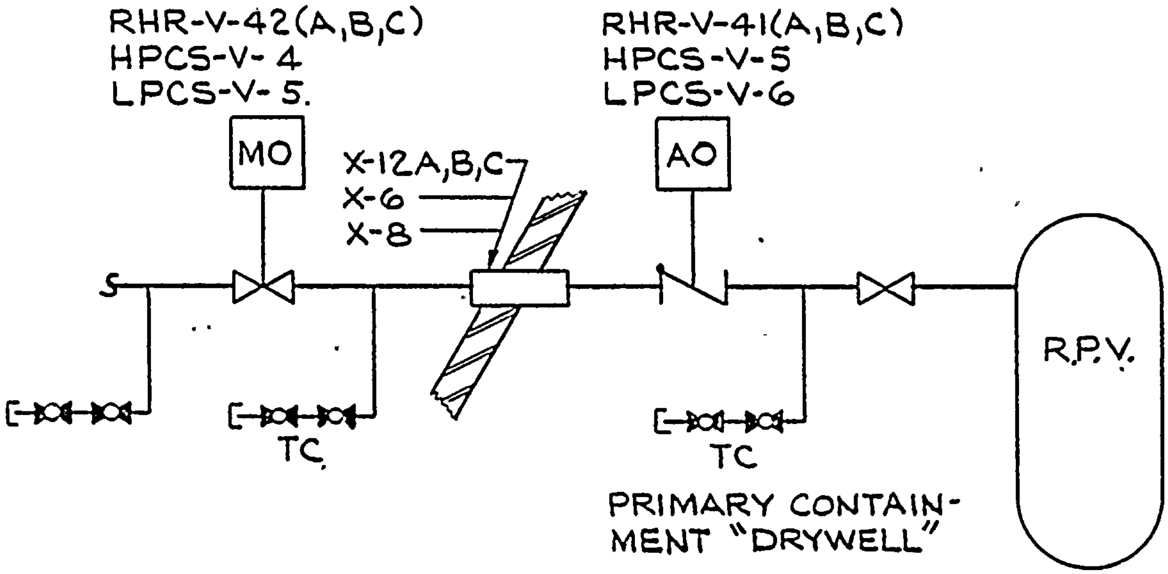
X-23 EDR FROM PRIMARY CONTAINMENT
X-24 FDR FROM PRIMARY CONTAINMENT

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-20, X-14, X-23 & X-24	FIGUR 6.2-31



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

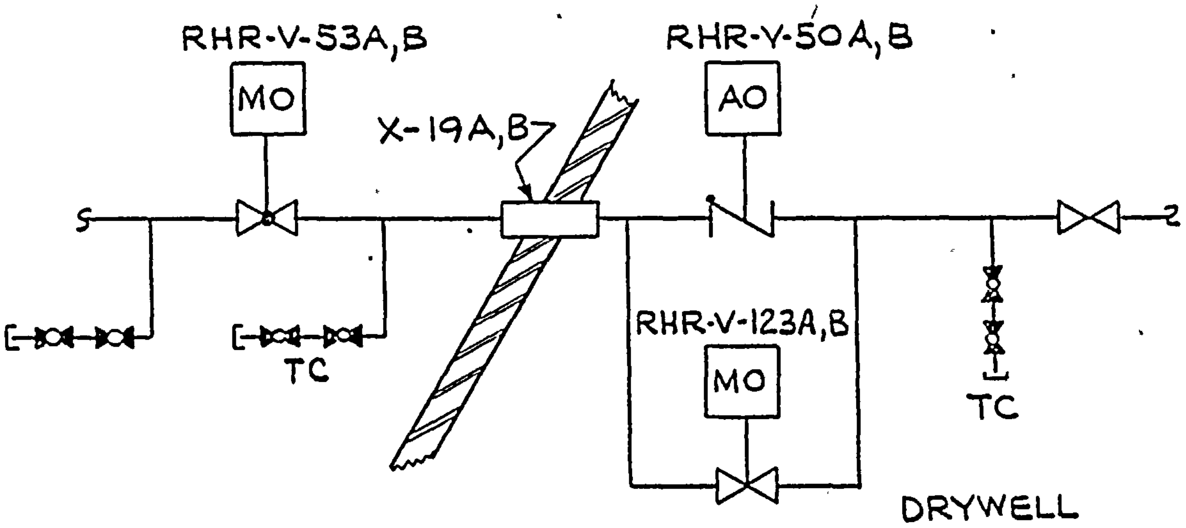
DW SYSTEM



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

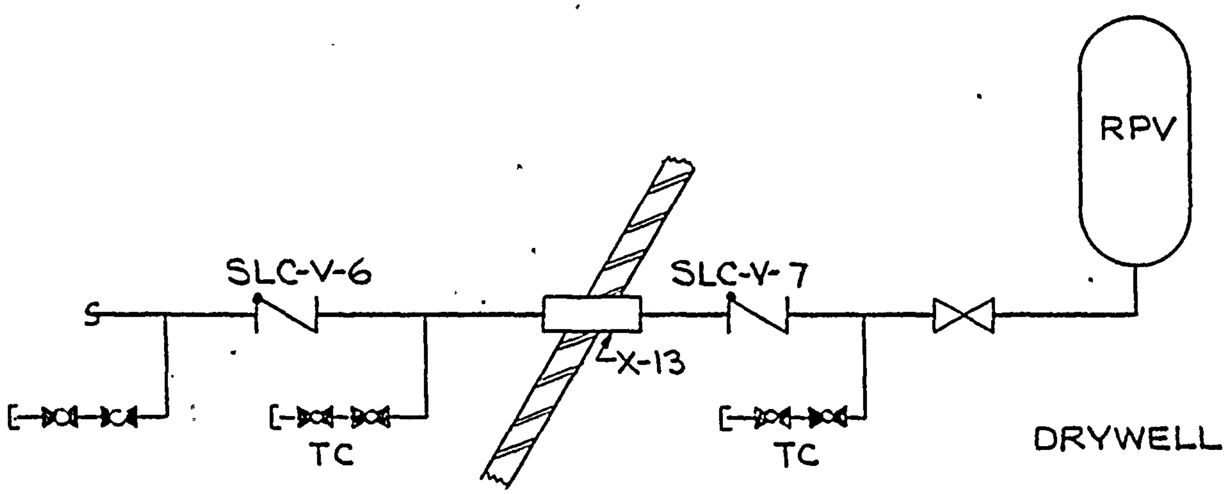
X-12A RHR LOOP A LPCI TO RPV
X-12B RHR LOOP B LPCI TO RPV
X-12C RHR LOOP C LPCI TO RPV
X-6 HPCS TO RPV
X-8 LPCS TO RPV

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	FIGUR 6.2-31



NOTE: SEE NOTE 1 ON FIG. 6.2-31a

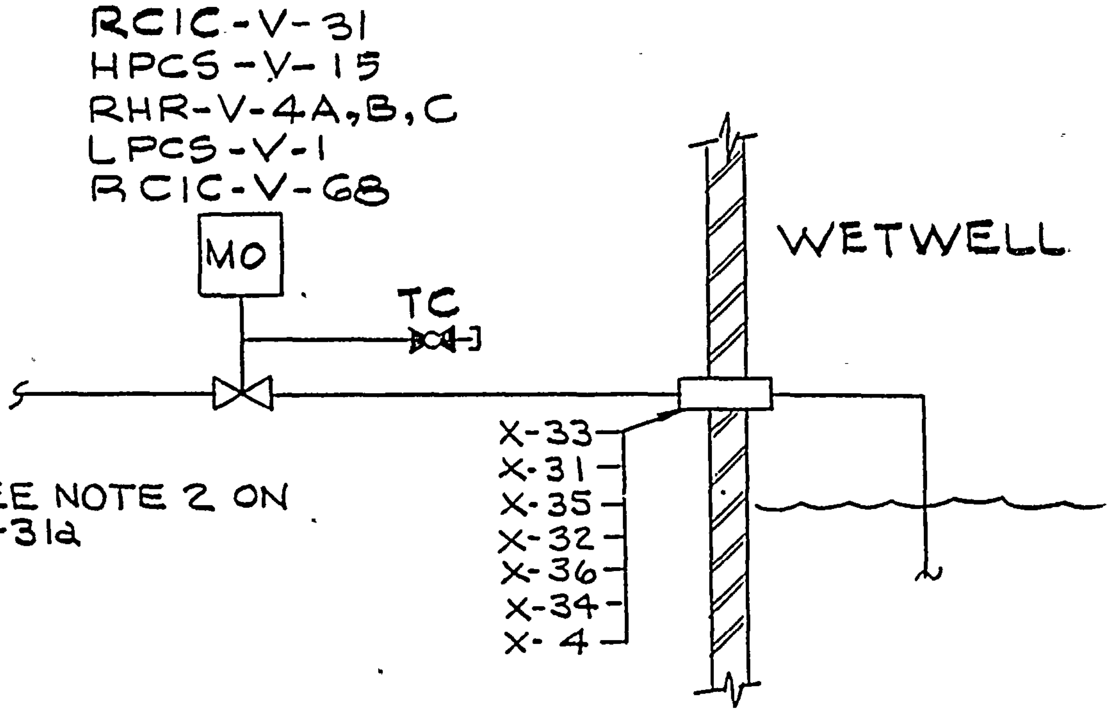
RHR SHUTDOWN COOLING RETURN



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 POO

X-31 HPCS PUMP SUCTION FROM SUPPRESSION POO

X-35 RHR "A" PUMP SUCTION FROM SUPPRESSION POO

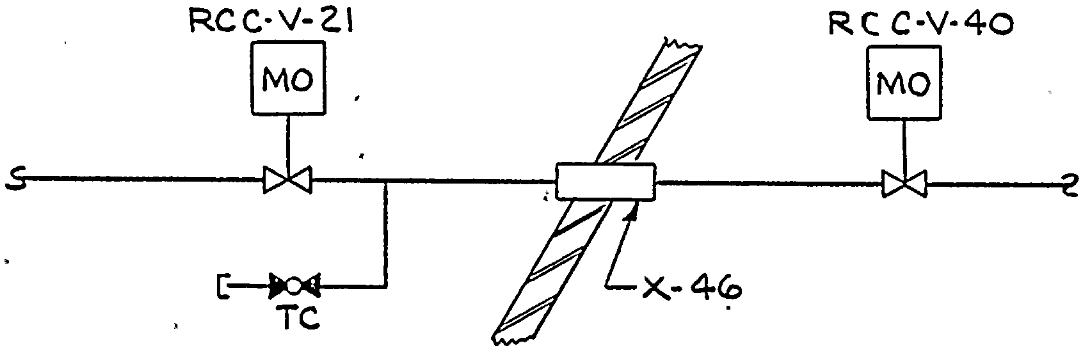
X-32 RHR "B" PUMP SUCTION FROM SUPPRESSION POOL

X-36 RHR "C" PUMP SUCTION FROM SUPPRESSION POOL

X-34 LPCS PUMP SUCTION FROM SUPPRESSION POOL

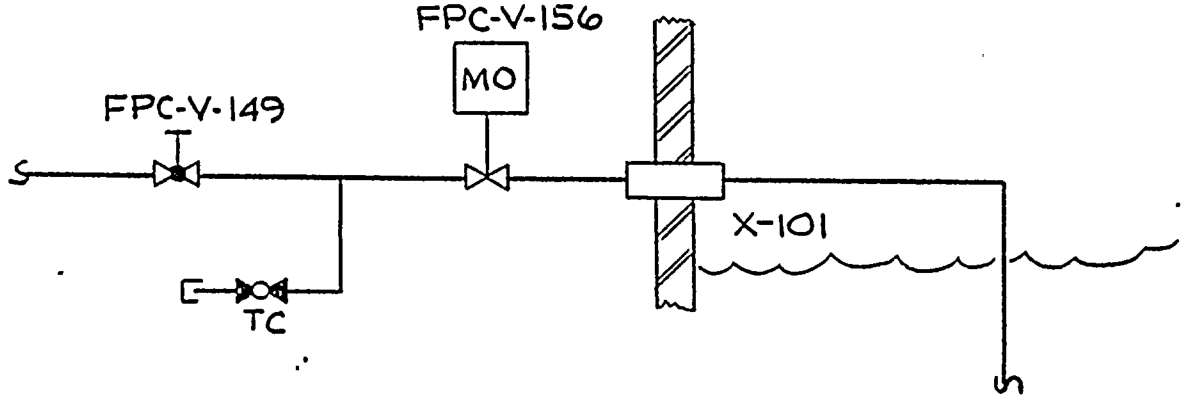
X-4 RCIC TURBINE EXHAUST

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	FI 6.2-31r



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

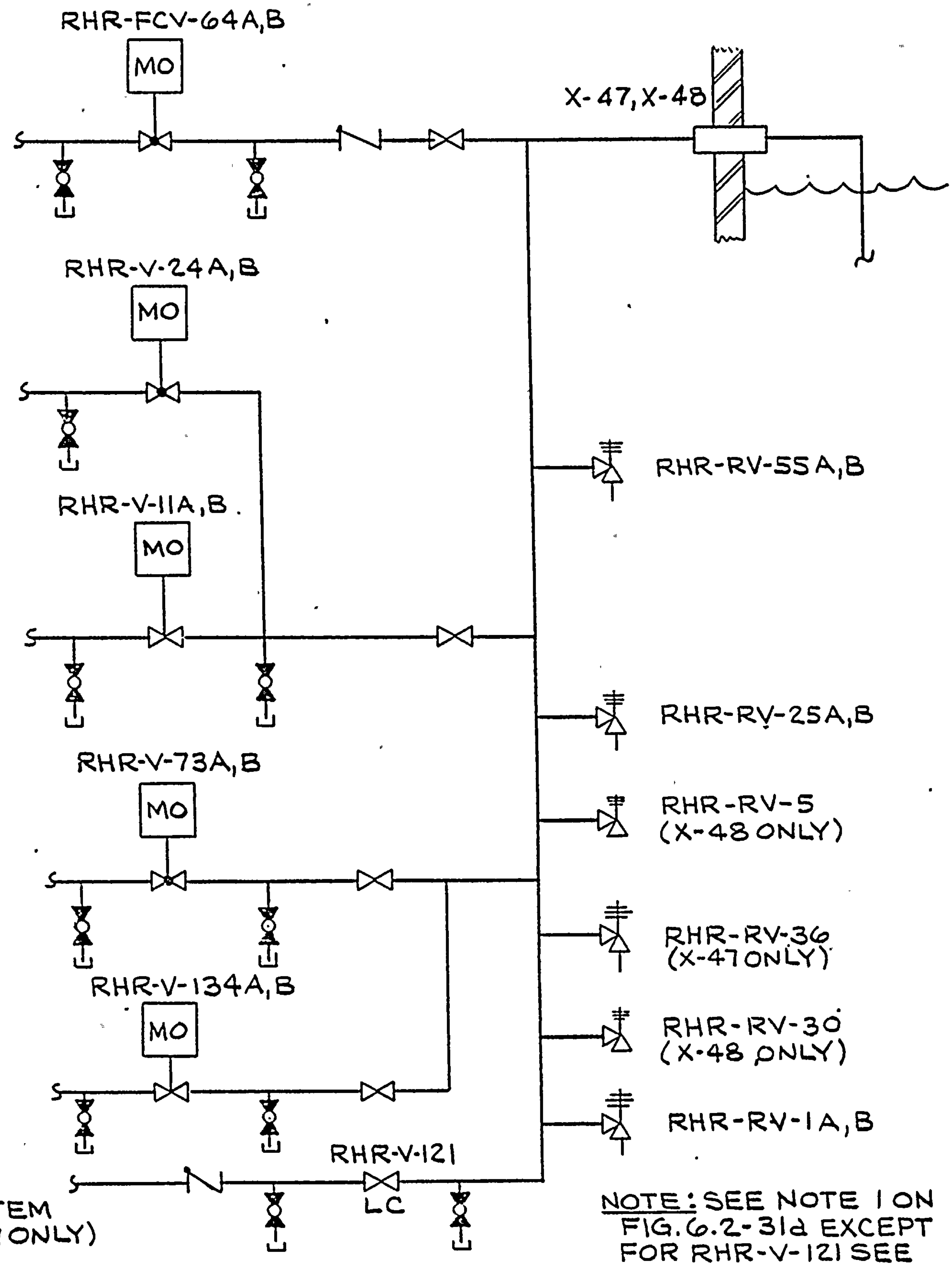
RCC RETURN LINE



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

SUPPRESSION POOL CLEAN-UP RETURN LINE

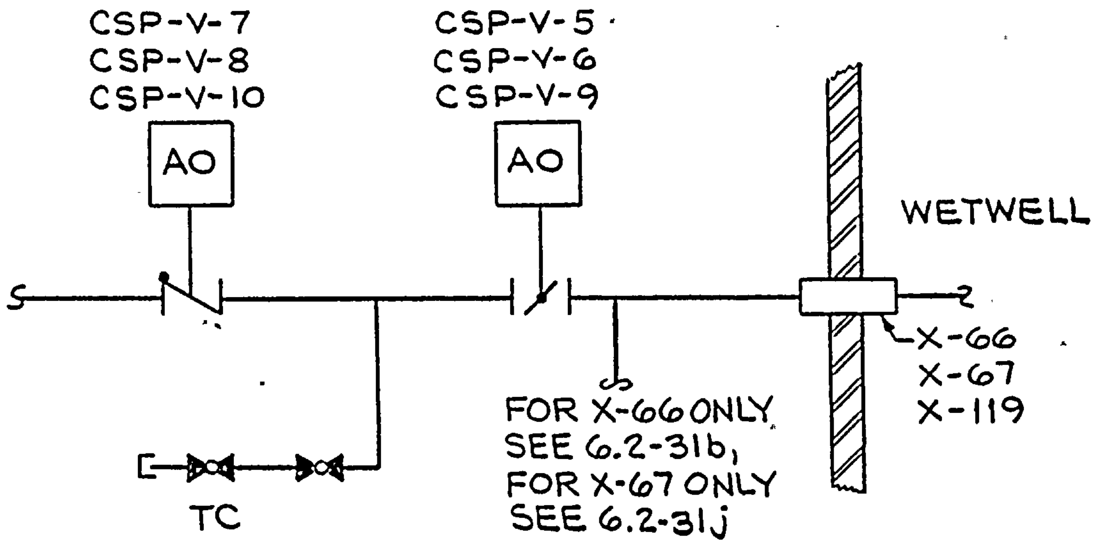
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2.	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-46 & X-101	FIGURE 6.2-31c



NOTE: SEE NOTE 1 ON FIG. 6.2-31a EXCEPT FOR RHR-V-121 SEE NOTE 5.

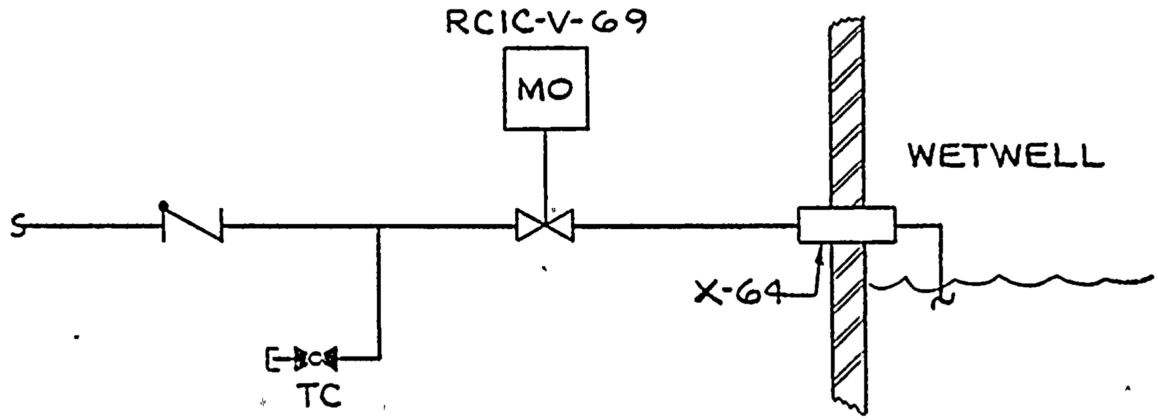
RHR COMBINED RETURN LINE TO SUPPRESSION POOL

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2.	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-47 AND X-48	FIGURE 6.2-31p



NOTE: SEE NOTE 4 ON FIG. 6.2-31a

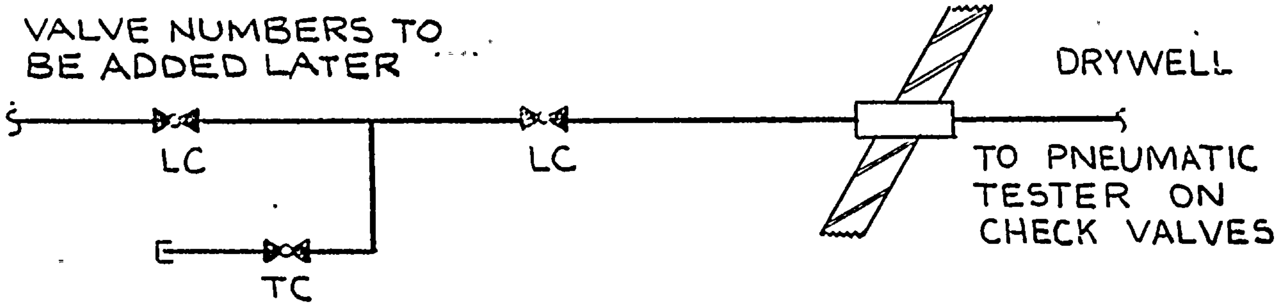
REACTOR BUILDING TO WETWELL VACUUM RELIEF



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 6.2-31g



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

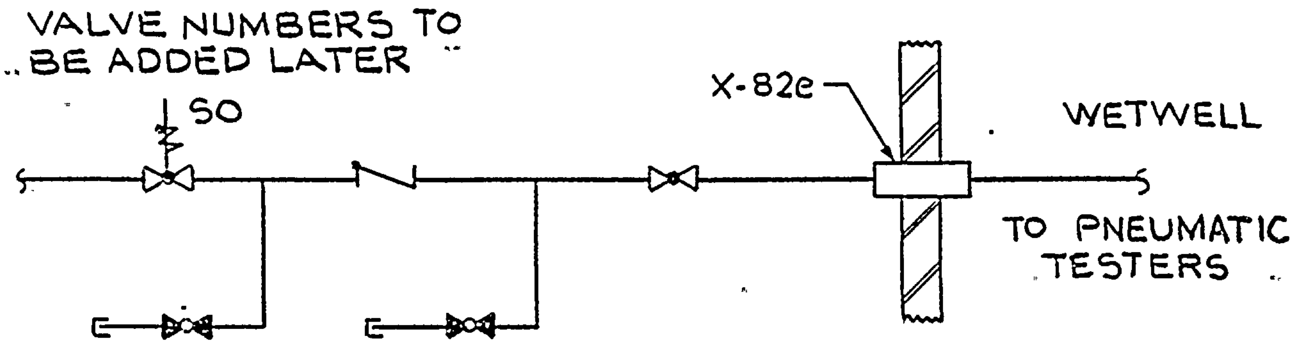
X-61f AIR LINE FOR TESTING RHR-V-41A

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

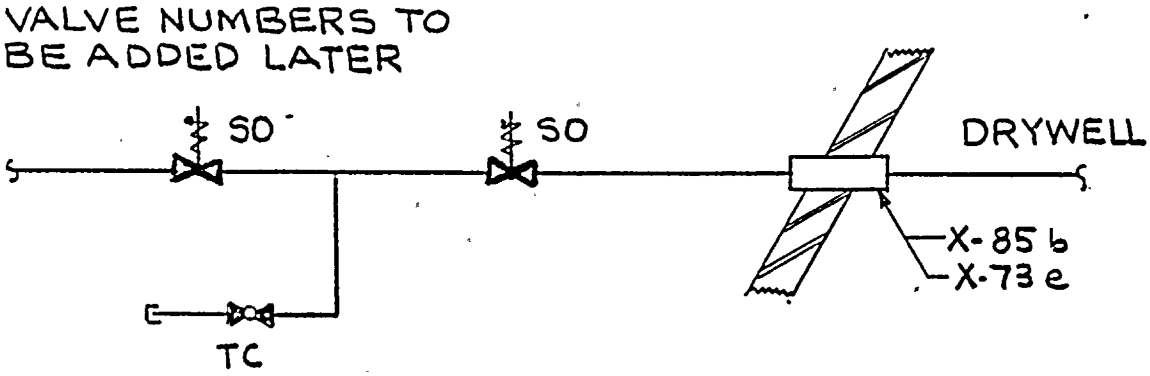
X-78e AIR LINE FOR TESTING HPCS-V-5



NOTE: SEE NOTE 1 ON FIGURE 6.2-31a

AIR LINE FOR TESTING WETWELL TO DRYWELL VACUUM BREAKERS

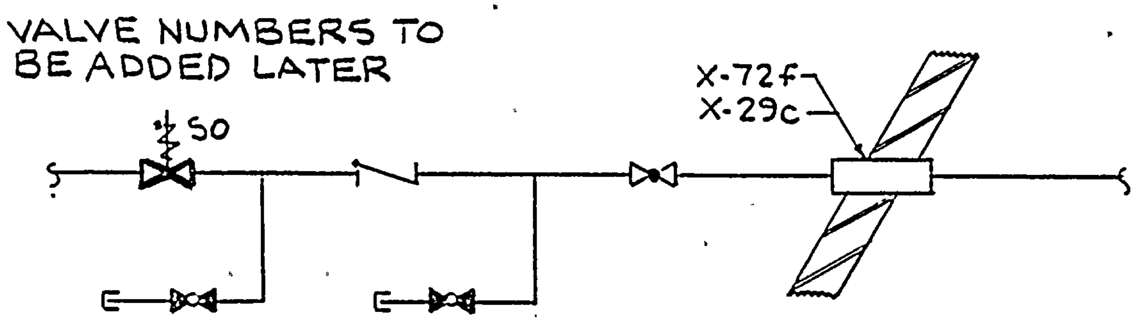
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-42d, 54Aa, 54Bf, 61f, 62f, 69c, 78d, 78e, & 82e	FIGURE 6.2-31r



NOTE: SEE NOTE 4 ON FIGURE 6.2-31a

RADIATION MONITOR SUPPLY LINE DIVISION A

RADIATION MONITOR SUPPLY LINE DIVISION B

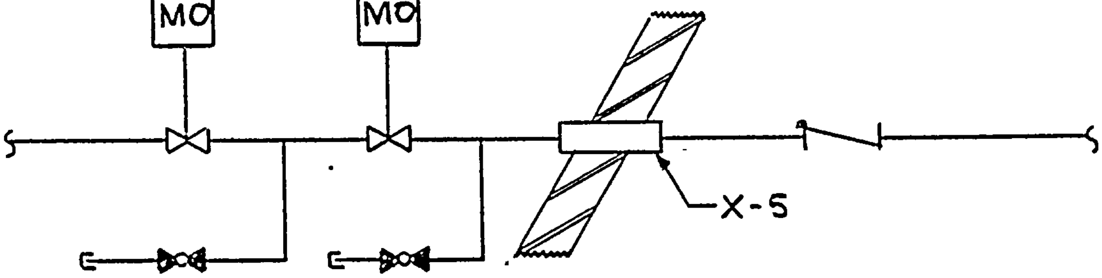


NOTE: SEE NOTE 1 ON FIGURE 6.2-31a

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

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATION X-5	FIGURE 6.2-31t

Q. 22.12

For those isolation systems which satisfy the requirements of Criterion 57 of the Design Criteria, provide adequate information to demonstrate that these systems meet the requirements of a closed system as stated in Section 8.9 of BTP-CSB 6-3.

RESPONSE:

The only lines presently listed as falling under GDC 57 (see 6.2.4.3.2.3) are the hydraulic control lines for the RRC flow control valve. Note 28 of the revised Table 6.2-16 evaluates these lines against GDC-57. See response to question 22.027 for revised Table 6.2-16.

RESPONSES TO CSB QUESTIONS*

*per letter, NRC to WPPSS, Sept. 18, 1978.

Q. 022.013
(6.2.1)

Provide the following additional information with respect to the secondary containment:

a. Appropriate plant elevation and section drawings for those structures and areas that will be maintained at negative pressure following a postulated loss-of-coolant accident and that were considered in the dose calculation model.

b. Your proposed technical specification limit for leakage which may bypass the filters of the standby gas treatment system (e.g., valve leakage and guard pipe leakage); and

c. A discussion of the testing methods which will be used to verify that the systems provided are capable of reducing and maintaining the pressure within all secondary containment volumes to a negative pressure of 0.25 inches (water guage).

RESPONSE:

Information on the secondary containment is provided in 6.2.3*; secondary containment functional design. Specifically,

a. The reactor building is the secondary containment, refer to Figures 1.2-3 thru 1.2-7 and 3.8-1 and -2.

b. Refer to response to NRC question 022.6. The isolation valves on the lines which have been identified as potential secondary containment bypass paths will be tested to ensure the total bypass leakage rate is equal to or less than .4 scfh as outlined in 6.2.3.3.

c. Refer to 6.2.3.4 and 7.3.1.1.7. The reactor building contains sufficient openings to maintain uniform pressure throughout the building, refer to previously listed figures.

*6.2.3 will be changed as per the attached draft.

rate of one psi per second. The effects of the design basis tornado pressures on the structure are discussed in 3.3, and tornado generated missiles are discussed in 3.5.

g. The reactor building is designed for all probable combinations of the design basis wind and the design basis tornado velocities and associated differences of pressure within the structure and atmospheric pressure outside the structure. For structural design purposes the various pressures considered acting within the secondary containment structure include:

1) A negative internal pressure of 0.25 inches of water under which the secondary containment normally operates.

2) A negative internal pressure of 0.012 psig, relative to the outside atmosphere, which exceeds the 0.25 inches of water in g(1), to account for any uncertainty in pressure measurement and to account for any negative pressures actually developed by the standby gas treatment system or by unknown causes.

3) A positive pressure of 0.25 psig relative to the outside atmosphere, to account for any positive pressure transient which the secondary containment may experience following a postulated pipe break in the secondary containment, to account for any outward positive differential pressures created by wind loads, and to account for any uncertainty in pressure measurements.

6.2.3.2 System Design

Refer to Figures 1.2-3 through 1.2-7 ~~1.2-3, 1.2-4, 1.2-5 and 1.2-9~~ for general arrangement drawings of the reactor building showing plan and elevation views of the boundary of the structure. Also refer to Figures 3.8-1 and 3.8-2. Refer to Table 6.2-12, Secondary Containment Design and Performance Data, for the design and performance data for the secondary containment structure.

Q. 22.014

Note 2 of Table 6.2-16 states that the suppression pool serves as an isolation barrier to the environment. It is our position that this is unacceptable.

RESPONSE:

The penetrations that reference note number 2 in Table 6.2-16 meet General Design Criteria 56 and include a single containment isolation valve and are closed systems outside containment. This provides double barrier containment isolation and the water in the suppression chamber is not considered as part of containment isolation. The closed systems outside containment are protected from missiles, are Seismic Category 1, Safety Class 2, and have a design temperature and pressure at least equal to that of the containment.

Note number 2 of Table 6.2-16 is being deleted. Table 6.2-16 will be revised to include the General Design Criteria for all penetrations. (See response to question 22.027).

Q. 22.15

Provide a detailed drawing showing the locations of the containment spray headers relative to the internal structures.

RESPONSE:

The containment spray system is discussed in 6.5.2.* The suppression chamber spray header is shown in Figure 3.5-3. The lower drywell spray header is shown in Figures 3.5-4, 3.5-21, 3.5-22, 3.5-25, 3.5-26, 3.5-27 and 3.5-28. The upper drywell spray header is shown in Figures 3.5-16, 3.5-17, and 3.5-18.

*See draft on following page

6.5.2 CONTAINMENT SPRAY SYSTEM

6.5.2.1 Design Bases

The containment spray system is capable of quickly reducing containment pressure during the post-accident period of a LOCA through condensation of steam in the drywell and through cooling of the non-condensable gases in the free volume above the suppression pool. Containment spray is not required to prevent overpressurization of the containment (see 6.2.1). The containment spray system is not used for fission product removal from the containment atmosphere.

6.5.2.2 System Design

The drywell spray consists of two independent loops and spray headers. The suppression chamber spray consists of one spray header supplied from two otherwise independent loops. Since the water source for all containment spray is the suppression pool, the spray system is a closed loop cooled by the RHR heat exchangers. The rated flows for drywell and suppression chamber sprays are 7450 gpm/loop and 450 gpm/loop, respectively. Containment spray is a subsystem of the RHR System (5.4.7).

The drywell spray valves are electrically interlocked to allow actuation of the drywell spray only when there is a high drywell pressure signal present. After a high drywell pressure signal is present, a second electrical interlock prevents actuation of either the drywell or the suppression chamber spray lines until the corresponding LPCI injection valve is shut.

A procedural restriction prohibits the operators during the first ten minutes following a LOCA from closing a LPCI injection valve and interrupting core cooling (Refer to 6.2.2.2). Containment spray must be initiated and secured by operator action. Hydrogen mixing which results from containment spray operation is discussed in 6.2.5.

The ~~wetwell~~ suppression chamber spray header location is shown in Figure 3.5-3. The lower drywell spray header location is shown in Figures 3.5-4, 3.5-21, 3.5-23, and 3.5-25 through 3.5-28. The upper drywell spray header is shown in Figures 3.5-16 through 3.5-18.

Q 22.016

Provide the value of the external design pressure for the containment structure.

RESPONSE:

The primary containment is designed for a total external pressure of 4 psid; however, since the compressed insulation between the concrete biological shield and the containment exerts a uniform 2 psid external pressure - half of the total external pressure differential allowed - the reactor building pressure may be no greater than 2 psi above the primary containment pressure. This value is given in Table 6.2-1, "Containment Design Parameters", and Section 6.2.1.1.2.

Q. 22.017

Provide the assumptions and initial conditions for the activation of both drywell spray loops following a postulated loss of coolant accident (LOCA) that has purged all the drywell noncondensable gases into the suppression chamber; provide the same information for the inadvertent drywell spray activation at normal operation conditions. This information is requested so that we may perform an independent evaluation of the reverse differential pressure across the drywell floor to establish the degree of conservatism in the small steam line break analysis.

RESPONSE:

The assumptions used for activation of both drywell spray loops following a small steam line break are discussed in 6.2.1.1.4. Initial conditions are given in Table 6.2-19*, Initial Conditions Employed in Negative Pressure Design Evaluation, and Table 6.2-1, Containment Design Parameters.

The same information should be used for inadvertent activation of drywell spray during normal operating conditions except that assumption d. and e. (in 6.2.1.1.4) are not applicable.

*See next page for draft

Table 6.2-19

Initial Conditions Employed in Negative Pressure Design Evaluation

A. Containment preincident conditions used for sizing internal vacuum breakers (wetwell to drywell)

	Drywell (DW)	Suppression Chamber (WW)
1. Pressure, psig	.75	.75
2. Temperature, ^oF	135	50
3. Relative Humidity,%	20	100

B. Containment preincident conditions used for sizing external vacuum breakers (reactor building to wetwell)

	Drywell (DW)	Suppression Chamber (WW)
1. Pressure, psig	0	0
2. Temperature, ^oF	150	50
3. Relative Humidity, %	30	100

6.2.1.1.4 Negative Pressure Design Evaluation

The limiting transient for the wetwell-to-drywell (WW-DW) and reactor building-to-wetwell (RB-WW) vacuum breaker systems is simultaneous operation of both drywell spray loops after a small LOCA. This transient has been determined by analysis to be more severe than any of the following:

a. ECCS flow from a recirculation line break

b. One drywell and one wetwell spray line activation following a small LOCA

c. Inadvertant operation of one drywell spray line during normal operating conditions

The analysis performed for the case of simultaneous operation of both drywell loops after a small LOCA made the following conservative assumptions:

a. Drywell spray flow of 8400 gpm from each loop. This flow corresponds to runout flow for the RHR pumps.

b. 100% spray efficiency

c. $50^{o}F$ spray temperature

d. All non-condensible gases are purged into the wetwell as a result of the LOCA.

e. The drywell is full of steam at a pressure corresponding to wetwell pressure plus the hydrostatic head corresponding to the downcomer submergence.

Drywell spray is not required to maintain the primary containment below design pressure nor is it required for containment cooling. If following a small LOCA all the noncondensible gases are purged into the wetwell, actuation of one of the two drywell sprays will rapidly depressurize the drywell. Actuation of the second drywell spray with the drywell full of steam is neither necessary nor desirable and represents the limiting transient for the vacuum breaker systems.

The initial conditions used in the analysis are provided in Table 6.2-19.

Q. 022.018 (6.2.1) — We require your compliance with our proposed position on containment steam bypass for small breaks. The details are contained in the Containment Systems Branch Technical Position, "Steam Bypass for Mark II Containments," a copy of which is enclosed.

Q. 031.070 (RSP) (6.2.2.2) (6.5.2.2) (7.3.1.1) (031.001) (031.011) — It is the staff's position that insufficient time is available for the operator to reliably take the manual actions which are necessary to initiate suppression pool spray during a small break. The staff has established the requirement for automatic initiation of suppression pool spray for the Mark II containment. Accordingly, we require you to provide a Class IE automatic control system for each suppression pool spray system.

Response: The WNP-2 design meets the intent of the proposed CSB Branch Technical Position on "Steam Bypass for Mark II Containments".

The history of the question of steam bypass on WNP-2 is extensive, dating back to January 1972. Questions 5.4, 5.22, and 5.24 to the PSAR all respond to the concern. The SER (pp 63-65) summarized the NRC position on the issue at the CP stage and noted that WPPSS agreed to study additional means to mitigate the consequences or minimize the potential for bypass leakage. This was formally documented as a Post CP item in the notes of a NRC-WPPSS meeting held on October 17-18, 1973 (Reference 1). In the notes WPPSS committed to submitting a report on the matter. In August 1974, Reference 2 transmitted the WPPSS report WPPSS-74-2-R5, "Drywell to Wetwell Leakage Study", satisfying the commitment. The NRC requested additional information concerning the report in Reference (3). References (4) and (5) provided WPPSS responses to the NRC questions. Reference (6) indicated that Structural Engineering Branch found the applicable WPPSS responses acceptable. WPPSS has no record of feedback from Containment

Systems Branch on the responses to its questions but assumed in Reference (5) that, in the absence of feedback, the post CP item was resolved. Accordingly, WPPSS has gone ahead with construction in these areas based on the above correspondence.

A point by point discussion summarizing the WPPSS design capabilities to mitigate Bypass Leakage problems based on the above correspondence and with respect to the proposed Branch Technical Position is given below:

1. NRC Proposed Requirement: Allowable bypass capability on the order of 0.05 ft^2 $(A/\sqrt{K})$

   Response: As documented in reference 5 and the FSAR, the maximum allowable bypass leakage capacity is $A/\sqrt{K}=.028$ ft^2 using conservative calculational techniques and assumptions.* WPPSS, therefore, believes the existing calculations meet the intent of $A/\sqrt{K} = 0.05$ ft^2.

2. NRC Proposed Requirement: An automatic system should be provided to initiate automatic wetwell sprays. The system should meet the standards of an Engineering Safety Feature including redundancy and diversity and be actuated automatically ten minutes following a LOCA. If the RHR system is used for this purpose, it must be analyzed to assure no degradation of its ECCS function.

   Response: WPPSS asserts that manual initiation is sufficient since the drywell floor will be routinely tested and evaluated against a Tech Spec limit of $A/\sqrt{K} = 0.0045$ ft^2, a level at which no operator action is required for the spectrum of small break sizes. (Reference 5 - see #3 below for testing details).

*The FSAR currently lists the capability as 0.026 ft^2. This is from a GE analysis and the FSAR is being amended to reflect the latest calculations (see attached draft change).

The construction, design, quality control, and surveillance requirements on the drywell floor give it the same level of safety as the containment itself. Reference 4 and Part VI of reference 2 showed that through-wall cracks will not develop through the concrete slab under postulated design conditions including the SSE and that leakage in excess of that accounted for due to permeability would not be possible. Reference 6 indicated the NRC Structural Engineering Branch's acceptance of these responses. Accordingly, WPPSS sees no reason to assume that an $A/\sqrt{K}$ of .0045 ft^2 is exceeded any more than there would be reason to assume the design containment leak rate of .5% per day is exceeded. Calculations documented in Reference 5 using the CONTEMPT - LT computer code were used in computing the maximum allowable leakage rate of $A/\sqrt{K}$ = .028 ft^2, six times the Tech Spec limit. In the calculation over 167 minutes was available for operator action before drywell design pressure was exceeded. Accordingly, a requirement that an automatic system be provided is unnecessary.

3. <u>NRC Proposed Requirement</u>: A single preoperational high pressure leakage test should be performed and periodic low pressure tests at each refueling outage with an acceptance criterion of 10% of the bypass capability at the test pressure.

   <u>Response:</u> The intent of this proposed requirement has been committed to by WPPSS. A single preoperational leakage test will be conducted with the downcomers capped at 15 psid and 25 psid (the design drywell to wetwell differential pressure). At each refueling outage a low pressure operational test will be performed as a Tech Spec Surveillance Requirement to verify .0045 ft^2. Details of the nature of this test are discussed in question 5.22 to the PSAR but will be summarized here since the specific numbers have been since updated.

Routine Leak Testing and Inspection: During entry to the drywell at each refueling outage, accessible drywell to wetwell barrier surfaces will be visually inspected to ascertain any possible leak paths. Vacuum relief valves will be visually inspected to insure they are clear of foreign material. At each refueling outage, before the primary system is pressurized, after all these containment inspections are complete, and after the vacuum breakers are exercised, the following test shall be carried out:

> The drywell will be pressurized to at least 1.0 psi above the wetwell. After an adequate stabilization period, the drywell to wetwell leakage rate will be measured. The acceptance criterion will correspond to an equivalent leakage capacity ($A/\sqrt{K}$) of 0.0045 ft^2, which is 16% of the allowable leakage. If a greater leakage rate is found, the containment shall be entered and the cause determined and corrected and the test repeated.

4. <u>NRC Proposed Requirement</u>: Vacuum relief valves should have redundant position indicators with indication and redundant alarms in the control room. The vacuum breakers should be operability tested at monthly intervals to assure free movement.

<u>Response:</u> WPPSS meets this requirement with the current design. Each vacuum breaker penetration consists of two discs in series, each disc with redundant position indication which display in the Control Room. Each vacuum breaker disc will be equiped with an exercising mechanism and each disc will be exercised at a frequency equivalent to the testing of ECCS valves.

References

1. Letter, WR Butler, NRC, to JJ Stein, WPPSS, "Meeting Summary October 17-18, 1973, dated November 26, 1973.

2. Letter, WPPSS to NRC, G02-74-17, dated August 9, 1974.

3. Letter, NRC to WPPSS, dated January 14, 1975.

4. Letter, WPPSS to NRC, G02-75-52, dated February 25, 1975.

5. Letter, WPPSS to NRC, G02-76-156, dated April 23, 1976.

6. Letter, NRC to WPPSS, dated May 15, 1975.

FSAR Change to Section 6.2.1.1.5.4
connected with SCN 78-25
and
NRC Questions 22.018 and 31.070

a. Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of a mixture of liquid and vapor, the total leakage mass flowrate is higher but the steam flowrate is less than for the case of pure steam leakage. Since only the steam entering the suppression chamber free space results in the additional containment pressurization, this is a conservative assumption.

b. There is no condensation of the leakage flow on either the suppression pool surface or the containment and vent system structures. Since condensation acts to reduce the suppression chamber pressure, this is a conservative assumption. For an actual containment there will be condensation, especially for the larger primary system break where vigorous agitation at the pool surface will occur during blowdown.

6.2.1.1.5.4 Analytical Results

The containment has been analyzed to determine the allowable leakage between the drywell and suppression chamber. Figure 6.2-17a shows the allowable leakage capacity ($A/\sqrt{K}$) as a function of primary system break area. A is the area of the leakage flow path and K is the total geometric loss coefficient associated with the leakage flow path.

Figure 6.2-17a is a composite of two curves. If the break area is greater than approximately 0.4 square feet, natural reactor depressurization will rapidly terminate the transient. For break areas less than 0.4 square feet, however, continued reactor blowdown limits the allowable leakage to small values. The maximum allowable leakage capacity is at $A/\sqrt{K}$ = .026 square feet. Since a typical geometric loss factor would be three or greater, the maximum allowable flow path would be about .052 square feet. This corresponds to a 3 inch line size.

[INSERT]

6.2.1.1.6 Suppression Pool Dynamic Loads

A generic discussion of the suppression pool dynamic loads and asymmetric loading conditions is given in Mark II Dynamic Forcing Function Information Report, Reference 6.2-4. A unique plant assessment of these dynamic loads is made in WNP-2 Design Assessment Report, Reference 6.2-5.

6.2.1.1.7 Asymmetric Loading Conditions

See comment in 6.2.1.1.6.

(Insert for page 6.2-30)

Burns and Roe, Inc. confirmed the results of the above analysis by GE in reference 6.2-7. Further investigation into the transient nature of the problem was then undertaken at the request of the NRC.

A transient analysis using the CONTEMPT-LT(Ref. 6.2-8) computer code was performed. The code was modified to include the mass and energy transfer to the suppression pool from relief valve discharge. The limiting case was a very small reactor system break which would not automatically result in reactor depressurization. For this limiting case, it was assumed that the response of the plant operators was to shut the reactor down in an orderly manner at 100^{o}F/hr cooldown rate. No other operator action was accounted for. Heat sinks considered were such items as major support steel inside containment, the reactor pedestal, the diaphragm floor and support columns and the steel and concrete of the primary containment. Based on this analysis, the allowable bypass leakage ($A/\sqrt{K}$) was 0.028 ft^2. The drywell pressure transient is shown in Figure 6.2-17b along with the corresponding curves of wetwell pressure, wetwell temperature and suppression pool temperature.

The allowable bypass leakage of 0.028 ft^2 is well above the maximum possible containment bypass leakage. Periodic testing will be performed to confirm that the containment bypass leakage does not exceed $A/\sqrt{K} = 0.0045$ ft^2. Figure 6.2-17c presents the resulting containment transient for $A/\sqrt{K} = 0.0045$ ft^2. The peak containment pressure shown in Figure 6.2-17c is well below the containment design pressure.

6.2.7 REFERENCES

6.2-1 James, A. J., "The General Electric Pressure Suppression Containment Analytical Model", April 1971, (NEDO-10320).

6.2-2 James, A. J., "The General Electric Pressure Suppression Containment Analytical Model", Supplement 1, May 1971 (NEDO-10320).

6.2-3 Moody, F. J., "Maximum Two-Phase Vessel Blowdown from Pipes", Topical Report APED-4824, General Electric Company, 1965.

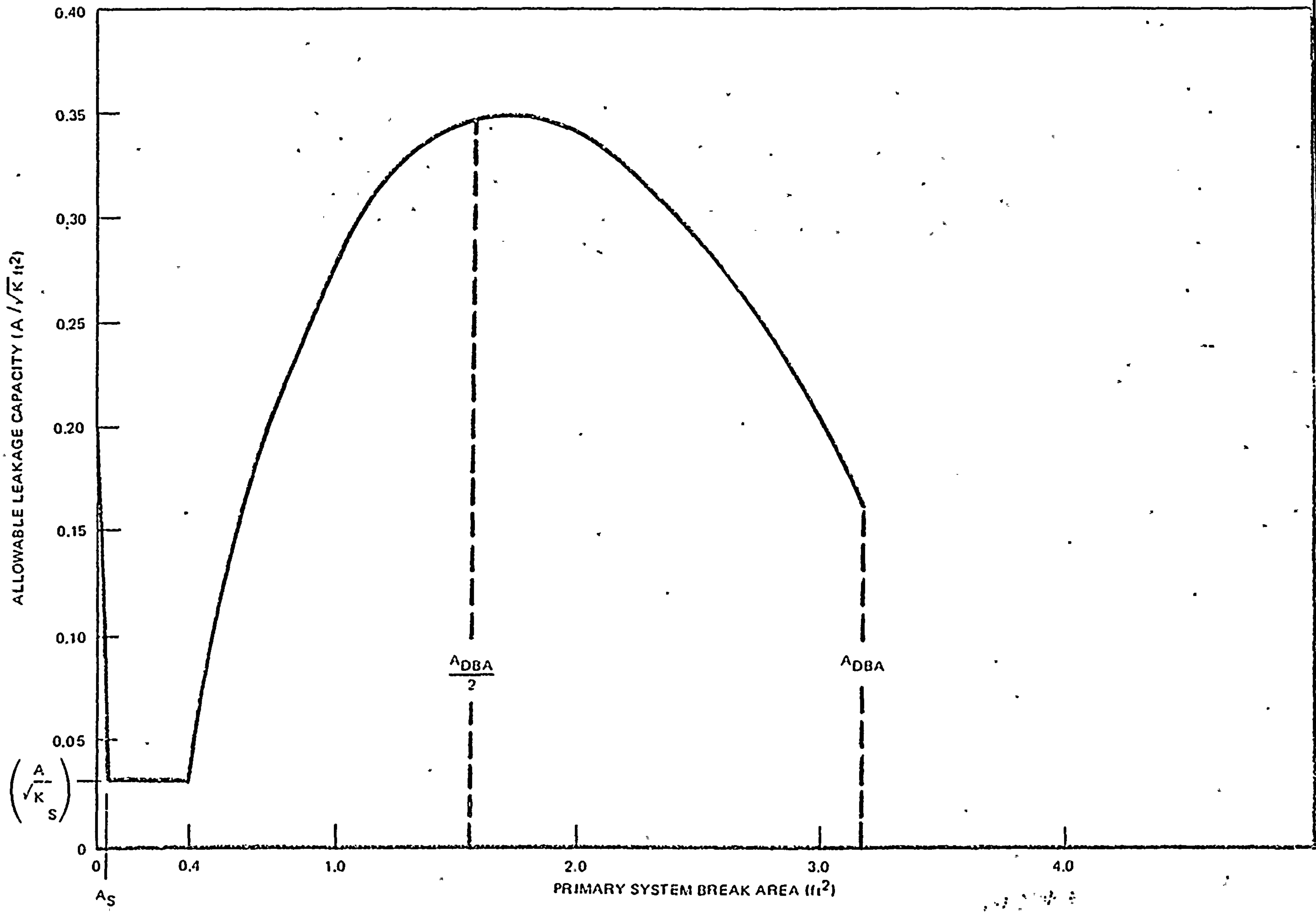
6.2-4 "MK II Containment Dynamic Forcing Functions Information Report (Revision 2)", General Electric and Sargeant and Lundy, NEDO-21061, September 1976.

6.2-5 "Plant Design Assessment Report for SRV and LOCA Loads (Revision 1)", Washington Public Power Supply System, February 1978.

6.2-6 J. D. Duncan and J. E. Leonard, "Emergency Cooling in BWR's Under Simulated Loss-of-Coolant (BWR PLECMP) Final Report," FEAP-13197, General Electric, June 1971.

6.2-7 WPPSS REPORT, "Drywell to Wetwell Leakage Study", WPPSS-74-2-RS, July, 1974. (Submitted to NRC by WPPSS to NRC, Ltr. G02-74-17, dated Aug. 9, 1974).

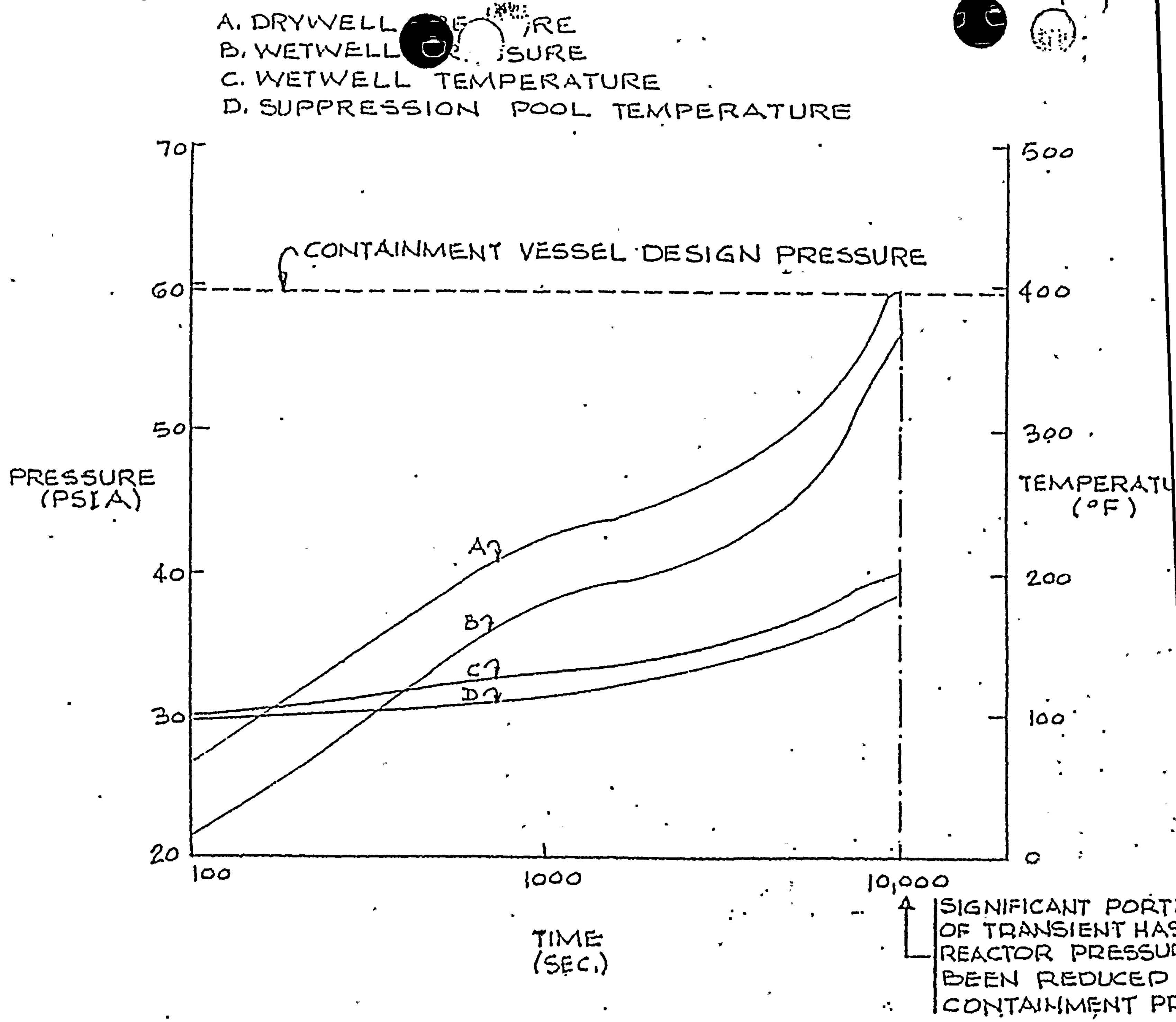
6.2-8 Wheat, L. L.; Wagner, R. J.; Niederauer, G. F.; Obenchain, C. F., CONTEMPT-LT--A COMPUTER PROGRAM FOR PREDICTING CONTAINMENT PRESSURE-TEMPERATURE RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, ANCR-1219, Aeroject Nuclear Company, June, 1975.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

ALLOWABLE LEAKAGE CAPACITY

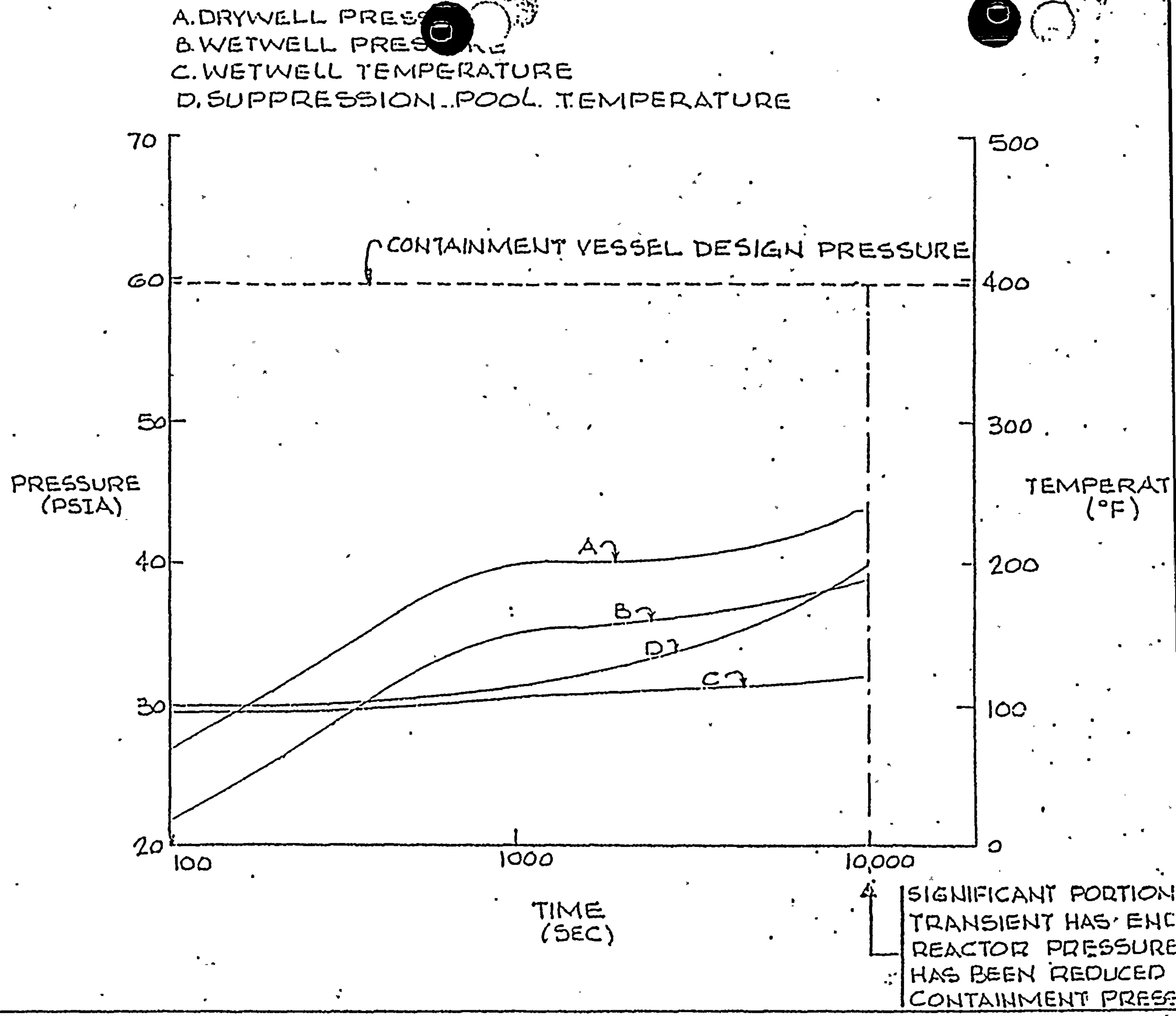
FIGURE 6.2-17



PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTAINMENT TRANSIENT FOR MAXIMUM ALLOWABLE BYPASS CAPACITY, $A/\sqrt{K} = 0.028$ FT^2

FIGURE 6.2-17b



ASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTAINMENT TRANSIENT FOR $A/\sqrt{K} = 0.0045$ FT²

FIGURE 6.2-17c

Q. 22.019
(6.2.1)

Provide the indormation requested in paragraphs B.1.a, B.1.d, B.1.g, B.4 and B.5 of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation".

RESPONSE:

Refer to the response to previous question 022.1. The valves in the containment purge and vent lines are containment isolation valves. The design criteria for the isolation valves is discussed in 6.2.4, Containment Isolation System, and 6.2.6, Containment Leakage Testing. In addition paragraph 6.2.1.1.8.2 has been revised to expand the definition of reactor operation and include information relative to purge isolation valve qualification.*

* See attached draft change

[illegible]

6.2.1.1.8.2 Primary Containment Purging

The primary containment above the drywell floor is provided with a purge system to reduce residual contamination prior to personnel access. This system is designed to produce a purge rate equivalent to 3 air changes per hour of the net free volume.

The drywell is purged once a year during scheduled refueling shutdown period and as required for inspection. The drywell purge rate is 10,500 cfm. Provision is made to automatically route a reduced purge rate of 4400 cfm to the standby gas treatment system if residual airborne contamination is higher than allowable limits for direct release to the atmosphere. Purge air is taken from the reactor building ventilation supply duct through two 30" normally closed isolation valves into the primary containment. Purge air is extracted from the drywell through two 30" normally closed isolation valves and is routed to one of two systems. The discharge can be routed through a normally closed isolation valve to the reactor building exhaust air plenum or to the standby gas treatment system (Figures 3.2-15 and -18). If a high airborne activity occurs, the radiation monitors at the exhaust air plenum would cause the reactor buildings ventilation and primary containment purge systems to isolate.

Provision is also made to purge the suppression chamber section of the primary containment. Purge air is taken from the reactor building supply duct through two 24" normally closed isolation valves into the suppression chamber. Purge air is extracted from the suppression chamber through two 24" normally closed isolation valves and routed to the exhaust air plenum or standby gas treatment system in the same manner as the drywell purge exhaust. The suppression chamber purge rate is 7500 cfm.

The above systems are designed to purge either the drywell or the suppression chamber. Provision is not made to purge both areas at rated flow simultaneously. Only one vent line and one purge line will be open at any one time during reactor operation.

The purge system may be used during reactor operation only for purging the primary containment prior to personnel entry. Purge system operation during reactor operation will be limited to less than 1% of reactor operating time.

including startup, hot standby, and hot shutdown

isolation

All containment purge valves, including the 2" bypass valves, are designed to shut within four seconds of receipt of a containment isolation signal. ~~and to shut against full containment design pressure of 45 psig.~~ The containment isolation signals and the purge valves are part of the containment isolation system which is an ESF system. Each purge line has two isolation valves. These valves are opened by allowing compressed air to oppose a spring in the valve actuator. On a loss of compressed air, loss of electrical signal, or on a containment isolation signal the valve is shut. If the purge system were operating at the time of a LOCA, the system will automatically be secured. The level of the activity released through the purge system before isolation would be limited to the activity present in the coolant prior to the accident since the purge system will be isolated before any postulated fuel failure could occur.

6.2.1.1.8.3 Post - LOCA

The unit coolers are not required after a LOCA since heat removal is then accomplished by the containment cooling system, a subsystem of the RHR system, as described in 6.2.2. Similarly, containment purge is not required following a LOCA. Two 100% redundant hydrogen recombiners are then placed in operation to ensure that the hydrogen buildup does not reach a dangerous level.

Any equipment located inside the primary containment which is required to operate subsequent to a LOCA has been designed to operate in the worst anticipated accident environment for the required period of time (See 3.11).

6.2.1.1.9 Post Accident Monitoring

A description of the post accident monitoring systems is provided in 7.5.

6.2.1.2 Containment Subcompartments

The two areas within the primary containment considered subcompartments are the area within the sacrificial shield wall and the area above the refueling bulkhead plate at elevation 583'.

All potential pipe breaks within the sacrificial shield wall have been evaluated. The information is contained in References 3.8-5, 3.8-6 and 3.8-7. These references have been previously submitted to the NRC.

Two analyses are being performed to ensure the adequacy of the refueling bulkhead and inner refueling bellows at elevation 583'. The first analysis, a break of the RCIC head spray line, will determine the maximum downward loading due to pipe breaks, and the second analysis, a break of the RRC suction line, will determine the maximum upward loading. These analyses will be incorporated into the FSAR by means of an amendment.

All purge isolation valves have been analytically qualified to close against a maximum design differential pressure of 150 psi in

Q 022.020
(6.2.1)

Provide the test results and method of analysis utilized to determine the seismic sloshing loads discussed in 6.2.1.1.3.1 of the FSAR. Provide the basis for the acceptance of these loads.

Response

Please refer to 6.2.1.1.3.1, 3.8.2.4.3 and 3.8.2.4.3.2 for the information requested.*

*Draft FSAR changes attached.

All of the analyses assume that the primary system and containment are initially at the maximum normal operating conditions. References are provided that describe relevant experimental verification of the analytical models used to evaluate the containment system response.

Seismic sloshing effects in the wetwell were evaluated for a SSE and were found to be minimal. The resulting vertical water displacement at the containment wall is less than one foot. The analytical results and method of analysis utilized to determine the seismic sloshing effects in the wetwell are discussed in 3.8.2.4.3.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-1 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2 provides the performance parameters of the related engineered safety feature systems which supplement the design conditions of Table 6.2-1 for containment cooling purposes during post blowdown long term accident operation. Performance parameters given include those applicable to full capacity operation and to those conservatively reduced capacities assumed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions resulting in release of reactor coolant to the containment. These accidents include:

a. An instantaneous guillotine rupture of a recirculation line,

b. An instantaneous guillotine rupture of a main steam line,

c. An intermediate size liquid line rupture, and

d. A small size steam line rupture.

Energy release from these accidents is reported in 6.2.1.3.

With the formulation of an overall mathematical model which provides for the realistic response of the containment system, response spectra and/or time histories are generated at the component interfaces, and at other desired points. These component response spectra and/or time histories are used to perform detailed dynamic analyses of the individual components as previously mentioned.

Effects due to the presence of water in the suppression pool under earthquake excitations are established following the procedures described in reference 3.8-2. The additional loads due to sloshing effect are included as part of the design loads of the primary containment vessel. The analytical results and methods of analysis utilized to determine the seismic sloshing effects in the suppression chamber are discussed in 3.8.2.4.3.2.

The model used for the seismic analysis and the method of seismic analysis to obtain the seismic moments, shears, displacements and floor response spectra are discussed in 3.7. To obtain a detailed stress/strain analysis in local areas, the following additional methods are used.

In the dynamic analysis of the steel primary containment vessel component, a dynamic mathematical model is formulated which incorporates the general structural geometry and all significant boundary conditions present. The design of the numerous penetrations is such that any restraining forces on the steel primary containment vessel which could be developed can be considered as negligible. The effects of rotational inertia and shear deformations are also considered as negligible in the response of the steel containment vessel. In the determination of the seismic response of the steel containment vessel, damping effects are considered. The incorporation of damping into the dynamic analysis is facilitated by the use of viscous (velocity proportional) damping. The various damping values for both the operating basis earthquake and SSE excitations for the steel containment vessel are discussed in 3.7.

The resulting equations of motion for the steel containment vessel are solved by the use of DACSR, a large capcacity computer program discussed in 3.12. The solution algorithm used depends on the analytical method incorporated to evaluate the equations of motion for the system. A complete discussion of the solution technique is provided in 3.7.

The results of the dynamic seismic analysis contain values for maximum translation and rotational displacements and accelerations, moments and shears, as well as response spectra and/or time histories at desired points throughout the steel containment vessel.

These resultant forces are then combined with the various loading conditions as described in 3.8.2.3.12 and in accordance with Sub-Article NE-3131 of Section III of the ASME Code. These combined forces are used in the structural analysis of the various critical areas present within the steel primary containment vessel. By using a response spectra and/or time history the cantilevered personnel locks, as well as any other appurtenance, are dynamically analyzed as previously discussed.

The resulting stress intensities due to the addition of seismic loads to the various loading conditions for the steel primary containment vessel and its appurtenances will be in accordance with the stress intensity limits as specified in Sub-Article NE-3131 of the ASME Code, Section III.

3.8.2.4.3.1 Computer Program Utilized in the Seismic Dynamic Analysis

The seismic dynamic analysis utilizes DACSR, a large capacity computer program discussed in 3.12. The program is capable of generating the required mass and stiffness matrices which are required to represent the mass and stiffness of the actual structure.

The model of the structure and program solutions and output are discussed in 3.7 and in 3.12.

3.8.2.4.3.2 Seismic Dynamic Analysis of Water in Suppression Chamber (Sloshing Effects)

Tests were not performed to arrive at the seismically induced sloshing loads in the suppression chamber. All of the loads are arrived at by calculations. The calculations provide the basis for the acceptance of these loads.

The method of analysis utilized to determine the seismic sloshing loads in the suppression chamber is taken from Chapter 6, Dynamic Pressure on Fluid Containers, and Appendix F, Dynamic Analysis of Fluids in Containers Subjected to Acceleration, both contained in reference 3.8-2.

Two separate analyses were performed, using the formulations given in the above referenced document, as follows:

a. In the first analyses, the entire suppression chamber is taken as a cylindrical rigid tank in plan having a flat bottom as modeled in the above referenced document, in lieu of the actual 2:1 bottom ellipsoidal head, and supported on the foundation mat. In this analysis the reactor pressure vessel (RPV) pedestal is excluded from the model, and the tank is considered as containing only the water to the full depth shown in Figure 3.8-1.

b. In the second analysis, the RPV pedestal is included in the model. To include the pedestal, the suppression chamber is modeled to consist of theoretical rectangular tanks in plan, of the minimum quantity and the maximum size that can be fitted or inscribed adjacent to each other within the annulus formed by the cylindrical wall of the suppression chamber and the concentric cylindrical RPV pedestal. The tanks are each assumed as independent rigid bodies supported on the foundation mat, flat-bottomed and containing water to the full depth shown in Figure 3.8-1.

In both analyses, the structures are subjected to the maximum floor accelerations due to the Safe Shutdown Earthquake (SSE). The acceleration values are obtained from the time history analysis performed for the reactor building given in 3.7.

Both analyses yield water displacements, velocities, and impulsive and convective water pressures on the walls of the suppression chamber and the reactor building foundation mat.

The first analysis, which considers the RPV pedestal excluded from the suppression chamber, yields the maximum impulsive pressures. The second analysis, which considers the RPV included in the suppression chamber, yields the maximum convective pressures. To obtain conservative values for the forces, bending moments and overturning moments on the suppression chamber and foundation mat, the maximum impulsive forces from the first analysis and the maximum convective forces from the second analysis are assumed to occur together.

The following tabulation gives the analytical results obtained for the additional horizontal wall pressures due to SSE. The additional wall pressures are found to be negligible.

Distance below water surface El. 466'-4 3/4" (feet)	Horizontal Wall Pressure due to SSE induced water sloshing (psi)
0	0.30
5	1.56
10	2.57
15	3.30
20	3.74
Below 20	5.84

The maximum vertical displacement (slosh height) and velocity of the oscillating water surface above the undisturbed equilibrium water surface elevation 466'-4 3/4" ~~is~~ are 9.5 inches at the suppression chamber face. This occurs in the second analysis which considers rectangular tanks with the RPV pedestal included.

The period of water oscillation (time required for the water to oscillate one complete cycle) is 6 seconds in the first analysis (circular tank) and 3.5 seconds in the second analysis (rectangular tanks).

The analytical results used for horizontal pressures in the suppression chamber due to the Operating Basis Earthquake are one-half of the values obtained for SSE.

3.8.2.4.4 Protective Coatings

Protective coatings are applied to all exposed steel surfaces of the primary containment vessel. Surfaces embedded in concrete are not coated. Coating systems used on the inside of the primary containment vessel are selected on the basis of their ability to withstand not only normal operating conditions but design basis accident conditions as well. The coating is able to withstand a DBA without being removed from the surface, so that it will not interfere with emergency pumping and spraying systems. The coating systems are subjected to tests designed to determine their radiation resistance, decontaminability, resistance to decontamination chemicals, and resistance to accident conditions. The

and 17.2 inches per second respectively.

TABLE OF CONTENTS (Continued)

Q. 22.021
(6.2.1)

With regard to all safety-related equipment located inside the containment building such as the control rod drive hydraulic system, the reactor vessel supports and all incore instrumentation leads, we require that the environment be maintained within the maximum temperatures and rate of temperature changes for which the equipment is qualified to operate. Indicate whether the reactor building ventilation system (RBVS) is required to assist in maintaining an acceptable temperature range. If it is, provide the following information regarding the RBVS:

a. Justification for not treating this system as an engineered safety feature.

b. The results of an analysis that the RBVS will not be a potential source of missiles; demonstrate that the RBVS meets our pipe whip criteria.

c. A discussion of the operating procedures to be initiated in the event that the RBVS is unavailable.

d. The location of all temperature sensors associated with the operation of the RBVS.

RESPONSE:

As discussed in 6.2.3 and 9.4.2 the RBVS is the ventilation system for the secondary containment and is designed to be automatically shutdown and isolated on receipt of an accident signal. Except for the isolation portion of the RBVS, the system is not an engineered safety feature. All equipment in the reactor building required to bring the reactor to a safe shutdown is either qualified to the accident environment of the secondary containment or enclosed in compartments which are cooled by the reactor building emergency cooling systems (see 3.11.1.1.2, 3.11.4.2, and 9.4.9). Seismic Class I supports are provided for all RBVS and emergency cooling system ducting to preclude damage to any safety related components in the secondary containment during a seismic event.

The primary containment is cooled by the primary containment cooling system as discussed in section 9.4.11. It should be noted that this system is not required for safe shutdown, and is not assumed to be operable for accident analysis or assessment of the primary containment accident environment.

Q. 22.022

Provide a detailed justification for the use of Quality Group C valves for containment isolation. Our position on this matter is that Quality Group B valves should be used.

RESPONSE:

All containment isolation valves are either Quality Group A or B. Table 7.3.13, is confusing in that there is a "class" column which reflected outdated WNP-2 project related terminology and does not pertain to code classification. This table will be deleted from the FSAR and reference will be made to Table 6.2-16. (Reference response to question 22.027).

Q. 022.023
(6.2.5)

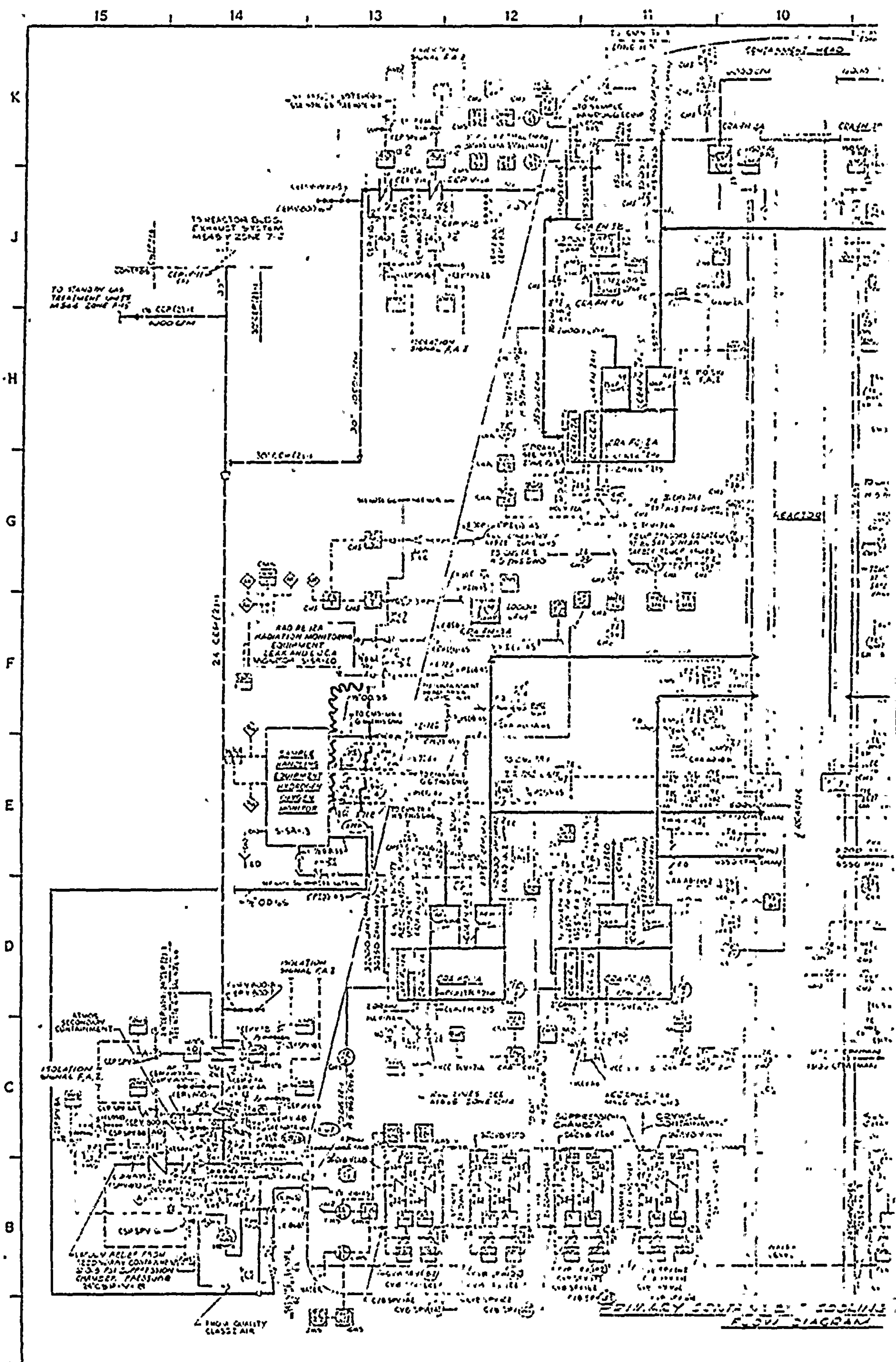
Identify the seismic category and the quality group of the hydrogen monitoring system.

RESPONSE:

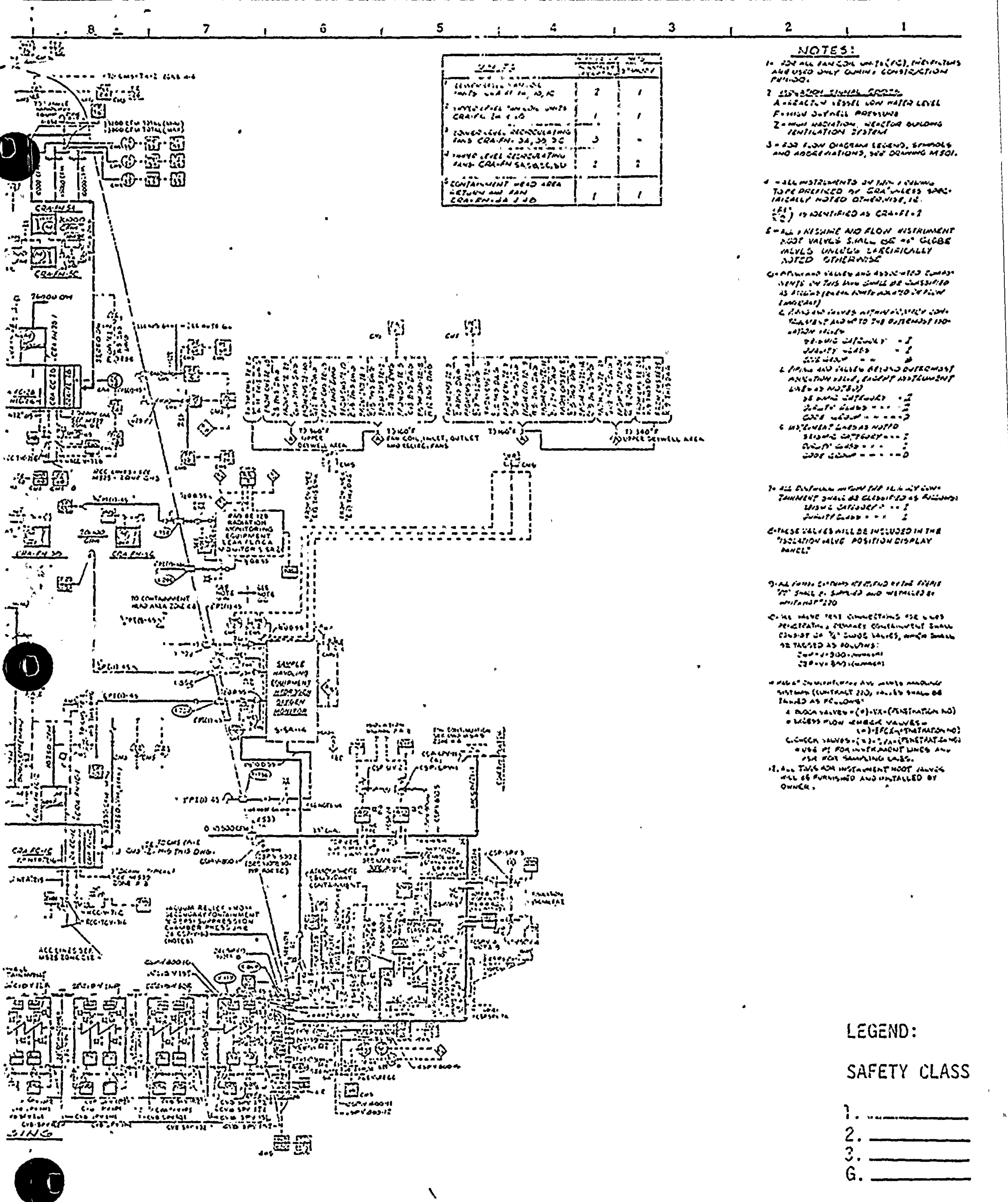
The hydrogen monitoring system is seismic category I. The system is Quality Group B for the suction lines to the downstream side of the excess flow check valves and for the discharge line from the upstream side of the check valve. Between these code breaks, including the monitor, the Quality Group is D.

Refer to Figure 3.2-15*, paragraphs 6.2.5.1 and 6.2.5.2.3, and the response to question 022.8 in Appendix D.

*Figure 3.2-15 will be revised as shown on the attached draft.



FSAR fig. 3.2-15



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM - REACTOR BUILDING PRIMARY CONTAINMENT COOLING & PURGING SYSTEM	FIGURE 3.2-15

Q 22.024
(6.2.5)

Section 6.2.5.2.1 of the FSAR states that all hydrogen mixing will be accomplished by a natural convection process. Provide a detailed analysis that will support your non-mechanical mixing assumption.

RESPONSE:

An assessment of the natural convection process as well as mechanical means of forced convection is in progress. A complete response to this question will be filed prior to April, 1979.

Q. 22.025

Identify the location of the hydrogen sampling points in the drywell and the suppression chamber. Identify the location of the suction and discharge points of the combustion gas control system with respect to local structures and equipment.

RESPONSE:

Please refer to FSAR Table 7.6-12 and Figures 6.2-32 through 6.2-35 for the requested information.*

* See attached drafts.

The system processes the primary containment atmosphere using a blower. The constant speed blower draws a minimum of 155 scfm from the containment. The gas first enters the water scrubber, where particulate matter, droplets and soluble trace impurities are removed from the gas by direct continuous contact with water in a packed bed column. The gas passes upward through the column and leaves the scrubber at the column top through a demister pad, which prevents entrained water from leaving with the gas. The water, with particulates and dissolved solids, leaves the bottom of the scrubber and is directed to the suppression pool.

The gas then enters the blower and is compressed 10 psi to provide flow through the system and connecting piping. The gas then enters the preheater, where it is heated to maintain a thermostatically controlled recombiner inlet temperature between 500°F and 550°F.

The heated and diluted gas enters the catalytic recombiner where the hydrogen and a stoichiometric amount of oxygen react on the catalyst bed to form water vapor. The catalyst bed operates between 550°F and 1130°F and provides essentially 100% conversion efficiency. Inlet temperature greater than 500°F prevents degradation of the catalyst bed from halogens that are present in the feed gas.

The hot recombiner effluent gas is then cooled below 150°F in the aftercooler. The condensate is separated in the moisture separator and is routed to the suppression pool.

A portion of the recombiner discharge is recycled to the blower suction to minimize blower horsepower when containment pressure is equal to 3 psig or greater.

During system operation, the containment atmosphere is drawn from the drywell and the recombiner effluent gas is discharged to the suppression chamber.

PHYSICAL LOCATIONS OF ACTIVE CONTAINMENT ATMOSPHERE CONTROL (CAC) SYSTEM PENETRATIONS INTO THE PRIMARY CONTAINMENT ARE SHOWN ON FIGURES 6.2-~~31, 32, 33 AND 34~~. 32, 33, 34 and 35.

Each hydrogen recombiner is skid mounted into an integral package having maximum dimensions of 11 feet long by 9 feet wide and 9 feet high. All pressure containing equipment including piping between components is considered an extension of the containment and is classified Quality Group B

7.6.1.13.8 Containment Hydrogen and Oxygen

Atmosphere samples from three locations inside the primary containment and one location in the suppression chamber are sequentially monitored for hydrogen and oxygen percentage levels by each of two completely redundant and separate analyzers. For precise locations of each sample point see Table 7.6-12.

Each gas analyzer cabinet contains a hydrogen and an oxygen analyzer with sample conditioning and sample programming means. The programmer also admits standardizing gases periodically to calibrate the analyzers. Vent gases are pumped back to the primary containment at all times.

The analyzers are single range, i.e., 0-10% hydrogen and 0-25% oxygen. System accuracy is ± 2% of full scale. The output signal from each analyzer is sent to individual recorders in the main control room. Each analyzer has two adjustable alarm contacts which annunciate abnormal conditions in the main control room. The analyzers are quality group classification D.

7.6.1.13.9 Suppression Chamber Pressure

Suppression chamber pressure is recorded in the main control room from two separate pressure transmitter systems. Range of recording is from 0-100 psig with an accuracy of ± 2.0% of span.

7.6.1.14 Suppression Pool Temperature Monitoring System Instrumentation and Controls

7.6.1.14.1 System Identification

The suppression pool temperature monitoring (SPTM) system is designed to monitor suppression pool water temperature and alert the plant operator to the potentially hazardous condition of elevated pool water temperature with subsequent severe structure vibration.

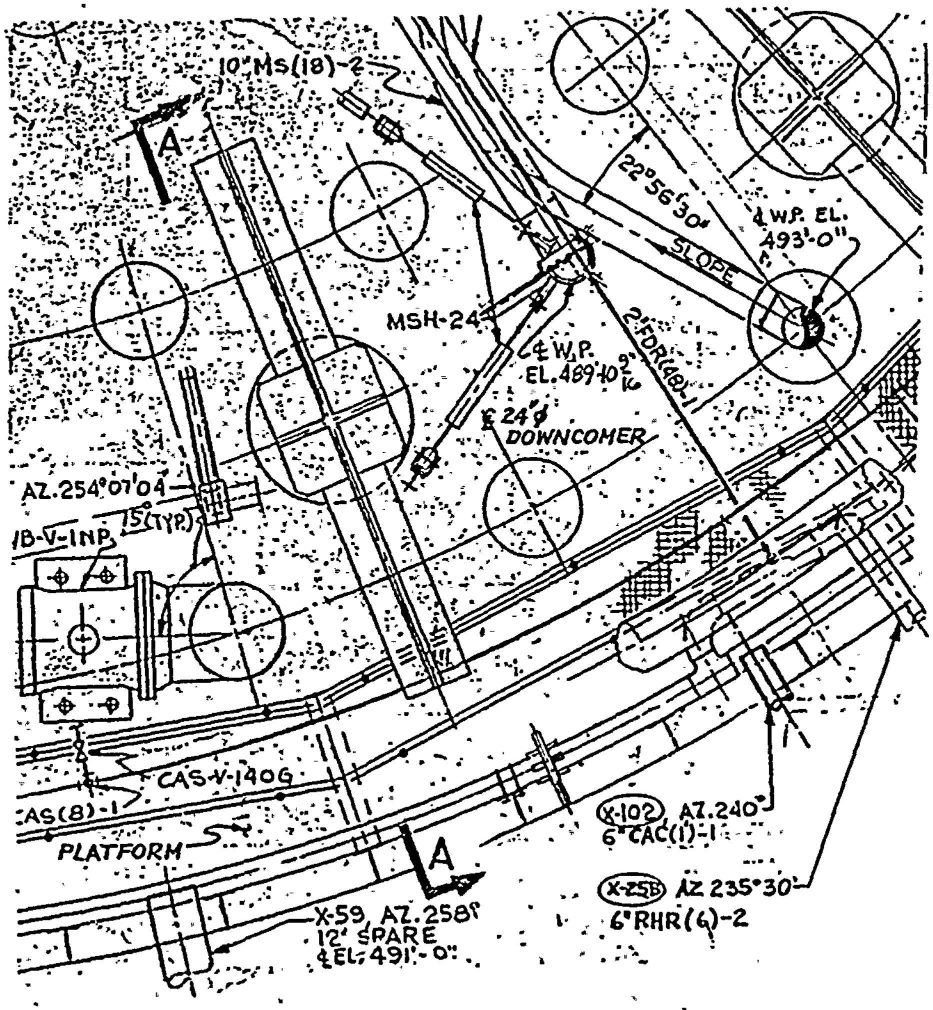
The instrumentation for the SPTM system is shown in Figure 3.2-8.

The following is a description of the suppression pool temperature monitoring system. The power supply for the suppression pool temperature monitoring system is from the 120 VAC instrument buses.

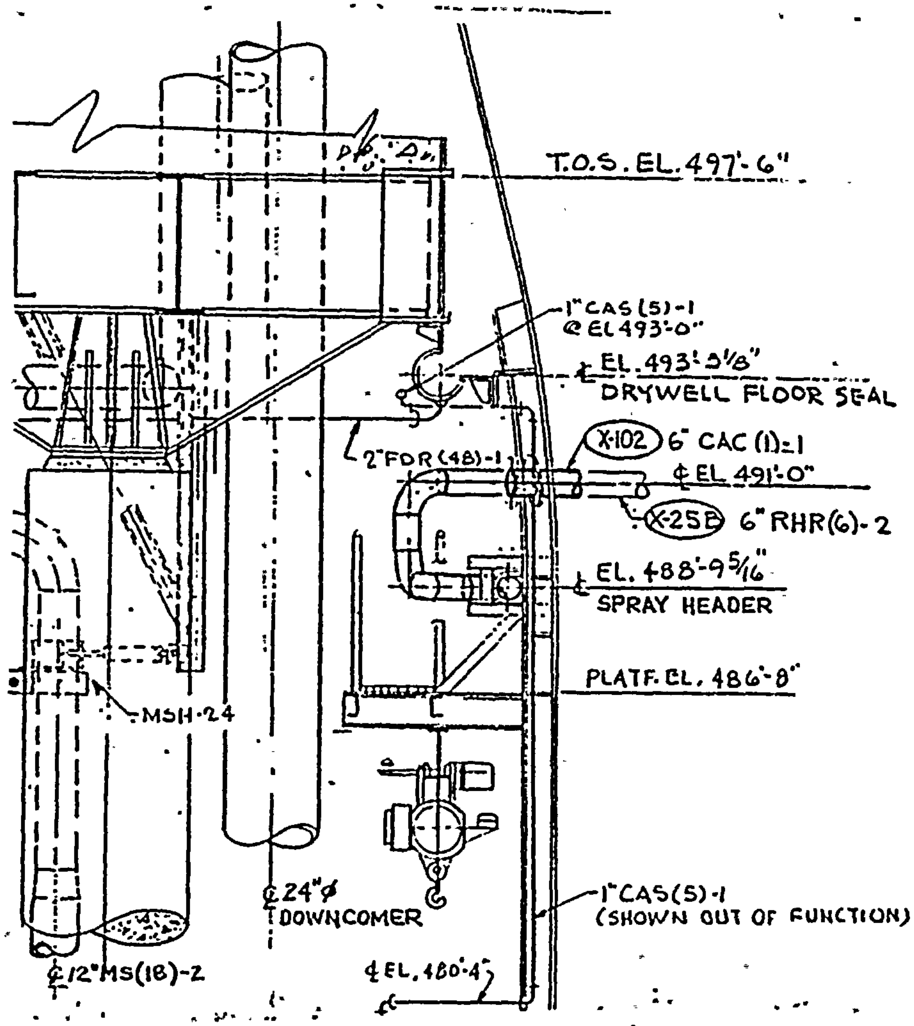
TABLE 7.6-12

CONTAINMENT HYDROGEN AND OXYGEN MONITORING SYSTEM SAMPLE POINT LOCATIONS

SP	PENETRATION #	SAMPLE POINT AZIMUTH	SAMPLE POINT ELEVATION
74	72c	188°-24'	560'-0"
75	72d	191°-36'	560'-0"
76	72e	193°	531'-0"
77	82c	230°	479'-4"
78	85d	18°-12'	545'-2-1/4"
79	85e	13°-44'	545'-1-1/2"
80	73d	45°	531'-0"
81	84b	40°	479'-4"

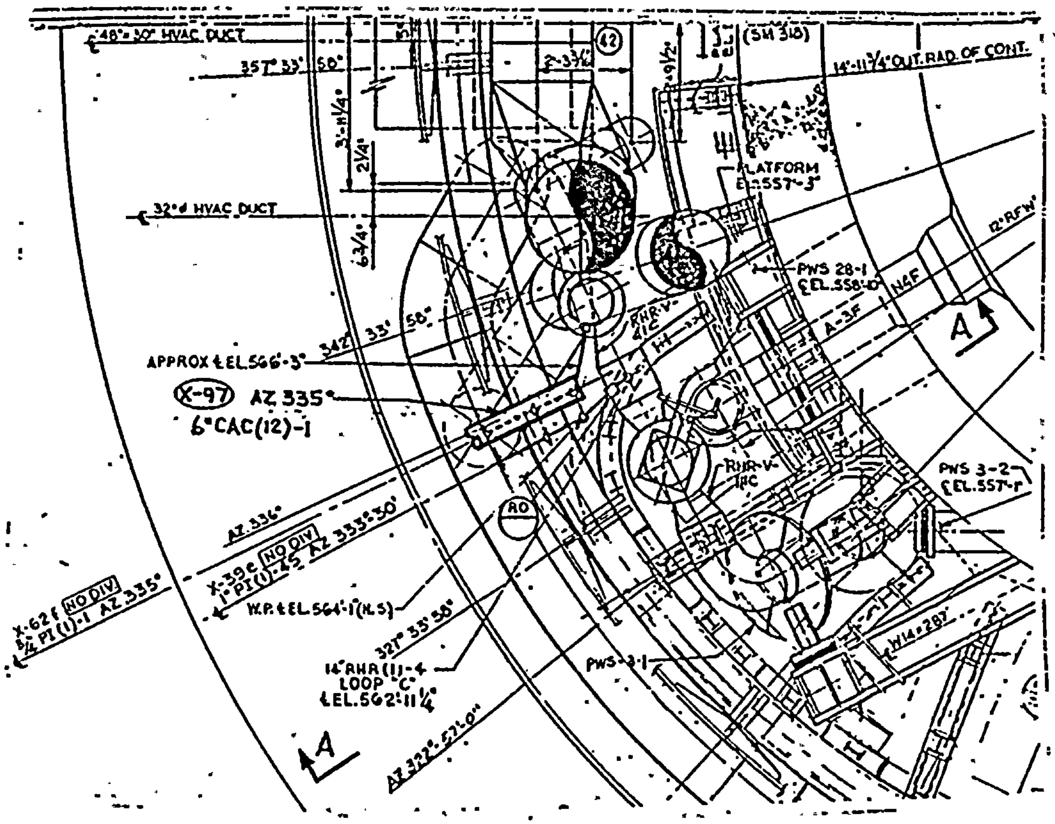


PLAN AT EL. 486'-8"

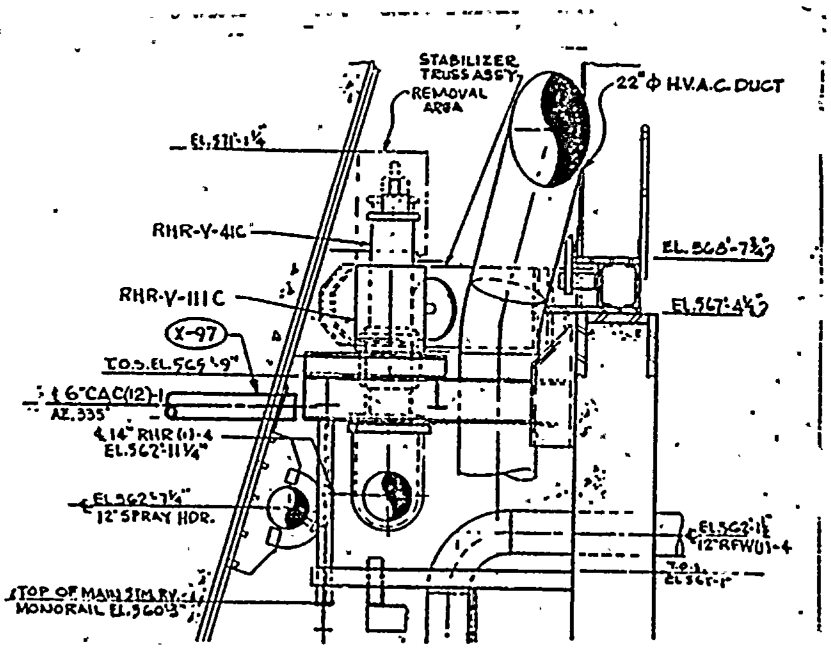


SECTION A-A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRIMARY CONTAINMENT (WETWELL) CAC SYSTEM DISCHARGE PENETRATION	FIGURE 6.2-34

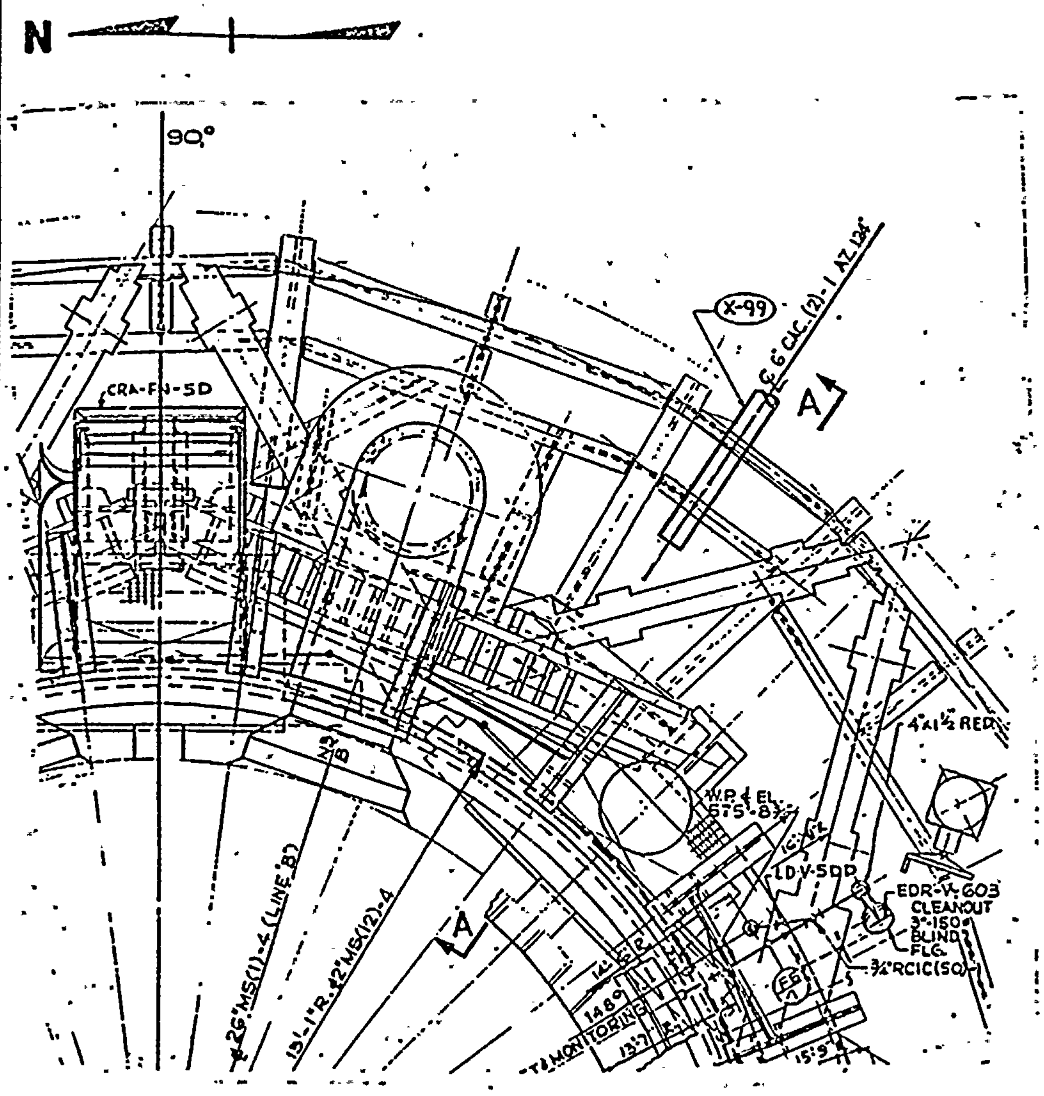


PLAN AT EL.556'-5"

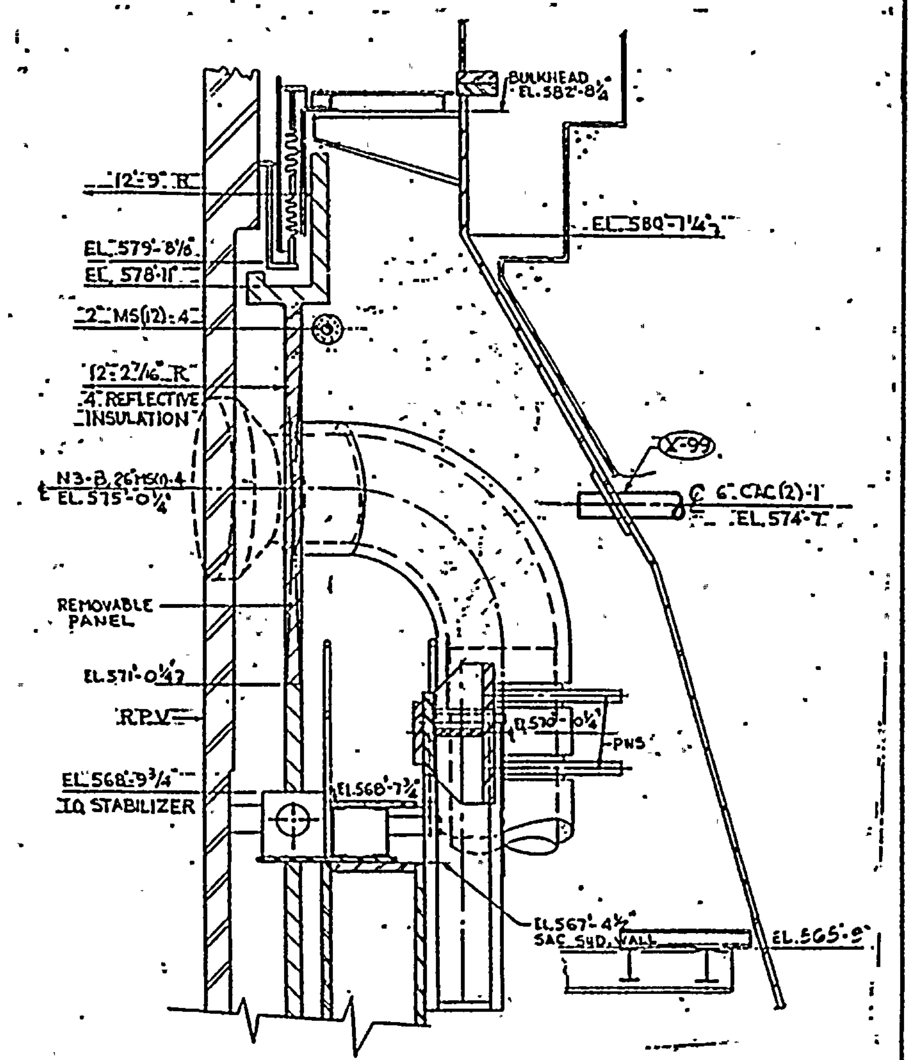


SECTION A-A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRIMARY CONTAINMENT (DRYWELL) CAC SYSTEM SUCTION PENETRATION	FIGURE 6.2-33

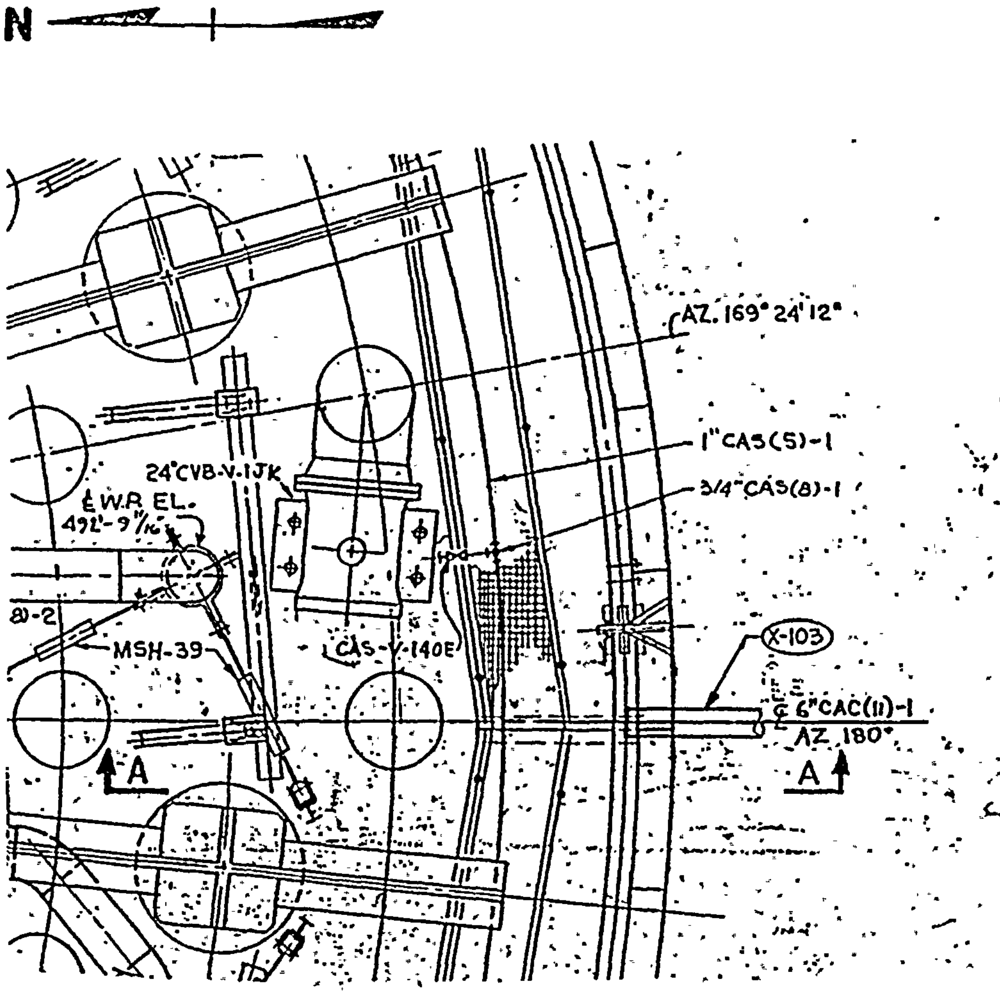


PLAN AT EL.567'-7¾"

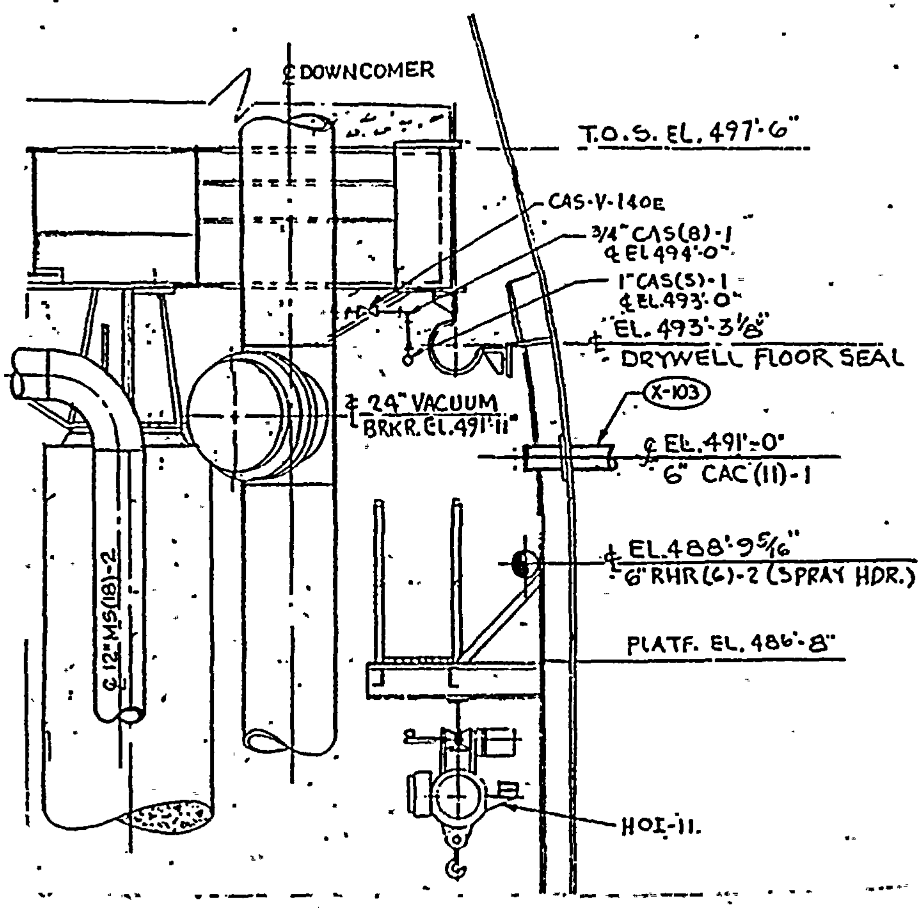


SECTION A-A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRIMARY CONTAINMENT (DRYWELL) CAC SYSTEM SUCTION PENETRATION	FIGURE 6.2-32



PLAN AT EL. 486'-8"



SECTION A-A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRIMARY CONTAINMENT (WETWELL) CAC SYSTEM DISCHARGE PENETRATION	FIGURE 6.2-35

Q. 22.026

In accordance with Appendix J of 10CFR50 we require that containment isolation valves for those systems not vented and drained during Type A tests, are to be tested in accordance with Section III.C of Appendix J and those results are to be reported to the commission.

RESPONSE:

In general the containment isolation valves for those systems not vented and drained during Type A tests are being Type C tested. The exceptions to this are listed in the response to question 22.10 along with the justification. The results of these tests will be reported to the Commission as stated in FSAR Section 6.2.6.4.

Q. 22.027
(6.2.6)

Augment table 6.2-16 to provide the information requested in Section 6.2.4.2, "Systems Design," of Regulatory Guide 1.70.

RESPONSE:

Tables 6.2-13, 6.2-16 and 7.3-13 have been combined into one table, number 6.2-16*. This table has been further expanded to include all the information required in Section 6.2.4.2 of Regulatory Guide 1.70. Ch. 6 will be appropriately revised to reflect the revised table 6.2.16.

*See attached draft table.

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W.O. No. ______ Date ______ Book No. ______ Page No. ______
Drawing No. ______ Calc. No. ______ Sheet ____ of ____
By ______ Checked ______ Approved ______
Title TABLE 6.2-16

LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO.s	GDC	CODE GP.(2)	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 (IN)	CLOSURE TIME (7)(11)	DISTANCE TO PENETRATION (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH)	NOTES
MS Line A	18A	3.2-2	55	A	MS-V-22A	AO Globe	I	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	–	NO	S	VALVES	T.B.	NO	1,15
		6.2-31j			MS-V-28A	AO Globe	O	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	4	NO	S	VALVES	T.B.	NO	1,15
					MS-V-67A	MO GATE	O	AC	AC	B,C,D,P,G,H	RM	O	C	C	AS-IS	1-½	std	5	NO	S	VALVES	T.B.	NO	15
					MSIC-V-3A	MO GATE	O	AC	AC	RM	MANUAL	C	C	O	AS-IS	1-½	std	10	yes	S	VALVES	R.B.	NO	
MS Line B	18B	3.2-2	55	A	MS-V-22B	AO Globe	I	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	–	NO	S	VALVES	T.B.	NO	1,15
		6.2-31j			MS-V-28B	AO Globe	O	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	4	NO	S	VALVES	T.B.	NO	1,15
					MS-V-67B	MO Gate	O	AC	AC	B,C,D,P,G,H	RM	O	C	C	AS-IS	1-½	std	5	NO	S	VALVES	T.B.	NO	15
					MSIC-V-3B	MO Gate	O	AC	AC	RM	MANUAL	C	C	O	AS-IS	1-½	std	10	yes	S	VALVES	R.B.	NO	
MS Line C	18C	3.2-2	55	A	MS-V-22C	AO Globe	I	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	–	NO	S	VALVES	T.B.	NO	1,15
		6.2-31j			MS-V-28C	AO Globe	O	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	4	NO	S	VALVES	T.B.	NO	1,15
					MS-V-67C	MO Gate	O	AC	AC	B,C,D,P,G,H	RM	O	C	C	AS-IS	1-½	std	5	NO	S	VALVES	T.B.	NO	15
					MSIC-V-3C	MO Gate	O	AC	AC	RM	MANUAL	C	C	O	AS-IS	1-½	std	10	yes	S	VALVES	R.B.	NO	
MS Line D	18D	3.2-2	55	A	MS-V-22D	AO Globe	I	AIR	AIR/SPR	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	–	NO	S	VALVES	T.B.	NO	1,15
		6.2-31j			MS-V-28D	AO Globe	O	AIR	Air/spr	B,C,D,P,G,H	RM	O	O/C	C	C	26	3-10 sec	4	NO	S	VALVES	T.B.	NO	1,15
					MS-V-67D	MO GATE	O	AC	AC	B,C,D,P,G,H	RM	O	C	C	AS-IS	1-½	std	5	NO	S	VALVES	T.B.	NO	15
					MSIC-V-3D	MO GATE	O	AC	AC	RM	MANUAL	C	C	O	AS-IS	1-½	std	10	yes	S	VALVES	R.B.	NO	
MS Line Drain	22	3.2-2	55	A	MS-V-16	MO GATE	I	AC	AC	B,C,D,P,G,H	RM	O	C	C	[illegible]	3	std	–	NO	S	VALVES	T.B.	.19	
		6.2-31f			MS-V-19	MO GATE	O	DC	DC	B,C,D,P,G,H	RM	O	C	C	AS-IS	3	std	6	NO	S				
RFW Line A	17A	3.2-2	55	A	RFW-V-11A	check	I	process	process	–	–	O	O/C	O/C	–	24	–	–	NO	W	VALVES	T.B.	1.5	16
		6.2-31b			RFW-V-32A	PC check	O	process	pro/spr	–	–	O	O/C	O/C	–	24	–	2	NO	W				
					RFW-V-65A	M.O. GATE	O	AC	AC	RM	MANUAL	O	O/C	O/C	AS-IS	24	Std	8	No	W				
RFW Line B	17B	3.2-2	55	A	RFW-V-10B	check	I	process	process	–	–	O	O/C	O/C	–	24	–	–	NO	W	VALVES	T.B.	1.5	16
		6.2-31b			RFW-V-32B	PC check	O	process	pro/spr	–	–	O	O/C	O/C	–	24	–	2	NO	W				
					RFW-V-65B	M.O. GATE	O	AC	AC	RM	MANUAL	O	O/C	O/C	AS-IS	24	Std	8	No	W				

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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 (4)	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH)	NOTES
DW Service Line	92	9.2-4	56	B	DW-V-157	GATE	I	MANUAL	MANUAL	–	–	L.C	L.C.	L.C.	–	2"	–		NO	W	VALVES	S.B	.13	
		6.2-31l			DW-V-156	GATE	O	MANUAL	MANUAL	–	–	L.C.	L.C.	L.C.	–	2"	–	5						
RHR Condensing mode	21	3.2-8	55	A	RCIC-V-63	MO GATE	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10"	16 sec.	–	YES	S	VALVES	R.B.	NO	
Steam Supply		6.2-31e			RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1"	5 sec.	–						
					RCIC-V-64	MO GATE	O	DC	DC	X	RM	C	O/C	C	AS-IS	10"	16 sec.	2						
RCIC Turbine Steam	45	3.2-8	55	A	RCIC-V-63	MO GATE	I	AC	AC	K	RM	O	O	O/C	AS-IS	10"	16 sec.	–	NO	S	VALVES	R.B.	NO	
Supply		6.2-31e			RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1"	5 sec.	–						
					RCIC-V-8	MO GATE	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4"	std	2						
RCIC Pump	65	3.2-8	56	B	RCIC-V-19	MO Globe	O	DC	DC	RM	MANUAL	C.	C	C.	AS-IS	2"	5 sec	7	NO	W	VALVES	R.B.	NO	22
minimum flow		6.2-31h																						
RCIC Turbine	4	3.2-8	56	B	RCIC-V-68	MO GATE	O	DC	DC	RM	MANUAL	O	O	O/C	AS-IS	10"	std	10	NO	S	VALVES	R.B.	NO	22
exhaust		6.2-31n																						
RCIC turbine exhaust	116	3.2-8	56	B	RCIC-V-110	MO GATE	O	DC	DC	X1	RM	O	O	O/C	AS-IS	2"	std	9	NO	A	VALVES	R.B.	NO	17,
VACUUM BREAKER		6.2-31l			RCIC-V-113	MO GATE	O	DC	DC	X1	RM	O	O	O/C	AS-IS	2"	std.	5						
RCIC vacuum pump	64	3.2-8	56	B	RCIC-V-69	MO GATE	O	DC	DC	RM	MANUAL	O	O	O/C	AS-IS	1-½"	std.	4	NO	W	VALVES	R.B.	NO	22
discharge		6.2-31g																						
RCIC pump suction	33	3.2-8	56	B	RCIC-V-31	MO GATE	O	DC	DC	RM	MANUAL	C.	C	O/C	AS-IS	8"	std.	2	NO	W	VALVES	R.B.	NO	23
from Suppression Pool		6.2-31m																						
RPV Head Spray	2	3.2-8	55	A	RCIC-V-66	check	I	process	process	–	–	C	O	O/C	–	6"	–	–	NO	W	VALVES	R.B.	NO	3
		6.2-31e			RCIC-V-13	MO GATE	O	DC	DC	RM	MANUAL	C	O/C	O/C	AS-IS	6"	15 sec	2	NO	W	VALVES	R.B	NO	
					RHR-V-23	MO Globe	O	DC	DC	A,U,M,X2	RM	C	O/C	C	AS-IS	6"	std	7	Yes	W	VALVES	R.B	NO	

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LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO's	GDC	CODE GR. (12)	VALVE NO.	VALVE TYPE	LOCATION	POWER TO OPEN (5)	POWER TO CLOSE (5)	ISOLATION SIGNAL (1)	BACK-UP	NORMAL POSITION (10)	SHUTDOWN POSITION	POST LOCA	FAILURE POSITION (6)	VALVE SIZE (14)	CLOSURE TIME (7) (11)	DISTANCE TO PENETRATION (F4)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (13) (FN)	NOTES
Drywell spray Loop A	11A	3.2-6	56	B	RHR-V-16A	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	16	10 sec.	26	Yes	W	VALVES	RB	NO	17
		6.2-31q			RHR-V-17A	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	16	10 sec	24						
Drywell spray Loop B	11B	3.2-6	56	B	RHR-V-16B	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	16	10 sec.	12	Yes	W	VALVES	RB	NO	17
		6.2-31q			RHR-V-17B	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	16	10 sec	2						
LPCI Loop A	12A	3.2-6	55	A	RHR-V-41A	check	I	PROCESS	PROCESS	—	—	C	C	O/C	—	14	—	—	Yes	W	VALVES	RB	NO	3,
		6.2-31l			RHR-V-42A	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	14	12 sec	21						
LPCI Loop B	12B	3.2-6	55	A	RHR-V-41B	check	I	PROCESS	PROCESS	—	—	C	C	O/C	—	14	—	—	Yes	W	VALVES	RB	NO	3,
		6.2-31l			RHR-V-42B	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	14	12 sec	20						
LPCI Loop C	12C	3.2-6	55	A	RHR-V-41C	check	I	PROCESS	PROCESS	—	—	C	C	O/C	—	14	—	—	Yes	W	VALVES	RB	NO	3, 24
		6.2-31l			RHR-V-42C	MO GATE	O	AC	AC	Rm	MANUAL	C	C	O/C	AS-IS	14	12 sec	20						
Shutdown Cooling Return Loop A	19A	3.2-6	55	A	RHR-V-50A	check	I	PROCESS	PROCESS	—	—	C	O	C	—	12	—	—	Yes	W	VALVES	RB	NO	3,
		6.2-31m			RHR-V-123A	MO GATE	I	AC	AC	A,U,M,X2,F	RM	C	O/C	C	AS-IS	1		—						
					RHR-V-53A	MO Globe	O	AC	AC	A,U,M,X2	RM	C	O	C	AS-IS	12	40 sec	5						
Shutdown Cooling Return Loop B	19B	3.2-6	55	A	RHR-V-50B	check	I	PROCESS	PROCESS	—	—	C	O	C	—	12	—	—	Yes	W	VALVES	RB	NO	3,
		6.2-31m			RHR-V-123B	MO GATE	I	AC	AC	A,U,M,X2,F	RM	C	O/C	C	AS-IS	1		—						
					RHR-V-53B	MO Globe	O	AC	AC	A,U,M,X2	RM	C	O	C	AS-IS	12	40 sec.	2						
Shutdown Cooling Suction	20	3.2-6	55	A	RHR-V-9	MO GATE	I	AC	AC	A,U,M,X2	RM	C	O	C	AS-IS	20	40 sec	—	Yes	W	VALVES	RB	NO	
		6.2-31k			RHR-V-8	MO GATE	O	AC	AC	A,U,M,X2	RM	C	O	C	AS-IS	20	40 sec	14						
Suppression Pool Spray Loop A	25A	3.2-6	56	B	RHR-V-27A	MO GATE	O	AC	AC	F, X3	RM	C	C	O/C	AS-IS	6	10 sec	5	Yes	W	VALVES	RB	NO	2, 18
		6.2-31h																						
Suppression Pool Spray Loop B	25B	3.2-6	56	B	RHR-V-27B	MO GATE	O	AC	AC	F, X3	RM	C	C	O/C	AS-IS	6	10 sec	6	Yes	W	VALVES	RB	NO	2, 18
		6.2-31h																						

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LINE DESCRIPTION	PENETRATION No.	FSAR FIGURE No's	GDC	CODE GP (2)	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 (FT)	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH) (13)	NOTES
RHR Loop A:	47	3.2-6	56	B																				
pump test line		6.2-31p			RHR-V-24A	MO Globe	O	AC	AC	F, X3	RM	C	C	O/C	AS-2S	18	std	12	Yes	W	VALVES	R.B.	NO	2,18
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, X3	RM	C	O/C	C	AS-2S	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	RM	MANUAL	O	C	O/C	AS-2S	3	8 sec	22	Yes	W	VALVES	R.B.	NO	18
heat exch. thermal relief					RHR-RV-1A	relief	O	PP	SPRING	–	–	C	C	C	–	1½	–	188	Yes	W	VALVES	R.B	NO	18,19
heat exch. vent					RHR-V-73A	MO Globe	O	AC	AC	RM	MANUAL	C	O/C	C	AS-2S	2	std	175	Yes	A	VALVES	R.B.	NO	18
FDR system intertie					RHR-V-121	Gate	O	MANUAL	MANUAL	–	–	L.C.	L.C.	L.C.	–	3	–	5	NO	W	VALVES	R.B.	NO	
CAC system Loop A drain					RHR-V-134A	MO Gate	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	2	std	44	Yes	W	VALVES	R.B.	NO	18
RHR Loop B:	48	3.2-6	56	B																				
pump test line		6.2-31p			RHR-V-24B	MO Globe	O	AC	AC	F, X3	RM	C	C	O/C	AS-2S	18	std	12	Yes	W	VALVES	R.B.	NO	2,
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
heat exch. condensate					RHR-V-11B	MO GATE	O	AC	AC	F, X3	RM	C	O/C	C	AS-2S	4	std	15	Yes	W	VALVES	R.B	NO	18
pump minimum flow					RHR-FCV-64B	MO Globe	O	AC	AC	RM	MANUAL	O	C	O/C	AS-2S	3	8 sec	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	–	1-½	–	189	Yes	W	VALVES	R.B.	NO	18,19
heat exch. vent					RHR-V-73B	MO Globe	O	AC	AC	RM	MANUAL	C	O/C	C	AS-2S	2	std	190	Yes	A	VALVES	R.B.	NO	18
CAC System Loop B drain					RHR-V-134B	MO Gate	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	2	std	44	Yes	W	VALVES	R.B.	NO	18
RHR Loop C:	26	3.2-6	56	B																				
pump test line		6.2-31S			RHR-V-21	MO Globe	O	AC	AC	F, X3	RM	C	C	C	AS-2S	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-FCV-64C	MO Globe	O	AC	AC	RM	MANUAL	O	C	O/C	AS-2S	3	8 sec	30	Yes	W	VALVES	R.B.	NO	18

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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 (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH) (13)	NOTES
RHR Loop A Suppression	35	3.2-6	56	B	RHR-V-4A	MO GATE	O	AC	AC	RM	MANUAL	O	O/C	O	AS-IS	24	Std	2	Yes	W	VALVES	RB	NO	18
Pool Suction		6.2-31n																						
RHR Loop B Suppression	37	3.2-6	56	B	RHR-V-4B	MO GATE	O	AC	AC	RM	MANUAL	O	O/C	O	AS-IS	24	Std	2	Yes	W	VALVES	RB	NO	18
Pool Suction		6.2-31n																						
RHR Loop C Suppression	36	3.2-6	56	B	RHR-V-4C	MO GATE	O	AC	AC	RM	MANUAL	O	O	O	AS-IS	24	Std	2	Yes	W	VALVES	RB	NO	18
Pool Suction		6.2-31n																						
RHR Loop A:	117	3.2-6	56	B																				
heat exch. steam relief		6.2-31d			RHR-RV-55A	relief	O	PP	Spring	—	—	C	C	C	—	10	—	24	Yes	S	VALVES	RB	NO	18, 19
condensate pot drain					RHR-V-124A	MO GATE	O	AC	AC	RM	MANUAL	C	C	C	AS-IS	1-1/2	Std	11	Yes	W	VALVES	RB	NO	18
condensate pot drain					RHR-V-124B	MO GATE	O	AC	AC	RM	MANUAL	C	C	C	AS-IS	1-1/2	Std	12	Yes	W	VALVES	RB	NO	18
RHR Loop B:	118	3.2-6	56	B																				
heat exch. steam relief		6.2-31d			RHR-RV-55B	relief	O	PP	Spring	—	—	C	C	C	—	10	—	21	Yes	S	VALVES	RB	NO	18, 19
condensate pot drain					RHR-V-125A	MO GATE	O	AC	AC	RM	MANUAL	C	C	C	AS-IS	1-1/2	Std	17	Yes	W	VALVES	RB	NO	18
condensate pot drain					RHR-V-125B	MO GATE	O	AC	AC	RM	MANUAL	C	C	C	AS-IS	1-1/2	Std	14	Yes	W	VALVES	RB	NO	18
RRC hydraulic	76f	3.2-3	57	B	HY-V-17a	S.O GLOBE	O	AC	SPRING	B, F	RM	O	O	C	C	3/4	<5 sec		NO	H	VALVES	RB	NO	28
control lines	76b				HY-V-18a											3/4								
	76e				HY-V-19a											1/2								
	76c				HY-V-20a											1/2								
	77f				HY-V-17b											3/4								
	77b				HY-V-18b											3/4								
	77e				HY-V-19b											1/2								
	77c				HY-V-20b											1/2								

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LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO.'S	GDC	CODE GR. (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 (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (13) LEAKAGE (SCFH)	NOTES
CAC Division 1	96	3.2-17	56	B	CAC-V-2	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	4	Yes	A	VALVES	RB	NO	17
discharge to drywell		6.2-31g			CAC-FCV-2A	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	6						
CAC Division 2	97	3.2-17	56	B	CAC-V-15	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	VALVES	RB	NO	17
Suction from drywell		6.2-31g			CAC-FCV-1B	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	4						
CAC Division 2	98	3.2-17	56	B	CAC-V-11	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	8	Yes	A	VALVES	RB	NO	17
discharge to drywell		6.2-31g			CAC-FCV-2B	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	10						
CAC Division 1	99	3.2-17	56	B	CAC-V-6	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	4	Yes	A,S	VALVES	RB	NO	17
Suction from drywell		6.2-31g			CAC-FCV-1A	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1	102	3.2-17	56	B	CAC-V-4	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	3	Yes	A	VALVES	RB	NO	17
discharge to wetwell		6.2-31g			CAC-FCV-4A	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	5						
CAC Division 2	103	3.2-17	56	B	CAC-V-13	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	7	Yes	A	VALVES	RB	NO	17
discharge to wetwell		6.2-31g			CAC-FCV-4B	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	9						
CAC Division 2	104	3.2-17	56	B	CAC-V-17	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	5	Yes	A,S	VALVES	RB	NO	17
suction from wetwell		6.2-31g			CAC-FCV-3B	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1	105	3.2-17	56	B	CAC-V-8	MO GATE	O	DC	DC	RM	MANUAL	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	VALVES	RB	NO	17
suction from wetwell		6.2-31g			CAC-FCV-3A	EHO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	C	2-1/2	Std	6						

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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 (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (13) (SCFH)	NOTES
RB to Wetwell Vacuum	119	3.2-15	56	B	CSP-V-9	AO Butfy	O	Spring	AIR	RM	-	C	C	C	O	24	4 sec	1	YES	A	Valves	RB	NO	17, 26
Breakers		6.2-31g			CSP-V-10	PC Check	O	Process	Process	-	RM	C	C	C	—	24	—	4	YES	A	Valves	RB	NO	
RB to Wetwell Vacuum	66	3.2-15	56	B	CSP-V-5	AO Butfy	O	Spring	AIR	RM	-	C	C	C	O	24	4 sec	7	YES	A	Valves	RB	NO	17, 26
Breakers and		6.2-31b			CSP-V-7	PC Check	O	Process	Process	-	RM	C	C	C	—	24	—	10	YES	A	Valves	RB	NO	
Wetwell Ventilation		6.2-31g			CSP-V-4	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	24	4 sec	14	NO	A	Valves	RB	NO	
Supply					CSP-V-3	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	24	4 sec	17	NO	A	Valves	RB	NO	
RB to Wetwell Vacuum	67	3.2-15	56	B	CSP-V-6	AO Butfy	O	Spring	AIR	RM	-	C	C	C	O	24	4 sec	9	YES	A	Valves	RB	NO	17, 26
Breakers and		6.2-31j			CSP-V-8	PC Check	O	Process	Process	-	RM	C	C	C	—	24	—	16	YES	A	Valves	RB	NO	
Wetwell Ventilation		6.2-31g			CEP-V-4A	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	24	4 sec	10	NO	A	Valves	RB	NO	
Exhaust					CEP-V-3A	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	24	4 sec	12	NO	A	Valves	RB	NO	
					CEP-V-4B	AO Gate	O	AIR	Spring	FAZ	RM	C	C	C	C	2	1 sec	10	NO	A	Valves	RB	NO	
					CEP-V-3B	AO Gate	O	AIR	Spring	FAZ	RM	C	C	C	C	2	1 sec	12	NO	A	Valves	RB	NO	
Drywell Ventilation	53	3.2-15	56	B	CSP-V-2	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	30	4 sec	1	NO	A	Valves	RB	NO	17
Supply		6.2-31b			CSP-V-1	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	30	4 sec	4	NO	A	Valves	RB	NO	
Drywell Ventilation	3	3.2-15	56	B	CEP-V-1A	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	30	4 sec	12	NO	A	Valves	RB	NO	17
Exhaust		6.2-31j			CEP-V-2A	AO Butfy	O	AIR	Spring	FAZ	RM	C	C	C	C	30	4 sec	8	NO	A	Valves	RB	NO	
					CEP-V-1B	AO Gate	O	AIR	Spring	FAZ	RM	C	C	C	C	2	1 sec	12	NO	A	Valves	RB	NO	
					CEP-V-2B	AO Gate	O	AIR	Spring	FAZ	RM	C	C	C	C	2	1 sec	8	NO	A	Valves	RB	NO	

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LINE DESCRIPTION	PENETRATION No.	FSAR FIGURE No.s	GDC	CODE GR. (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 (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH) (13)	NOTES
RCC Inlet header	5	3.2-14	56	B	RCC-V-164	MO GATE	O	AC	AC	RM	manual	O	O		AS-IS	10	Std	5	NO	W	VALVES	RB	NO	17
		6.2-31t			RCC-V-5	MO GATE	O	AC	AC	RM	manual	O	O		AS-IS	10	Std	3						
RCC Outlet header	46	3.2-14	56	B	RCC-V-21	MO GATE	O	AC	AC	RM	manual	O	O		AS-IS	10	Std	3	NO	W	VALVES	RB	NO	
		6.2-31o			RCC-V-40	MO GATE	I	AC	AC	RM	—	O	O		AS-IS	10	Std	—						
Suppression Pool Clean up suction	100	3.2-12	56	B	FPC-V-153	MO GATE	O	AC	AC	FA	RM	C	C	C	AS-IS	6	Std	2	NO	W	VALVES	RB	NO	17
		6.2-31x			FPC-V-154	MO GATE	O	AC	AC	FA	RM	C	C	C	AS-IS	6	Std	7						
Suppression pool clean up return	101	3.2-12	56	B	FPC-V-156	MO GATE	O	AC	AC	FA	RM	C	C	C	AS-IS	6	Std	6	NO	W	VALVES	RB	NO	17
		6.2-31o			FPC-V-[illegible]	Globe	O	MANUAL	MANUAL	—	—	L.C	LC	L.C.	—	6	—	41						
RWCU from Reactor	14	3.2-11	55	A	RWCU-V-1	MO GATE	I	AC	AC	B,U,A,X2	RM	O	O	C	AS-IS	6	Std	1	NO	W	VALVES	RB	NO	
		6.2-31k			RWCU-V-4	MO GATE	O	DC	DC	B,U,M,X2	RM	O	O	C	AS-IS	6	Std	4						
RRC pump A seal water	43A	3.2-3	56	B	RRC-V-13A	check	I	process	process	—	—	O	O	C	—	3/4	Std	—	NO	W	VALVES	RB	NO	
		6.2-31c			RRC-V-16A	MO GATE	O	AC	AC	RM	manual	O	O	C	AS-IS	3/4	Std.	2						
RRC pump B seal water	43B	3.2-3	56	B	RRC-V-13B	check	I	process	process	—	—	O	O	C	—	3/4	Std	—	NO	W	VALVES	RB	NO	
		6.2-31c			RRC-V-16B	MO GATE	O	AC	AC	RM	MANUAL	O	O	C	AS-IS	3/4	Std	2						
RRC sample line	69d	3.2-3	55	A	RRC-V-19	SO Globe	I	AC	Spring	B,C,D,P	RM	C	C	C	C	3/4	<5 sec	—	NO	W	VALVES	T.B.	.05	
		6.2-31d			RRC-V-20	AO Globe	O	AIR	Spring	B,C,D,P	RM	C	C	C	C	3/4	Std							

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LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO.s	GDC	CODE GR. (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 (4)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (13) (SCFH)	NOTES
Drywell Equipment	23	3.2-9	56	B	EDR-V-19	AO GATE	O	AIR	SPRING	F,A	RM	O	O	C	C	3	std	2	NO	W	VALVES	RB	NO	17
Drain		6.2-31h			EDR-V-20	AO GATE	O	AIR	SPRING	F,A	RM	O	O	C	C	3	std	4						
Drywell Floor	24	3.2-10	56	B	FDR-V-3	AO GATE	O	AIR	SPRING	F,A	RM	O	O	C	C	3	std	2	NO	W	VALVES	RB	NO	17
Drain		6.2-31h			FDR-V-4	AO GATE	O	AIR	SPRING	F,A	RM	O	O	C	C	3	std	3						
Decontamination Soltn.	94	3.2-10	NA	B	—	—	—	—	—	—	—	—	—	—	—	4	—	—	—	—	Blanked close	RB	NO	
Supply Header																								
Decontamination Soltn.	95	3.2-10	NA	B	—	—	—	—	—	—	—	—	—	—	—	4	—	—	—	—	Blanked close	RB	NO	
Return Header																								
CIA for Safety Relief	56	3.2-21	56	B	CIA-V-21	check	O	process	process	—	—	C	C	C	—	3/4	—	3	NO	A	VALVES	RB	NO	17
Valve Accumulators		6.2-31c			CIA-V-20	MO Globe	O	AC	AC	RM	manual	O	O	O	AS-25	3/4	std	10						
CIA Line A for ADS	89B	3.2-21	56	B	CIA-V-31A	check	O	process	process	—	—	C	C	C	—	1/2	—	5	NO	A	VALVES	RB	NO	17
accumulators		6.2-31c			CIA-V-30A	MO Globe	O	AC	AC	RM	MANUAL	O	O	O	AS-25	1/2	std	15						
CIA Line B for ADS	91	3.2-21	56	B	CIA-V-31B	check	O	process	process	—	—	C	C	C	—	1/2	—	2	NO	A	VALVES	RB	NO	17
accumulators		6.2-31c			CIA-V-30B	MO Globe	O	AC	AC	RM	MANUAL	O	O	O	AS-25	1/2	std	15	—					
CRD Insert lines (185 separate lines)	9	3.2-4	54			SEE NOTE 4																		4
CRD Withdrawal lines (185 separate lines)	10	3.2-4	54			SEE NOTE 4																		4

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LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO.s	GDC	CODE GR.(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 (FT)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH) (13)	NOTES
air line for testing	42d	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
RHR-V-50A		3.2-6			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	69c	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
RHR-V-50B		3.2-6			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	61f	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
RHR-V-41A		3.2-6			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	57Bf	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
RHR-V-41B		3.2-6			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	62f	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	R.B	NO	25
RHR-V-41C		3.2-6			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	78d	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	R.B	NO	25
LPCS-V-6		3.2-7			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	78e	6.2-31r	56	B	*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
HPCS-V-5		3.2-7			*	Globe	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
air line for testing	82e	6.2-31r	56	B	*	SO Globe	O	AC	SPRING	RM	MANUAL	C	C	C	C	1	<5sec		NO	A	VALVES	RB	NO	25
Wetwell - drywell		9.3-1			*	Check	O	process	process	—	—	C	C	C	—	1	—							
vacuum relief valves																								
airline for testing	59fa	6.2-31r	56	B	*	GLOBE	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—		NO	A	VALVES	RB	NO	25
RCIC-V-66		3.2-8			*	GLOBE	O	MANUAL	MANUAL	—	—	L.C.	L.C.	L.C.	—	1	—							
TIP LINES	72a- THRU 27a	—	54	—	CA 3004	SO. BALL	O	AC	AC	B,F	RM	C	C	C	C	3/8		2	NO	—	VALVES	EB	NO	29
						SHEAR	O	—	AC	RM	—	O	O	O	O	3/8	—	2	NO	—				
* VALVE NUMBERS TO BE ASSIGNED																								

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LINE DESCRIPTION	PENETRATION NO.	FSAR FIGURE NO.s	GDC	CODE GR.(2)	VALVE No.	VALVE TYPE	LOCATION	POWER TO OPEN (5)	POWER TO CLOSE (5)	ISOLATION SIGNAL (4)	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 (15)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (SCFH) (3)	NOTES
Radiation Monitor (S-SR-20)	85b	6.2-31s	56	B	-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-	NO	A	VALVES	RB	NO	25
Supply line		3.2-15			-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-						
Radiation Monitor (S-SR-20)	72f	6.2-31s	56	B	-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-	NO	A	VALVES	RB	NO	25
return line		3.2-15			-	check	O	PROCESS	PROCESS	-	MANUAL	O	O	O/C	C	1	-	-						
Radiation Monitor (S-SR-21)	73e	6.2-31s	56	B	-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-	NO	A	VALVES	RB	NO	25
supply line		3.2-15			-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-						
Radiation Monitor (S-SR-21)	29c	6.2-31s	56	B	-	SO GATE	O	AC	SPRING	RM	MANUAL	O	O	O/C	C	1	< 5 sec	-	NO	A	VALVES	RB	NO	25
Return line		3.2-15			-	check	O	PROCESS	PROCESS	-	-	O	O	O/C	C	1	-	-						
All Instrument lines	-		55	A	-	EF check	O	SPRING	EF	-	-	O	O	O	-	3/4 & 1	-	-	-	-	VALVES	RB	NO	27
from reactor						Globe	O	MANUAL	MANUAL	-	-	O	O	O	-	3/4 & 1	-	-						
All Instrument lines	-		56	B	-	EF check	O	SPRING	EF	-	-	O	O	O	-	1 & 1½	-	-	-	-	VALVES	RB	NO	27
from primary containment						Globe	O	MANUAL	MANUAL	-	-	O	O	O	-	1 & 1½	-	-						
Instrument lines	-	3.2-15	56	B	-	check	O	PROCESS	PROCESS	-	-	C	C	O/C	-	1	-	-	YES	A, S	VALVES	RB	NO	27
(hydrogen monitors) return to containment						Globe	O	MANUAL	MANUAL	-	-	O	O	O	-	1	-	-						

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LINE DESCRIPTION	PENETRATION No.	FSAR FIGURE No.s	GDC	CODE Gp.(8)	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 (IN)	CLOSURE TIME (7) (11)	DISTANCE TO PENETRATION (Ft)	LEADS TO ESF SYSTEM	PROCESS FLUID	LEAKAGE BARRIER (13)	TERMINATION ZONE (13)	POTENTIAL BYPASS LEAKAGE (13) (SCFH)	NOTES
HPCS to Reactor	6	3.2-7	55	A	HPCS-V-5	check	I	PROCESS	PROCESS	—	—	C	C	O/C	—	12	—	—	yes	W	VALVES	RB	NO	3,24
		6.2-31l			HPCS-V-4	MO GATE	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	12	17 sec.	9						
LPCS to Reactor	8	3.2-7	55	A	LPCS-V-6	check	I	—	—	—	—	C	C	O/C	—	12	—	—	yes	W	VALVES	RB	NO	3,24
		6.2-31l			LPCS-V-5	MO GATE	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	12	27 sec.	22						
HPCS pump suction	31	3.2-7	56	B	HPCS-V-15	MO Gate	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	18	18 sec.	3	yes	W	VALVES	RB	NO	18,24
from suppression pool		6.2-31n																						
LPCS pump suction	34	3.2-7	56	B	LPCS-V-1	MO GATE	O	AC	AC	RM	MANUAL	O	O	O/C	AS-2S	24	std	2	Yes	W	VALVES	RB	NO	18,24
		6.2-31n																						
HPCS test line	49	3.2-7	56	B	HPCS-V-23	MO Globe	O	AC	AC	F,X3	RM	C	C	C	AS-2S	12	std	6	yes	W	VALVES	RB	NO	18
HPCS pump min. flow		6.2-31f			HPCS-V-12	MO GATE	O	AC	AC	RM	MANUAL	C	C	C	AS-2S	4	4 sec.	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	56	B	LPCS-V-12	MO Globe	O	AC	AC	F,X3	RM	C	C	C	AS-2S	12	std	4	yes	W	VALVES	RB	NO	18
LPCS pump min. flow		6.2-31f			LPCS-RV-11	MO Globe	O	AC	AC	RM	MANUAL	C	C	O/C	AS-2S	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	55	A	SLC-V-7	check	I	—	—	—	—	C	C	C	—	1-1/2	—	—	NO	W	VALVES	RB	NO	
		6.2-31m			SLC-V-6	check	O	—	—	—	—	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

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TABLE 6.2-16

ISOLATION SIGNAL CODES FOR TABLE 6.2-16

Signal	Description
A*	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.)
B*	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.

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" + RHR/RCIC EQUIPMENT AREA HIGH TEMPERATURE
X1	HIGH DRYWELL PRESSURE AND LOW REACTOR PRESSURE
X2	RHR EQUIPMENT AREA HIGH TEMPERATURE
X3	REACTOR VESSEL LOW WATER LEVEL (TRIP 1)

*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

1. Valve Type

AO	air operated
MO	motor operated
PC	positive closing
EHO	electro-hydraulic operated
SO	solenoid operated

2. Location

I	inside containment
O	outside containment

3. Power to Open/Close

AC	AC electrical power
DC	DC electrical Power
Process, pro	process flow
PP	process fluid overpressurization
spr	spring

4. Normal Position

O	open
C	close

5. Process fluid

W	water
A	air
S	steam
H	HYDRAULIC FLUID

6. Termination Zone

TB	turbine building
RB	reactor building
Rad W	radwaste building
SB	SERVICE BUILDING

TABLE 6.2-16 (Continued)

NOTES FOR TABLE

These notes are keyed by number to correspond to number in parenthesis, in Table 7.3-13. Type C testing is discussed in Figure 6.2-31 which shows the isolation valve arrangement. Unless otherwise noted all valves listed in Table 6.2-16 are Type C tested (except 4, 27, 28, 29)

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. Control rod hydraulic lines can be isolated ~~by the~~ solenoid ~~valves outside~~ the ~~primary~~ containment. Lines that exter REPLACED – SEE ATTACHMENT FOLLOWING NOTE 29. and terminate ~~in a~~ system that ~~is~~ designed to prevent out-leakage. Solenoid valves normally ~~are~~ closed, but ~~they~~ open on rod movement and during reactor ~~scram.~~

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.

TABLE 6.2-16 (Continued)

7. The standard minimum closing rate is 12 inches per minute for gate valves and 4 inches 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. These are isolation signals of the PCRVIS. Process control signals controlling containment isolation valves are not indicated here. For process control signals, see the appropriate section pertaining to the subject system. An example of this note is the RCIC pump minimum flow bypass line to the suppression pool (Flow Diag. M519, Pene. No. 65). The table indicates RM for isolation signal. This valve (RCIC-V-19) receives a process control signal as indicated in 7.4.1.1.3.3.

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 only.

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. Size indicated is containment side of relief valve when relief valve size is not equal on both sides.

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.

Table 6.2-16 (Continued)

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, and -15 to be updated to reflect the configurations shown on Figures 6.2-31r and -31s.

26. An air operator is provided on the check valve to enable the operator to positively close the check valve. The check valve position indication will be utilized to determine need for operator action. The air operator switch has an automatic return to neutral so the vacuum breaker function will not be impaired.

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

27. (continued)

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 1 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 1, Code Group B and the following criteria:

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 additon, 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 fo 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. 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 the 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 Manufactureres 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 F
Leak Rate	10^{-3} cm^3/s

A statistically chosen sample of the shear valves is tested by the manufacturer to the following criteria:

Pressure	0 - 125 psig
Temperature	340° F
Leak Rate	10^{-3} cm^3/sec STP

The shear valves have explosive squibs and require testing to destruction. They cannot therefore be 100% tested.

29. (continued)

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 flang/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.

NOTE 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.

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.

Q. 022.028

Several of the loads presented in Table 3.4-1 of the Plant Design Assessment Report (DAR) have been generated using computer codes which have not been reviewed by the NRC staff. Provide a complete description of your method of analysis for all codes presented in Appendix D of the DAR.

RESPONSE:

Three computer codes have been used to develop short term LOCA hydrodynamic loads for WNP-2 plant assessment. The three codes are the downcomer vent clearing analytical model computer code VENT, the pool swell analytical model computer code SWELL, and the LOCA bubble charging analytical model computer code BUBBLE. Complete documentation on each of the above three codes is provided or referenced in Appendix D of the DAR (see below for specific references). Documentation on the WNP-2 load calculation procedure is provided in 3.2.1 of the DAR.

1. Vent Code

   a) Assumptions

      See Section D.2 of the DAR.

   b) Equations

      See Fig. D-1 of the DAR.

   c) Methodology

      See Fig. D-1 of the DAR.

2. Swell Code

   The assumptions, equations, and methodology for the swell code are identical to that described in reference D-1.

3. Bubble Code

   The assumptions, equations, and methodology for the Bubble Code are identical to that described in reference D-7.

Not sent WPPSS 11/2

Q 022.029

Safety issues such as the proposed reductions in the pool boundary chugging loads and the safety/relief valve quenchers loads are being resolved generically and are not scheduled for resolution until about 1980. Discuss your short term solutions for these types of problems which require a long period of time to be resolved.

Response

Two important suppression pool hydrodynamic loads that were described in DAR revision 1 as being under study to improve the load definitions are chugging loads on the pool boundary and quencher safety relief valve air clearing loads on the pool boundary.

The definition of the chugging pool boundary load is based on data obtained by General Electric in the 4T test facility, which is representative of a full scale single cell in a Mark II geometry. Influence of certain test facility parameters has been evident in the pressure measurements made during the test. Identification of these parameters has been the subject of an analysis effort by Burns and Roe with the goal being to define a chugging forcing function, independent of the test facility, i.e., that could be applied in a Mark II suppression pool at vent exits. A computational methodology and the associated computer code, has been developed by Burns and Roe; it allows application of this forcing function at downcomer vent exits in a Mark II suppression pool and calculation of pool boundary loads and building responses properly accounting for fluid/structure interaction effects.

The B&R computational methodology will be discussed with NRC in early November. The documentation required in support of the chugging forcing function definition and the computational methodology is scheduled for submittal for NRC review in the WNP-2 Design Assessment Report, revision 2.

The current safety relief valve quencher load definition in the DFFR is very conservative and results in high building response loads for WNP-2. WPPSS is currently involved in a program that will utilize existing test data and the data from the ongoing Caorso safety relief valve tests to develop a more realistic SRV quencher load definition. As discussed in the NRC-WPPSS meeting of October 10, 1978, a report covering SRV load definition for WNP-2 is scheduled to be available for NRC review in May, 1979.

Not sent
WPPSS 11/7/78

Q 022.030

Provide a more detailed description of the significant modifications to the WNP-2 facility which are being made or scheduled to be made based upon the results of your ongoing experimental and analytical efforts.

Response

Modifications made to the WNP-2 plant as a result of investigations of the SRV and LOCA phenomenon are listed below:

1) Seven horizontal ring tee stiffeners added to the submerged circumference of the steel containment vessel.

2) Redesign of the downcomer bracing system from a system of radial beams to a pipe truss system which includes braces for the suppression pool columns and lateral restraints for the SRV piping.

3) Revised location of platforms in the suppression pool and revision of their connection to the containment vessel.

4) Revised location of wetwell to drywell vacuum breakers.

5) Provided a quencher discharge device and support tower for each SRV line. Added a redundant vacuum breaker on each SRV discharge piping line. SRV piping was rerouted to optimize the line air volumes.

6) Provided additional stiffening in the area of many containment vessel piping penetrations.

7) Redesign of piping systems resulting in additional pipe supports and snubbers.

8) Downcomer pipes were locally reinforced where the SRV pipe penetrates the downcomer wall and the downcomer flanges were removed.

9) Installation of a suppression pool temperature monitoring system.

10) Revised the design of piping suction strainers.

RESPONSES TO CORE PERFORMANCE BRANCH QUESTIONS*

*per letter, NRC to WPPSS, Sept. 18, 1978.

QUESTION 231.001 (4.2)

Section 4.2 of the WNP-2 FSAR references the General Electric Topical Report, NEDO-20944, as the sole input for fuel design. During our review of this GE topical, another topical report "BWR/4 and BWR/5 Fuel Design, Amendment 1," NEDO-20944-1P dated January 1977, was submitted. It is our position that this report is applicable to the WNP-2 facility. Accordingly, revise your FSAR to include both G.E. topical reports.

RESPONSE

NEDE-20944-1P (Amendment One to NEDE-20944-P) is applicable to WNP-2 and ~~should~~ will be referenced.*

* See attached draft.

4.2 FUEL SYSTEM DESIGN

Information covering the following subjects in 4.2 are provided in topical report NEDO-20944*. Proprietary information is contained in NEDE-20944-P** WNP-2 is a BWR 5 with 251 inch vessel, 764 fuel assemblies, loaded on C lattice. Topical report paragraph, table and figure numbers are consistent with FSAR numbers, except that the initial digit (4) has been suppressed in the topical numbering.

4.2 FUEL SYSTEM DESIGN

4.2.1 General and Detailed Design Bases

4.2.1.1 General Design Bases

4.2.1.2 Detailed Design Bases

4.2.2 General Design Description

4.2.2.1 Core Cell

4.2.2.2 Fuel Assembly

4.2.2.3 Fuel Bundle

4.2.2.4 Reactivity Control Assembly

4.2.3 Design Evaluations

4.2.3.1 Results of Fuel Rod Thermal-Mechanical Evaluations

4.2.3.2 Results from Fuel Design Evaluations

4.2.3.3 Reactivity Control Assembly Evaluation (Control Rods)

4.2.4 Testing and Inspection

4.2.4.1 Fuel, Hardware and Assembly

4.2.4.2 Testing and Inspection (Enrichment and Burnable Poison Concentrations)

4.2.4.3 Surveillance Inspection and Testing of Irradiated Fuel Rods

4.2.5 Operating and Developmental Experience

4.2.6 References

** *NEDO-20944, "BWR 4 and BWR 5 Fuel Design," October 1976, NEDE-20944P - Proprietary Version, Including NEDE-20944-1P (Amendment 1 to NEDE-20944-P).

QUESTION 231.002 (4.2)

The NRC staff is concerned with the validity of fission product gas release calculations in most fuel performance codes, including GEGAP-III, for a fuel burnup greater than 20,000 MWd/tU. General Electric was informed of this concern on November 23, 1976, and was provided with a method of correcting fission product gas release calculations for fuel burnups greater than 20,000 MWd/tU. Since there was no question of the adequacy of GEGAP-III for fuel burnups below 20,000 MWd/tU, your calculations are acceptable only for that time in reactor core life when the peak local burnup is less than 20,000 MWd/tU. For fuel burnups in excess of this specific value, GEGAP-III calculations and all other affected analyses, must be redone using the correction cited above. Alternatively, you may submit a modified method which addresses the staff's concerns.

RESPONSE

NRC concern regarding the validity of fission gas release calculations for burnup greater than 20,000 MWd/tU was transmitted to General Electric in Reference 1. Reference 1 requested an analysis to describe the impact of higher fission gas release for G.E. operating power reactors (BWR 2-4 product line). Reference 1 did not indicate that such analyses would be necessary for licensing support for operating plants or that the NRC would require the application of the fission gas release correction factor in future analyses. Hence, the use of the correction factor is not part of the design-basis analysis.

Reference 2 provided G.E.'s response to the NRC request. The NRC fission gas release correction was employed to modify the GEGAP(3) thermal performance code. The modified GEGAP code was then employed to calculate the following parameters as a function of exposure for 7x7 and 8x8 fuel:

(1) Percent of fission gas released
(2) Fuel rod internal pressure
(3) Pellet-to-cladding gap conductance at the peak power axial position
(4) Fuel centerline temperature at the peak power axial position
(5) Fuel volume average temperature at the peak power axial position

These parameters have been compared with results of the standard (unmodified) GEGAP code in Reference 2.

The only affected safety analyses, as indicated in Reference 2, were the loss-of-coolant analyses. Although the calculations were not specifically performed for the WNP-2 fuel, the 8x8 analysis performed for early reflooding plants will bound the WNP-2 case. Consequently, based on the results indicated in Reference 2, the NRC fission gas release correction results in less than an 85°F increase in calculated peak cladding temperature at a target planar average exposure of 30,000 MWd/t (The WNP-2 initial core is not expected to exceed 20,000 MWd/t).

References: 1. Ross, Denwood F., letter to Dr. Glen Sherwood, November 23, 1976

2. Sherwood, G. G., letter to Denwood F. Ross, December 22, 1976

3. "GEGAP III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," NEDO-20181, November 1973

QUESTION 232.001 (15.4.1)

General Electric has performed a generic analysis of the consequences of the continuous withdrawal of an out-of-sequence control rod during reactor startup. This analysis has been documented on the Hatch-2 docket (Docket No. 50-366). Adopt this analysis either by reference or submit it in its entirety on your docket.

RESPONSE

The detailed analysis of the consequences of a RWE in the startup range is provided in NEDM-23842, "Continuous Control Rod Withdrawal Transient in the Startup Range," April 18, 1978 by R. C. Stirn and J. F. Klapproth.

RESPONSES TO GEOSCIENCES BRANCH QUESTIONS*

*per letter, NRC to WPPSS, Sept. 18, 1978.

(1) one copy of the references is attached to the original of Question 362.3

360.0 GEOSCIENCES BRANCH

362.0 Geotechnical Engineering Section

Q. 362.1
(2.5.4.5.3) and (2.5.H2.3)

Provide summaries of field test results which support the statement on page 2.5-137 of the FSAR that "... the relative density values were within the specified limits..."

RESPONSE:

Summaries and an evaluation of the compacted fill placed for support of all Seismic Category I Structures, except the Condensate Storage Tanks, are provided in Reference 2.5-127 ("Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System Nuclear Project No. 2, Benton County, Washington" by Shannon & Wilson, Inc., May 11, 1976). Relative density test summaries are presented on Figures B-2, C-3, C-4, C-5, and C-6 of Reference 2.5-127. An evaluation and summary of relative density tests for the fill placed to support the Condensate Storage Tanks is provided by Tables A, B, C, and D in Reference 2.5-127A ("Soil Compaction Evaluation Quality Class I Backfill Condensate Storage Tanks Area, WPPSS Nuclear Project No. 2, Benton County, Washington", by Shannon & Wilson, Inc., February 15, 1977). In both references, the evaluations and summaries demonstrate that the degree of fill compaction was within the specification requirements for relative density and is adequate for support of all Seismic Category I structures. References 2.5-126, 127, and 127A are attached in response to Question 362.3.

Q. 362.2
(2.5.4.5.2) and (2.5H.2.1)

On page 2.5H.2, third paragraph, line 10, revise your requirement for void ratio to read "A void ratio of less than 0.25 was specified"[1]. Provide justification for this upper limit, including its relationship to the potential for liquefaction and/or settlement[2]. In particular, demonstrate that the top of the Ringold gravel was acceptably dense prior to placement of construction fill[3]. Describe the excavation testing procedures recommended by the Foundation Engineer in Section 4.11a of Appendix 2.5F of the FSAR[4].

RESPONSE:

[1]On page 2.5H.2, third paragraph, line 10 should read: "A void ratio equal to or less than 0.25 was specified."*

[2]The upper void ratio limit of 0.25 was selected because it represents a very dense soil (Glacial till, very mixed-grained), as listed on "Soil Mechanics in Engineering Practice" by Terzaghi and Peck, July 1955 (reproduced in Table 3, Appendix A, reference 2.5-127).

The data presented on Figure 2.5H-2 of the WNP-2 FSAR demonstrates that Ringold gravels do not fall within the gradation range of soils susceptible to liquefaction. Also, very dense granular soils are not subject to liquefaction and/or excessive settlement.

[3]All density tests to determine void ratio were taken near the surface of the proof-rolled Ringold gravel after completing six coverages with the specified roller. All void ratio determinations met the specification requirements, thereby demonstrating that the surface of the Ringold gravel was acceptably dense prior to placement of construction fill.

[4]As the general excavation in the central plant area approached final grade, close observation by the engineer was required to verify that the very dense Ringold gravel had been reached. This was accomplished as follows:

a) The excavation was extended to an elevation just lower than those at which the very dense Ringold gravel had been encountered in the previous test borings. Between borings, similar gravel was exposed.

b) The very dense Ringold gravel exhibits a distinct tan or light gray-brown color. In contrast, the

* see attached draft.

overlying sands and gravels are gray or black. In all cases, excavation was continued until the distinct tan or light gray-brown soil was reached, which further demonstrated that the very dense Ringold gravel had been encountered.

c) Due to the very dense nature of the Ringold gravel compared to the overlying sand and gravel, there was a noticeable and distinct increase in the degree of difficulty required for excavation. Also the movement of the construction equipment over the surface of the exposed Ringold gravel would produce a very dense stable surface compared to the overlying soils which would exhibit a loose and unstable condition. Probing into the very dense Ringold gravel with a pointed steel rod reached refusal within a few inches. further demonstrating that the very dense gravel zone had been reached.

d) Testing to determine the in situ density and corresponding void ratio was performed in accordance with specification requirements to further demonstrate that the very dense Ringold gravel had been encountered.

Dewatering during excavation in this area was accomplished with a perforated 55-gallon oil drum installed in the deepest portion of the excavation for a pump sump. Pumping continued during excavation in order to observe and determine that the excavation extended to the Ringold gravel and permit the removal of all loose and medium dense sandy soils and/or disturbed Ringold gravel.

Pumping continued after the excavation was complete and the area was backfilled except for the small area in the vicinity of the sump. The pump and perforated oil drum were subsequently removed and the small area backfilled quickly in order to keep placement ahead of the rising groundwater. Lift thickness and compaction were performed in accordance with specification requirements, but coarser than average sandy material was used to provide maximum stability. Backfilling in this area was carried at least two to three feet above the static groundwater level before this phase of backfilling was stopped.

The very dense Ringold gravel encountered in the bottom of the excavation was identified by its gradation and color. Whereas the upper loose to medium dense sand is dark gray to black and composed mainly of basalt grains, the Ringold gravel is tan to brown, grades to 3-inch maximum size and contains quartzitic sand. Typical gradations of Ringold gravel, based on samples retrieved from the base of the central plant excavation, are shown on Figure 2.5H-2. In addition, the specifications required identification of the Ringold gravel also be based on void ratio. A void ratio equal to or less ~~measurement of no less~~ than 0.25 was specified. Table 2.5-1 presents results of the 12 void ratio measurements taken in the base of the WNP-2 central plant excavation. As shown, the maximum void ratio, e, measured was 0.20, and the average was 0.17.

2.5H.2.1.1 Geologic Mapping of Excavation Slopes

Excavation in the central plant area was accomplished in two phases, as shown on Figures 2.5H-3 and 2.5H-4. Detailed geologic profiles of the slopes developed from the mapping are presented on Figures 2.5H-5 and 2.5H-6.

Geologic mapping of the excavation slopes was performed by geologists from Shannon and Wilson, Inc. The slopes of the initial excavation were mapped intermittently during the period of November 27, 1972 through January, 1973. The slopes of the final excavation were mapped during the period from May 7, 1973 through May 11, 1973.

Q. 362.3
(2.5.8)

Provide references 2.5-126 and 2.5-127.

RESPONSE:

Reference 2.5-126 "Letter Report, Soil-Structure Interaction, Washington Public Power Supply System Hanford No. 2 Nuclear Station Richland, Washington" by E. D'Appolonia Consulting Engineers, Inc., August, 1972, is enclosed. Reference 2.5-127 "Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System Nuclear Project No. 2 Benton County, Washington" by Shannon and Wilson, Inc., May 11, 1976 is also enclosed. Reference 2.5-127A "Soil Compaction Evaluation Quality Class I Backfill Condensate Storage Tanks Area WPPSS Nuclear Project No. 2 Benton County, Washington" by Shannon and Wilson, Inc., February 15, 1977, is also included and is hereby incorporated as a separate reference.

Q. 362.4
(2.5.4.10)

Describe the methods used to calculate the dynamic coefficient (K_D) of 0.3 shown in Figure 2.5-69 of the FSAR. Indicate how the effects of compaction are included in the calculations of lateral earth pressures.

RESPONSE:

The dynamic coefficient (K_D) was calculated from the Mononabe-Okabe analysis (Seed and Whitman, 1970). Additional items considered in our calculation included:

1) The assumption that the walls of the Seismic Category I structures would be rigid with resulting at-rest pressures.

2) An assessment of model test results performed by Japanese investigators and summarized in Seed and Whitman, 1970.

3) A review of coefficients required by building codes around the world as summarized in Seed and Whitman, 1970.

The distribution of lateral earth pressure included on Figure 2.5-69, and discussed in Appendix 2.5F was based on experimental data presented in Seed and Whitman, 1970, and Aggour, 1972.

The effects of compaction on the static and dynamic earth pressures are somewhat offsetting. Heavy compactive effort is expected to increase the static pressure but would result in a decreased dynamic pressure (see Figure 13 and 19 in Seed and Whitman, 1970). On the other hand, lesser compactive effort would result in lower static pressure, and potentially a higher pressure under dynamic loading. Therefore, the static and dynamic coefficients presented in Figure 2.5-69 allow for any effects of compaction since they would take into account both the case of a less compacted material with a potentially higher pressure under dynamic loading or a denser compacted material with a corresponding lower pressure under dynamic loading.

References

Seed, H. B., and Whitman, R. V., 1970, "Design of Earth Retaining Structures for Dynamic Loads", Specialty Conference, Lateral Stress in the Ground and Design of Earth-Retaining Structures, Soil Mechanics and Foundations Division, ASCE.

Aggour, M. S. 1972, "Retaining Walls in Seismic Areas", Dissertation submitted in partial fulfillment of the requirements for the degree of Doctor of Philosophy, University of Washington, Seattle, Washington.